ENCLOSURE 2

SHINE MEDICAL TECHNOLOGIES, LLC

SHINE MEDICAL TECHNOLOGIES, LLC OPERATING LICENSE APPLICATION RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION AND SUPPLEMENT NO. 7

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION PUBLIC VERSION

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RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION PUBLIC VERSION

The U.S. Nuclear Regulatory Commission (NRC) staff determined that additional information was required (Reference 1) to enable the continued review of the SHINE Medical Technologies, LLC (SHINE) operating license application (Reference 2). The following information is provided by SHINE in response to the NRC staff's request.

Chapter 9 – Auxiliary Systems

RAI 9-1

SHINE FSAR Section 9a2.1, "Heating, Ventilation, and Air Conditioning (HVAC)," provides a description of the facility radiological ventilation (RV) systems including supply air, recirculating, and exhaust subsystems required to condition the air and provide the confinement and isolation needed to mitigate design basis accidents. It also includes a description of the HVAC systems serving the nonradiologically controlled areas. The SHINE facility utilizes three ventilation systems in the radiologically controlled area (RCA) to maintain the temperature and humidity of the RCA and to progress air from areas of least potential for contamination (Zone 3) to areas with the most potential for contamination (Zone 1). During its review of the application, the NRC staff determined that additional information is needed for to confirm that SHINE has satisfied the following regulations:

- In accordance with 10 CFR 20.1101, "Radiation protection programs," the licensee shall use, to the extent practical, procedures and engineering controls based upon sound radiation protection principles to achieve occupational doses and doses to members of the public that are as low as is reasonably achievable (ALARA).

- In accordance with 10 CFR 20 Subpart C, "Occupational Dose Limits," the licensee shall control the occupational dose to individual adults within dose limits.

- In accordance with 10 CFR 20 Subpart D, "Radiation Dose Limits for Individual Members of the Public," requires each licensee shall conduct operations so that the dose in any unrestricted area from external sources is maintained within specified limits.

- In accordance with 10 CFR 50.34(b)(2), the final safety analysis report shall include a description and analysis of structures, systems, and components (SSCs), with emphasis upon performance requirements, the bases upon which such requirements have been established, and the evaluations required to show that safety functions will be accomplished. The description shall be sufficient to permit understanding of the system designs and their relationship to safety evaluations.

a. SHINE FSAR Section, 9a2.1.1.3, “System Operation,” states that HEPA filters and carbon adsorbers are provided to remove airborne contaminants from the ventilation air stream. The filters and adsorbers capture airborne contaminants in Radiological Ventilation Zone 1 exhaust subsystem (RVZ1e) and Radiological Ventilation Zone 2 exhaust subsystem (RVZ2e) that may include radioactive material; however, it is not clear from the submittal how compliance with 10 CFR Part 20 will be maintained.

   Provide a discussion of how SHINE will assure compliance of the occupational dose requirements of 10 CFR Part 20 as a result of the concentration of radioactive material in the filters and adsorbers during all modes of operation and maintenance.

b. The SHINE application does not discuss potential failure modes for FVZ4, which could have adverse consequences. Specifically, the interface areas between the RCA and non-RCA areas (labyrinths, vestibule, and shipping/receiving) are supplied with ventilation air from RVZ2s. A failure of the FVZ4s (supply) subsystem with continued operation of the FVZ4e (exhaust) subsystem has the potential to lower the pressure in the non-RCA area below that in the interface areas. SHINE FSAR Figure 9a2.1-7, “Radiological Ventilation Zone 3 (RVZ3) Flow Diagram,” shows makeup air from FVZ4. There does not appear anything to prevent the transfer of air from RVZ2s through the interface areas to the non-RCA through the FVZ4 makeup flow path.

   Provide a discussion of operational and equipment failure modes to confirm there are no failure modes which could cause an inadvertent release of radioactive airborne material to the unrestricted environment which could violate 10 CFR Part 20 occupational dose limits or dose limits to the general public.

   Confirm if non-RCA areas are included in the surveys required by 10 CFR 20.1501 in the event of an inadvertent release of radioactive airborne material from a HVAC subsystem or component failure.

c. Figure 9a2.1-2, “Radiological Ventilation Zone 1 Recirculating Cooling Subsystem (RVZ1r) Flow Diagram” shows condensate collected from the Irradiation Unit (IU) Cell cooling coil being sent to the light water pool. There is no discussion regarding the condensate from the target solution vessel offgas system (TOGS) Cell cooling coil.

   Provide a discussion of the potential for condensate from the TOGS Cell cooling system to contain radioactive material, and if so, discuss how the condensate is monitored and disposed. This information is needed to ensure that there is not an inadvertent release of radioactive material to the unrestricted environment which could violate 10 CFR Part 20 occupational dose limits or dose limits to the general public.

d. The SHINE Application states that the RVZ2r recirculates, filters, and conditions air within the RCA. The system includes AHUs, filters, ductwork, and dampers. The RVZ2r units are located within the RCA. RVZ2r provides additional cooling for systems within ventilation zone 2. Figure 9a2.1-6, “Radiological Ventilation Zone 2 Supply Subsystem (RVZ2s) and Radiological Ventilation Zone 2 Recirculating Cooling Subsystem (RVZ2r) Flow Diagram,” shows multiple RVZ2r recirculation units. The application does not discuss if the air from the RVZ2r subsystems is cooled by cooling coils.

   Discuss if condensate is collected from any of the RVZ2r units and if the condensate can contain radioactive material. Include in the discussion how the condensate is monitored and

is disposed. This information is needed to ensure that there is not an inadvertent release of radioactive material to the unrestricted environment which could violate 10 CFR Part 20 occupational dose limits or dose limits to the general public.

e. Operating License Application Supplement No. 2 (ADAMS Accession Number ML20105A295) revised the tritium production method. The new method states that the Deuterium source gas exhaust is exhausted to the facility ventilation. There is no discussion in Section 3.4.5.2, “Explosion Hazards,” or in Section 9a2.3.3, “Fire Hazard Analysis,” regarding the potential for a combustible mixture of deuterium – air mixture.

   Provide a discussion regarding the potential for the formation of a combustible mixture of deuterium and air. Include in the discussion how a combustible mixture will be prevented, or if present, how ignition will be prevented under normal and anticipated degraded ventilation system operation. This information is needed to ensure that necessary safety functions will be accomplished in accordance with 10 CFR 50.34(b)(2).

SHINE Response

a. SHINE has analyzed the radiological doses from filters and adsorbers during all modes of operation and maintenance. The probable radiation area designations within the SHINE radiologically controlled area (RCA), which include the radiological doses from filters and adsorbers, are identified in Figure 11.1-1 of the final safety analysis report (FSAR), consistent with the definitions in 10 CFR Part 20.

   SHINE will assure compliance with the occupational dose requirements of 10 CFR Part 20 by following the radiation protection practices described in Section 11.1 of the FSAR. Maintenance activities, including filter replacement, will be performed in accordance with the radiation protection program practices described in Section 11.1 of the FSAR. Subsection 11.1.5.5 of the FSAR discusses the use of external dosimetry devises when personnel are working in restricted areas, including the use of supplemental dosimetry with dose and dose rate alarm capability when any individual is entering a High Radiation or Very High Radiation Area. Subsection 11.2.5.6 of the FSAR discusses how SHINE will comply with occupational dose exposures due to internal sources.

b. The facility ventilation zone 4 (FVZ4) system is a nonsafety-related system, as described in Subsection 9a2.1.2.2 of the FSAR; therefore, the FVZ4 system does not perform a function to prevent radiological releases that could cause undue risk to health and safety of workers and the public. There are no FVZ4 failure modes which could cause an inadvertent release of radioactive airborne material to the unrestricted environment which could violate 10 CFR Part 20 occupational dose limits or dose limits to the general public.

   The RCA ventilation pressure gradients create flow patterns that direct air towards areas of increasing contamination potential, as described in Subsection 9a2.1.1.3 of the FSAR. As such, the radiological ventilation zone 2 (RVZ2) system does not supply the interface areas (e.g., labyrinths, vestibule, and shipping/receiving) between the RCA and non-RCA areas with ventilation air. The FVZ4 system provides transfer air into the radiological ventilation zone 3 (RVZ3) system interface areas (e.g., labyrinths, vestibule, and shipping/receiving), as described in Subsection 9a2.1.2.2 of the FSAR. The RVZ3 system transfers air from ventilation zone 4 to ventilation zone 3, then from ventilation zone 3 to ventilation zone 2 via engineered pathways. The safety related design of the RVZ3 system inhibits backflow within

ductwork that could spread contamination into FVZ4 or non-RCA areas as described in Subsection 9a2.1.1.2 of the FSAR.

Non-RCA areas are included in the surveys required by 10 CFR 20.1501 in the event of an inadvertent release of radioactive airborne material into non-RCA areas from a ventilation system or component failure.

c. Condensate from the target solution vessel (TSV) off gas system (TOGS) cell radiological ventilation zone 1 recirculation subsystem (RVZ1r) cooling coil is not expected. Two piping passthroughs exist between each irradiation unit (IU) cell and TOGS cell wall. One passthrough is an open pipe that allows for the free transport of gases between the two cells. The other passthrough contains piping that draws ambient air from the TOGS cell atmosphere through the primary closed loop cooling system (PCLS) expansion tank, and exhausts to radiological ventilation zone 1 exhaust subsystem (RVZ1e) due to the negative source of pressure that RVZ1e provides. Air is made up to the primary confinement via cover plug seal in-leakage. Condensate generation from the cover plug seal in-leakage is not expected because the TOGS cell cooling coil removes sensible heat only. IU cell air transferred to TOGS is preconditioned by the IU cell RVZ1r cooling coil. The TOGS RVZ1r cooling coil is maintained above the TOGS cell air dewpoint.

   The TOGS RVZ1r cooling coil is provided with a drip pan and drip pan heating elements in the event small amounts of condensate were to form. The RVZ1r heating elements are designed to operate while the unit fans circulate air. Additionally, the RVZ1r TOGS units are designed with a low point drain where fluid can be removed from the watertight unit, tested, and processed as radiological waste if required during maintenance operations.

d. Condensate is not collected from the radiological ventilation zone 2 recirculation subsystem (RVZ2r) units during normal operations. Condensate is not generated because the system functions to remove sensible heat only, at temperatures above the dewpoint, from the RCA preconditioned airstream. The radiological ventilation zone 2 supply subsystem (RVZ2s) air handling unit (AHU) conditions outside air by removing moisture prior to supply into the RCA. The RVZ2r subsystem cooling coil is maintained at a temperature above the RCA air dewpoint to preclude generation of condensate within the AHU.

e. Deuterium source gas is exhausted to the RVZ1e line in the IU cell that also ventilates the PCLS expansion tank. Once the deuterium source gas enters the RVZ1e, it mixes with the hydrogen generated by radiolysis in both the PCLS cooling water and the light water pool. The deuterium and hydrogen is diluted by the air being drawn from the IU cell atmosphere by RVZ1e. The deuterium source gas flowrate is less than [ ]$^{PROP/ECI}$ of the maximum hydrogen generation rate in the PCLS and light water pool.

   During normal ventilation system conditions, formation of flammable mixtures of deuterium source gas and other sources of hydrogen in the IU cell is prevented by the forced RVZ1e ventilation, which maintains hydrogen concentration in the IU cell atmosphere and RVZ1e exhaust stream below 1 percent hydrogen by volume, as described in Subsection 5a2.2.7 of the FSAR.

   During degraded ventilation system conditions resulting in a loss of RVZ1e flow, ignition in the PCLS expansion tank and RVZ1e ductwork of deuterium source gas and hydrogen generated by PCLS is prevented by the PCLS flame arrestor. The PCLS flame arrestor is

described in Table 5a2.2-2 of the FSAR. An accident scenario involving the potential deflagration of hydrogen in the PCLS and the PCLS flame arrestor is described in Subsection 13a2.1.9.2 of the FSAR.

During degraded ventilation system conditions resulting in a loss of RVZ1e flow, ignition of deuterium source gas in the IU cell atmosphere is prevented by limiting the amount of material at risk. Deuterium gas is stored in containers with quantities small enough to prevent reaching the lower flammability limit for hydrogen in the IU cell atmosphere in the event of the entire contents of the storage container enter the IU cell. Subsection 2.2.3.1.2.4 of the FSAR describes storage of deuterium at SHINE.

RAI 9-2

Section 9b.7.2 “Material Handling System,” of SHINE FSAR includes description of overhead cranes and hoists that are used to move or manipulate radioactive material within the radiologically controlled area.

The main emphasis in the material load handling system review is on critical load handling where inadvertent operations or equipment malfunctions, separately or in combination, could cause a release of radioactivity. In accordance with 10 CFR 50.34(b)(2), as appropriate, the final safety analysis report shall include a description and analysis of auxiliary and fuel handling structures, systems, and components (SSCs), with emphasis upon performance requirements, the bases upon which such requirements have been established, and the evaluations required to show that safety functions will be accomplished. The description shall be sufficient to permit understanding of the system designs and their relationship to safety evaluations.

As discussed in Section 9b.7.2 “Material Handling System” of SHINE application, the ASME NOG-1 (Type I) overhead crane located in Irradiation Facility (IF) is designed to secure its load in place upon a loss of power and any fault condition. The hoisting machinery and wire rope system, in addition to other affected components, are designed to withstand the most severe potential overload, including two-blocking and load hang-up.

With use of a crane designed to the criteria of ASME NOG-1 for a Type 1 crane, a likelihood of failure is extremely low due to the single failure-proof handling features. In addition, Section 5.1.1 of NUREG-0612 provides certain guidance on slings and special handling devices. As indicated in NUREG-0612, the handling system for use with single-failure-proof design should include lifting devices that should be selected to satisfy either of the following criteria:

- A special lifting device that satisfies ANSI N14.6 should be used for recurrent load movements in critical areas (reactor head lifting, reactor vessel internals, spent fuel casks). The lifting device should have either dual, independent load paths or a single load path with twice the design safety factor specified by ANSI N14.6 for the load.

- Slings should satisfy the criteria of ASME B30.9 and be constructed of metallic material (chain or wire rope). The slings should be either (a) configured to provide dual or redundant load paths or (b) selected to support a load twice the weight of the handled load.

While SHINE commits to these ANSI and ASME codes, the staff is unable to locate any details or description of lifting devices, sling selection criteria, rigging design (i.e., redundant load path or double factor of safety), and sling attachment to the load.

In addition to above, the staff noted inconsistent versions of the ASME NOG-1 code were referenced in the application:

- Section 3.4.2.6.4.6 “Crane Load” of SHINE application, the building is evaluated for loads associated with two overhead bridge cranes, one servicing the IU cell area and one servicing the RPF area. Crane loading is evaluated in accordance with American Society for Mechanical Engineers (ASME) NOG-1, Rules for Construction of Overhead and Gantry Cranes (ASME, 2004).

- However, Section 9b.7.2 “Material Handling System” commits to single failure proof crane design in accordance with the ASME NOG-1, Rules for Construction of Overhead and Gantry Cranes (Top Running Bridge, Multiple Girder) (AMSE, 2015).

a. Provide additional details on special lifting devices, having redundant load paths or double the normal factors of safety, the particular components associated with special lifting devices, and details of the interfacing lift points for these components consistent with SHINE’s commitments applicable ANSI and ASME codes.

b. Verify which version of ASME NOG-1 code is applied to the SHINE design.

This information is necessary for the NRC staff to determine that systems, components, and methods for handling, moving, or storing components are adequately designed to prevent mechanical damage that could significantly decrease integrity of facility safety-related SSCs or release fission products.

SHINE Response

a. The SHINE facility has one single-failure-proof overhead bridge crane, designed in accordance with American Society of Mechanical Engineers (ASME) NOG-1-2004, “Rules for Construction of Overhead and Gantry Cranes (Top Running Bridge, Multiple Girder)” (Reference 3), which is located in the irradiation facility (IF). The overhead bridge crane located in the radioisotope production facility (RPF) is not designed as single-failure-proof.

   The anticipated recurrent load movements within critical areas of the IF do not require the use of specially designed lifting devices; therefore, SHINE has not designed special lifting devices as defined in NUREG-0612, “Control of Heavy Loads at Nuclear Power Plants” (Reference 4). As such, the details for special lifting devices, the particular components associated with special lifting devices, and details of the interfacing lift points for these components, are not currently applicable. If special lifting devices are developed, Subsection 9b.7.2.3 of the FSAR states that special lifting devices used in the vicinity of safety-related structures, systems, and components (SSCs) will satisfy the guidelines of American National Standards Institute (ANSI) N14.6, “Radioactive Materials - Special Lifting Devices for Shipping Containers Weighing 10,000 Pounds (4500 kg) or More” (Reference 5), consistent with the guidance of NUREG-0612. Subsection 9b.7.2.3 of the FSAR also states that lifting devices that are not specially designed will be installed and used in accordance with the guidelines of ASME B30.9, “Slings” (Reference 6).

   For cranes operating in the vicinity of safety-related SSCs, SHINE has applied the guidance from NUREG-0612 as stated in Subsection 9b.7.2.3 of the FSAR, including procedures to cover load handling operations for heavy loads. These procedures include requirements to

ensure that applicable lifting devices have redundant load paths or double the normal factors of safety, consistent with the guidance of Sections 5.1.1 and 5.1.6 of NUREG-0612.

b. The code year applied to the SHINE design is ASME NOG-1-2004 (Reference 3). SHINE has revised Subsection 9b.7.2.1 and Section 9b.8 of the FSAR to correct the referenced ASME NOG-1 code year. A mark-up of the FSAR incorporating these changes is provided as Attachment 1.

Chapter 11– Radiation Protection Program and Waste Management

RAI 11-1

Section 11.1, “Radiation Protection,” of the NUREG-1537, Part 2, states that acceptance criteria for information concerning sources of radiation should include the following:

- Conservative best estimates of the predicted concentrations, locations, and quantities of airborne radionuclides during the full range of normal operation in areas occupied by personnel.

- Conservative best estimated of the predicted locations and magnitude of external radiation fields during the full range of normal operation in areas occupied by or accessible to personnel.

Section 11.1.1, “Radiation Sources” of the SHINE FSAR, page 11.1-3 states that, “Shielded vaults, cells, and rooms designated as high radiation areas or very high radiation areas as denoted in Figure 11.1-1...” However, it appears that Figures 11.1-1 and 11.1-2 do not have room designations (labels for rooms) that inform the staff of what equipment may be in the rooms.

Provide updated versions of Figures 11.1-1 and 11.1-2 in the FSAR with labeled rooms with identified equipment that are sources for designation as high and very high radiation areas.

SHINE Response

SHINE has revised Figures 11.1-1 and 11.1-2 of the FSAR to include room labels. A mark-up of the FSAR incorporating these changes is provided as Attachment 1. The following equipment are the sources for designation as high and very high radiation areas within the applicable rooms:

- IUs within the IU cells
- TOGS skids within the TOGS cells
- Radioactive liquid waste immobilization (RLWI) equipment within the RLWI enclosure
- Equipment within the supercell, including process vessel vent system (PVVS), vacuum transfer system (VTS), iodine and xenon purification (IXP) system, molybdenum extraction and purification system (MEPS), and molybdenum isotope product packaging system (MIPS)

RAI 11-2

10 CFR 50.34(b)(3) states that the FSAR shall include the kinds and quantities of radioactive material expected to be produced in the operation of the facility as it relates to tracking the amount of radioactive material contained in the various facility components as well as the anticipated radionuclides inside and outside the plant environment. In addition, NUREG-1537 Part 2, Section 11.1.1, “Radiation Sources,” states that applicant should identify the “quantities and concentrations expected to be released.” 10 CFR 20.1101(b) states that the applicant shall use, to the extent practical, procedures and engineering controls based upon sound radiation protection principles to achieve occupational doses and doses to members of the public that are as low as is reasonably achievable (ALARA).

Section 11.1.1, “Radiation Sources,” of the SHINE FSAR, page 11.1-2 states that the normal operation internal facility radiation dose rates are consistent with ALARA principles and that the “dose rates were calculated using the maximum specified shield plug gap sizes, minimum density shielding materials, and the nominal inventories for full power operation.”

a. In order for NRC staff to determine whether SHINE’s dose rate estimates are appropriate for meeting 10 CFR 20.1101(b) and consistent with ALARA principles, provide a summary of the calculations/evaluations, assumptions, methodology, and input parameters that resulted in the estimated dose rates. The current information provided in FSAR section 11.1 tables generically provides a total curie content for components but does not provide enough details for the staff to perform an independent evaluation. Provide FSAR tables that provide the expected activities by isotope, in the components located in the SHINE facility to demonstrate compliance with 10 CFR 50.34(b)(3). The tables that staff are referring to include: Tables 11.1-5, 11.1-9, and 11.1-10. In addition, the staff requests the applicant provide a table that summarizes the volumes in components such as tanks and systems assumed for dose calculations and provide thickness for tanks and pipes to allow staff to verify the stated radiation zoning in FSAR section 11.1. If other assumptions were necessary for the dose calculations provide that information within the response to this question.

Section 11.1.1.1, “Airborne Radioactive Sources,” of the SHINE FSAR, page 11.1-4 states that SHINE maintains airborne radioactive material at very low concentrations in normally occupied areas and that systems are designed to protect workers in keeping with the ALARA principles of 10 CFR Part 20.

b. In order for NRC staff to determine whether SHINE airborne doses to occupational workers are consistent with the ALARA principles, provide a summary of the calculations/evaluations, assumptions, methodology, and input parameters for the calculated expected doses rates, include DAC estimates, from gaseous radioactive sources presented in Section 11.1.1.1.

Section 11.1.1.1., “Airborne Radioactive Sources,” of the SHINE FSAR, page 11.1-5 states only nuclides with greater than 1 Ci/year released are included in Table 11.1-8.

c. Describe why SHINE limited the nuclides in Table 11.1-8 to only nuclides with greater than 1 Ci/year, which appears contrary to 10 CFR 50.34 and the guidance in NUREG-1537. Update the FSAR and Table 11.1-8 to include the quantities expected to be released, as necessary.

Consistent with the evaluation findings in Section 11.1.1 of NUREG-1537, Part 2, the information requested in parts a., b., and c. of this RAI is necessary for the NRC staff to confirm that the FSAR identifies potential radiation safety hazards associated with the SHINE facility and conduct an independent review of the SHINE radiation protection program.

SHINE Response

a. SHINE has revised Tables 11.1-5, 11.1-9, and 11.1-10 of the FSAR to provide the conservative best estimate activities by isotope within the components located in the SHINE facility. A mark-up of the FSAR incorporating these changes is provided as Attachment 1.

The information provided in Sheet 1 of Tables 11.1-5, 11.1-9, and 11.1-10 of the FSAR includes the estimated maximum activity for specified components based on the safety basis operating conditions described in Table 11.1-1 of the FSAR. As discussed in Subsection 11.1.1 of the FSAR, isotope inventories based on the safety basis operating conditions are unsuitable for use in analyzing normal operations. As such, the exterior dose rate estimates provided in Sheet 1 of Tables 11.1-5, 11.1-9, and 11.1-10 of the FSAR are based on conservative best estimate activities within the specified components, consistent with Part 2 of NUREG-1537 (Reference 7). The conservative best estimate activities by isotope provided in subsequent sheets of Tables 11.1-5, 11.1-9, and 11.1-10 of the FSAR are based on nominal operating conditions as described in Table 11.1-1 of the FSAR.

A summary of the methodology used for shielding calculations in the IF and the RPF are provided in Subsections 4a2.5.3.1 and 4b.2.3.1 of the FSAR, respectively. Additional information regarding calculation assumptions and input parameters, which resulted in the estimated dose rates provided in Tables 11.1-5, 11.1-9, and 11.1-10 of the FSAR, is described below for each of the SHINE facility locations listed in the tables.

Table 11.1-5, Airborne Radioactive Sources

TPS; tritium purification system; TPS gloveboxes
The estimated maximum activity for the tritium purification system (TPS) is 300,000 Ci of tritium (including tritium within the neutron driver assembly system [NDAS] units) as listed in Table 11.1-5 of the FSAR. There is no direct dose contribution from tritium because tritium beta decays with low energy. The beta radiation is shielded by the process piping and tanks that contain the tritium. The dose rate estimate provided in Table 11.1-5 of the FSAR includes contributions from the derived air concentration (DAC) as stated in the table footnotes. The SHINE Response to RAI 11-2b discusses DAC estimates.

NDAS; driver vacuum hardware; IU cell
The estimated maximum activity for a single NDAS unit is [ ]$^{PROP/ECI}$ Ci of tritium as listed in Table 11.1-5 of the FSAR. There is no direct dose contribution from tritium because tritium beta decays with low energy. The beta radiation is shielded by the process piping and tanks that contain the tritium. The dose rate estimate provided in Table 11.1-5 of the FSAR includes contributions from the DAC as stated in the table footnotes. The SHINE Response to RAI 11-2b discusses DAC estimates.

TOGS; off-gas piping, zeolite beds; TOGS shielded cell
The TOGS isotopic activity inventory is provided in Sheet 2 of Table 11.1-5 of the FSAR. The isotopic activity inventory accounts for greater than 99 percent of the expected activity. The inventory of airborne nuclides in the TOGS skid, within the TOGS shielded cell, is based on the nominal operating conditions as described in Table 11.1-1 of the FSAR, accounting for continuous gas removal from the target solution during operation and the fractional amount of gas in the TOGS cell compared to the rest of the airspace in the primary system boundary (PSB). Tanks and piping are not credited in the shielding analysis. The geometry and configuration of the TOGS cell shielding credited in the shielding analysis is described in Subsection 4a2.5.2.2 of the FSAR.

RVZ1; IU cell atmosphere and PCLS; IU cell
The estimated maximum activity in each IU cell atmosphere and PCLS is 1E-05 Ci of Ar-41 and 10 Ci of N-16 as listed in Table 11.1-5 of the FSAR. The inventory of airborne nuclides in the IU cell airspace and PCLS in the IU cell is based on the nominal operating conditions

as described in Table 11.1-1 of the FSAR, accounting for all generated N-16 in the PCLS and light water pool system (LWPS) and all activated Ar-41 in the IU cell air. The direct dose contribution in normally occupied areas from these radionuclides is not evaluated because the Ar-41 and the N-16 in the IU cell airspace and light water pool are shielded by the IU cell and the N 16 in PCLS piping and tanks is primarily under water which provides additional shielding. The geometry and configuration of the IU cell shielding credited in the shielding analysis is described in Subsection 4a2.5.2.2 of the FSAR.

RVZ1; supercell atmosphere; supercell gloveboxes
The activity inventory of airborne nuclides in the supercell atmosphere due to process leakage is provided in Sheet 3 of Table 11.1-5 of the FSAR. The inventory of airborne nuclides in the supercell atmosphere is based on the nominal operating conditions accounting for decay as described in Table 11.1-1 of the FSAR and accounting for production material extraction. The isotope inventory within the extraction, purification, and PVVS cells of the supercell accounts for one batch worth of release to the respective cell. The isotope inventory in Sheet 3 of Table 11.1-5 of the FSAR accounts for greater than 99 percent of the estimated activity in the supercell atmosphere. The airborne nuclides in the extraction and purification atmosphere do not significantly add to the estimated dose rates in normally occupied areas from the liquid and solid sources in the supercell described in Tables 11.1-9 and 11.1-10 of the FSAR, respectively. The geometry and configuration of the supercell shielding is described in Subsection 4b.2.2.2 of the FSAR.

PVVS and VTS; PVVS and VTS piping; pipe trenches, valve pits, and PVVS hot cell
The isotopic activity inventory of the PVVS and VTS in the pipe trenches, valve pits, and PVVS hot cell is provided in Sheet 4 of Table 11.1-5 of the FSAR. The isotopic activity inventory accounts for greater than 99 percent of the expected activity. The activity inventory is for the entire PVVS and VTS systems. The inventory of airborne nuclides in the PVVS and VTS is based on the nominal operating conditions as described in Table 11.1-1 of the FSAR, accounting for additional decay period and loading of carbon delay beds and iodine guard beds. Loading of the carbon delay beds is based on the filling of the TSV; decay periods are applied which account for expected bed loading periods. The vessels housing the isotopes are not credited in the shielding analysis. All the krypton and xenon is assumed to be in the carbon delay beds in the carbon delay bed vault, and all the iodine is assumed to be in the iodine guard beds. Dose estimates which account for the entire PVVS and VTS inventory within the carbon delay beds and iodine guard beds is bounding for the PVVS and VTS piping, pipe trenches, valve pits, and PVVS hot cell.

Table 11.1-9, Liquid Radioactive Sources

TSPS; target solution, unirradiated; target solution preparation area
Sources in the target solution preparation system (TSPS) include uranium from unirradiated target solution. Because the main radiation from the unirradiated target solution is alpha radiation, there is no direct dose contribution. Therefore, an isotopic activity inventory is not included.

SCAS; target solution in TSV (operating); IU cell
The isotopic activity inventory for the subcritical assembly system (SCAS) target solution in the TSV, while operating, is listed in Sheet 3 of Table 11.1-9 of the FSAR. The isotopic activity inventory accounts for greater than 99 percent of the expected activity per IU cell, and is based on the nominal operating conditions as described in Table 11.1-1 of the FSAR.

The target solution is modeled in a tank approximately [ ]PROP/ECI in length with an outer diameter of approximately [ ]PROP/ECI and an inner diameter of approximately [ ]PROP/ECI. The thickness of the tank is not relied on for shielding. The top of the tank is modeled under approximately 5.9 feet of water inside the IU cell. In addition to the estimated dose from the target solution isotope inventory, the NDAS is driving the subcritical multiplication in the TSV; therefore, there is a significant dose impact from the NDAS and SCAS neutrons as well as photons generated during fission. Dose rates on the IU cell walls from these sources of radiation are provided in Subsection 4a2.5.3.1 of the FSAR.

SCAS; Target solution in TSV, TSV dump tank (shutdown); IU cell
The isotopic activity inventory for the SCAS target solution in the TSV or dump tank at shutdown is listed in Sheet 3 of Table 11.1-9 of the FSAR. The isotope inventory accounts for greater than 99 percent of the expected activity per IU cell, and is based on the nominal operating conditions as described in Table 11.1-1 of the FSAR. The thickness of the tank is not relied on for shielding. The target solution is modeled in the dump tank approximately 12.5 feet underwater. IU cell shielding is described in Subsection 4a2.5.2.2 of the FSAR.

LWPS; water in the light water pool; IU cell
The estimated maximum activity in the light water pool is listed in Table 11.1-9 of the FSAR as 30 Ci of tritium per IU cell. There is no direct dose contribution from tritium because tritium beta decays. The beta radiation is shielded by the process piping and tanks that contain the tritium.

NDAS; oil in NDAS pumps; IU cell
The estimated maximum activity for the NDAS pump oil is listed in Table 11.1-9 of the FSAR as 2000 Ci of tritium per IU cell. There is no direct dose contribution from tritium because tritium beta decays. The beta radiation is shielded by the process piping and tanks that contain the tritium.

PCLS; primary cooling water in pump and piping; IU cell and primary cooling room
The isotopic activity inventory in the PCLS skid in the primary cooling room is listed in Sheet 4 of Table 11.1-9 of the FSAR. The inventory accounts for greater than 99 percent of the expected activity per IU cell. The inventory of nuclides in the PCLS skid is based on the nominal operating conditions as described in Table 11.1-1 of the FSAR, accounting for additional decay period in the N-16 delay tank and in the PCLS skid. The shielding properties of the PCLS skid and piping are not considered in the shielding analysis. The nuclide inventory is not identified for the PCLS components in the IU cell because they are shielded by the IU cell and light water pool and do not significantly contribute to the dose in normally occupied areas.

MEPS; target solution in pump, extraction column, and lift tanks; supercell
The isotopic activity inventory of target solution is listed for select radionuclides pre-extraction in Table 11.1-3 of the FSAR, which accounts for greater than 95.8 percent of the expected activity. The isotopic activities listed in this table are based on the nominal operating conditions as described in Table 11.1-1 of the FSAR for one batch of target solution. The isotopic activity inventories in the molybdenum extraction and purification system (MEPS) pump, extraction column, and lift tanks is approximately 10 percent of the isotopic inventory activities identified in Table 11.1-3 of the FSAR based on MEPS process volumes. The shielding properties of the MEPS pump, extraction column, and lift tanks are not considered in the shielding analysis. The supercell shielding is discussed in Subsection 4b.2.2.2 of the FSAR.

MEPS; Mo eluate in Mo eluate hold tank; supercell
The isotopic activity inventory for the molybdenum (Mo) eluate in the Mo eluate hold tank is listed in Sheet 5 of Table 11.1-9 of the FSAR, which accounts for greater than 99 percent of the expected activity per batch. The inventory of nuclides in the Mo eluate hold tank is based on the nominal operating conditions and decay as described in Table 11.1-1 of the FSAR, accounting for production material extraction. The shielding properties of the Mo eluate tank are not considered in the shielding analysis. The Mo eluate tank shape and placement has negligible contribution to external dose rates outside the supercell shielding. The supercell shielding is discussed in Subsection 4b.2.2.2 of the FSAR.

MEPS; Mo-99 product; supercell
The isotopic activity inventory for a single batch of molybdenum product is [ ]$^{PROP/ECI}$ Ci of molybdenum-99, which accounts for greater than 99 percent of the total activity of the product. The inventory of nuclides in the molybdenum product is based on the nominal operating conditions and decay as described in Table 11.1-1 of the FSAR, accounting for additional decay period associated with processing times, and accounting for production material extraction. The shielding properties of the product bottle are not considered in the shielding analysis. The supercell shielding is discussed in Subsection 4b.2.2.2 of the FSAR.

TSSS; target solution in target solution hold tank; tank vault
The isotopic activity inventory for the target solution hold tank is listed in Sheet 6 of Table 11.1-9 of the FSAR. The isotopic activity inventory accounts for greater than 99 percent of the expected activity. The inventory of nuclides in the target solution hold tank is based on the nominal operating conditions as described in Table 11.1-1 of the FSAR, accounting for additional decay period and production material extraction. The target solution is modeled in an annular tank with an inner solution diameter of approximately [ ]$^{PROP/ECI}$ and an outer solution diameter of [ ]$^{PROP/ECI}$. The thickness of the tank is approximately [ ]$^{PROP/ECI}$. The tank is modeled in a vault with the top of the tank approximately [ ]$^{PROP/ECI}$ from the floor of the RPF. The vault shield plug considered in the shielding analysis is discussed in Subsection 4b.2.2.2 of the FSAR.

RLWS; liquid waste in annular waste tank; tank vault
The isotopic activity inventory for the annular waste tank is listed in Sheet 7 of Table 11.1-9 of the FSAR. The isotopic activity inventory accounts for greater than 99 percent of the expected activity. The RLWS annular waste tanks are described in Subsection 9b.7.4.2 of the FSAR. The inventory of nuclides in the annular waste tank is based on the nominal operating conditions as described in Table 11.1-1 of the FSAR, accounting for additional decay period and production material extraction. The annular waste tank is modeled with the same dimensions as the target solution hold tank. The tank is modeled in a vault with the top of the tank approximately [ ]$^{PROP/ECI}$ from the floor of the RPF. The vault shield plug considered in the shielding analysis is discussed in Subsection 4b.2.2.2 of the FSAR.

RLWS; liquid waste in RLWS collection tank; tank vault
The isotopic activity inventory for the collection waste tank is listed in Sheet 8 of Table 11.1-9 of the FSAR. The isotopic activity inventory accounts for greater than 99 percent of the expected activity. The RLWS collection tanks are described in Subsection 9b.7.4.2 of the FSAR. The inventory of nuclides in the collection tank is based on the nominal operating conditions as described in Table 11.1-1 of the FSAR, accounting for additional decay period and production material extraction. Extraction column washes are considered in the shielding analysis to be bounding of the liquid waste that would

normally be in the tank. The RLWS collection tank is modeled as approximately [ ]PROP/ECI. The top of the tanks are approximately [ ]PROP/ECI from the floor of the RPF. The vault shield plug considered in the shielding analysis is discussed in Subsection 4b.2.2.2 of the FSAR.

Table 11.1-10, Solid Radioactive Sources

NDAS; neutron driver; IU cell
The isotopic activity inventory for the activated neutron driver equipment is not provided. The neutron driver is not a significant source of radiation for the normally occupied area because of low activity and the neutron driver is shielded by multiple feet of concrete inside the IU cell. IU cell shielding is described in Subsection 4a2.5.2.2 of the FSAR.

TOGS; TOGS components; IU cell and TOGS cell
The isotopic activity inventory for the TOGS solid sources is listed in Sheet 2 of Table 11.1-10 of the FSAR. The isotopic activity inventory accounts for greater than 99 percent of the expected activity. The inventory of solid nuclides in the IU cell and the TOGS cell is based on the nominal operating conditions as described in Table 11.1-1 of the FSAR, accounting for continuous gas removal from the target solution during operation. The distribution of daughter products is assumed to be 30 percent in the TOGS cell and 70 percent in the IU cell. The shielding properties of the tanks and piping in the TOGS cell are not considered in the shielding analysis. The geometry and configuration of the TOGS cell shielding credited in the shielding analysis is described in Subsection 4a2.5.2.2 of the FSAR.

SCAS; neutron multiplier, SASS; IU cell
The isotopic activity for the neutron multiplier and the SASS are not provided. The neutron multiplier and SASS are not a significant source of radiation for the normally occupied area because they are located inside the light water pool in the IU cell. IU cell shielding is described in Subsection 4a2.5.2.2 of the FSAR.

MEPS, spent columns, supercell
The isotopic activity inventory for the spent columns is listed in Sheet 3 of Table 11.1-10 of the FSAR. The inventory of nuclides in the molybdenum product is based on the nominal operating conditions as described in Table 11.1-1 of the FSAR, accounting for additional decay period and production material extraction. The shielding properties of the MEPS columns are not credited in the shielding analysis. The supercell shielding is discussed in Subsection 4b.2.2.2 of the FSAR.

MEPS; glassware; supercell and solid waste drum storage
The isotopic activity for the MEPS glassware is not provided. The MEPS glassware does not significantly contribute to the dose rates in normally occupied areas in comparison to other source terms inside of the supercell and solid waste drum storage.

TSPS and URSS; uranium metal and uranium oxide; target solution preparation and storage areas
The isotopic activity for the fresh uranium metal and uranium oxides is not provided. The uranium metal and uranium oxides emit alpha radiation and do not significantly contribute to the direct dose rate in normally occupied areas.

RLWI; solidified waste drum; liquid waste solidification cell
The isotopic activity inventory for the solidified waste drum is listed in Sheet 4 of Table 11.1-10 of the FSAR. The isotopic activity inventory accounts for greater than 99 percent of the expected activity. The inventory of nuclides in the solidified waste drum is based on the nominal operating conditions as described in Table 11.1-1 of the FSAR, accounting for additional decay period and production material extraction. Two grouted barrels of 0.85 m length and 0.57 m diameter are modeled in the shielding analysis. The RLWI shielded enclosure (i.e., liquid waste solidification cell) is discussed in Subsection 4b.2.2.2 of the FSAR.

Solid radwaste; spent filters; supercell
The isotopic activity inventory for the spent supercell filters is listed in Sheet 5 of Table 11.1-10 of the FSAR. The isotopic activity inventory accounts for greater than 99 percent of the expected activity. The inventory of nuclides in the supercell filters is based on the nominal operating conditions as described in Table 11.1-1 of the FSAR, accounting for additional decay period and production material extraction. The shielding properties of the filters are not credited in the shielding analysis. Lead shielding with a thickness of 3 inches is placed around the filters.

SCAS; subcritical multiplication source; IU cell
The isotopic activity inventory of the subcritical multiplication source is not provided. The subcritical multiplication source is not a significant contributor to the dose rates in normally occupied areas. The subcritical multiplication source is below multiple feet of water in the light water pool inside the IU cell. IU cell shielding is described in Subsection 4a2.5.2.2 of the FSAR.

b. A summary of the assumptions, methodology, and input parameters for the exterior dose rate estimates from gaseous radioactive sources, identified in Table 11.1-5 of the FSAR, is provided in the SHINE Response to RAI 11-2a. A summary of the assumptions, methodology, and input parameters, for the DAC estimates from gaseous radioactive sources, identified in Table 11.1-6 of the FSAR, is described below for each of the facility locations listed in Table 11.1-6 of the FSAR.

   Radioactive sources that could become airborne at the SHINE facility are primarily tritium and radioactive gas produced as a byproduct of the production process. The systems handling gaseous radioactive materials include the TPS, the NDAS, the TOGS, the radiologically controlled area ventilation zone 1 (RVZ1), the PVVS, and the VTS. DAC calculations consider leakage from systems containing radioactive gases in the facility. A conservative best estimate of airborne release, due to normal operation and maintenance, is performed to estimate DACs for the facility.

   The estimated DACs are provided in Table 11.1-6 of the FSAR as a percentage of the DAC limits specified in 10 CFR Part 20, Appendix B. The overall DAC is calculated using the summation over all nuclides (i.e., summation of the concentration of nuclide i [µCi/mL] over the DAC limit for nuclide i [µCi/mL]). The following assumptions and input parameters are applied to the DAC estimate calculations:
   - IF maintenance activities that breach the PSB are not regular maintenance activities and are preceded by flushing and decontamination efforts.
   - In the RPF, solution transfer is done using vacuum lifts rather than pumps, and valves are hermetically sealed and designed for maintenance without breaching the process

boundary. As a result, leakage from regular maintenance activities is expected to be minimal compared to the normal system leakage.

- Unoccupied areas connected by the RVZ1r are assumed to be above 1 DAC during operation for conservatism.
- The dilution volume used to calculate the concentration for a source term in the IF is 1.26E+07 cubic meters per year based on the following:
  - Square footage of the IF general area: 8,700 square feet
  - Minimum outdoor airflow rate: 7.5 cfm per person
  - Estimated maximum occupant load: 13 persons per 1,000 square feet
- The dilution volume used to calculate the concentration for a source term in the RPF is 1.82E+07 cubic meters per year based on the following:
  - Square footage of the RPF general area: 12,540 square feet
  - Minimum outdoor airflow rate: 7.5 cfm per person
  - Estimated maximum occupant load: 13 persons per 1,000 square feet

Primary System Boundary, IF General Area

The PSB consists of the TSV, TOGS, TSV dump tank, and interconnecting piping. Most of the leakage from this system is the krypton, xenon, and iodine that are being circulated through TOGS. The mass of each isotope time averaged over an irradiation cycle is provided in Table 11-2-1. The fractional leak rates (fraction/second) for krypton, xenon, and iodine from the PSB to the primary confinement boundary are 8.35E-11, 6.68E-11, and 4.80E-11, respectively. Leak path factors (fraction/second) to the IF from the primary confinement boundary for noble gases and iodine are 3.81E-07 and 3.79E-07, respectively.

The mass of each nuclide in the primary confinement boundary is tracked with an iterative timestep calculation accounting for the amounts leaked into and out of the primary confinement boundary and accounting for decay at each timestep. The timesteps continue until no isotope shows greater than a 1 percent change in mass. The leakage out of the primary confinement boundary for the final timestep is used as the steady state leakage out of the primary confinement boundary. This is converted to annual leakage and applied to all 8 operating units to get the total leakage per year into the general area. The annual leakage values are then converted from grams per year to curies per year and divided by the dilution volume to get the concentration.

Tritium Systems, TPS Room

Tritium within the TPS glove box permeates to the TPS room at a yearly release rate of 0.5 Ci/yr. The dilution volume used to calculate the concentration for the TPS room is 1.81E+06 cubic meters per year based on a 1,248 square feet TPS room, 7.5 cfm per person minimum outdoor airflow rate, and 13 persons per 1,000 square feet maximum occupant load. The concentration is the quotient of the yearly release rate due to TPS glovebox permeation and the dilution volume in the TPS room.

Tritium Systems, IF General Area, Normal Operation

No leak path factor is used for tritium from the IU cell to the IF. It is assumed that any material released into the cell will be released to the IF general area through leakage or when the cell is opened for maintenance. NDAS component permeation in the IF general area is assumed at a yearly release rate of 8 Ci/yr of tritium. The concentration is the quotient of the yearly release rate due to permeation and the dilution volume in the IF.

Tritium Systems, IF General Area, Maintenance
No leak path factor is used for tritium from the IU cell to the IF. It is assumed that any material released into the cell will be released to the IF general area through leakage or when the cell is opened for maintenance. The directed airflow and flexible exhaust ducts in the NDAS service cell (NSC) are assumed to reduce tritium released through NDAS maintenance operations by 90 percent. The NDAS maintenance releases are assumed to occur during the one-week per year scheduled driver maintenance period for each driver, for a total of an 8-week duration for this release. NDAS maintenance releases to the IF general area occur at a yearly release rate of 20 Ci/yr of tritium. The concentration is the quotient of the yearly release rate due to maintenance and the eight-week dilution volume in the IF.

Below-Grade Vaults, RPF General Area
Out-leakage from the PVVS carbon delay beds is negligible due to most of the material being adsorbed onto the carbon and unavailable for release. The PVVS leak rate is calculated based on an assumption of 5,000 components and fittings with a maximum leak rate of 1E-4 standard cubic centimeters per second helium leakage equivalent at one atmosphere pressure differential (accounting for the partial pressure and molecular mass of the elements leaking out). TOGS purges are not taken into account for the DAC contribution from PVVS; the PSB leakage accounts for all TOGS gas that would be purged to PVVS.

The PVVS airborne isotopic activity inventory rate leaking to the IF is provided in Table 11-2-2 on a per batch basis. The same isotopic inventory is used for the available leakage from the PVVS equipment in the hot cell or in the below-grade vaults. The fractional leak rates out of the PVVS equipment for iodine, krypton, and xenon are 6.70E-05, 1.17E-04, and 9.33E-05, respectively. Leak path factors are used to calculate fractional leak rates from the below-grade vaults; the maximum fractional leak rates for noble gases and iodine are 7.60E-8 and 1.78E-8, respectively. The concentration inside the system is calculated using a PVVS flowrate of 12.8 scfm.

The isotopic inventory for the PVVS represents the cumulative material in PVVS from a single batch-cycle and is calculated for various timesteps after transfer to the RPF. The timestep calculation accounts for decay, release rate to the PVVS equipment, the flowrate of PVVS, and the leak rate into the below-grade vaults to provide the total leakage per batch-cycle. The leakage per batch-cycle into the RPF is multiplied by 8 batches per cycle and 50 cycles per year to calculate the annual release. The concentration is the quotient of the annual release and the RPF dilution factor.

PVVS Hot Cell, PVVS Hot Cell and RPF General Area
The isotopic activity inventory released per batch for the PVVS hot cell is provided in Sheet 3 of Table 11.1-5 of the FSAR. The PVVS hot cell leakage estimate uses the same method as the below-grade vaults description, with the addition of the PVVS guard beds, which have an efficiency of 99.5 percent for the capture of iodine, and rather than using the leak rate from the below-grade vault, the leak rate of the hot cells is used. The volume of the PVVS hot cell is modelled as approximately 350 cubic feet. The leak path factor for the hot cell confinements is 1.25E-4. The leak rate into the PVVS hot cell is 1E-06 L/s. The concentration is the quotient of the release rate per year and the dilution volume.

Extraction Hot Cell, Extraction Hot Cell and RPF General Area
Liquid releases are only considered for the processes where connections are regularly broken to install new separation columns. Three batches are assumed to be processed

through the extraction cells for calculation of the DAC within the hot cells. Eight batches are used to calculate the release to the RPF general area.

The extraction cell airborne isotopic activity inventory per batch is provided in Sheet 3 of Table 11.1-5 of the FSAR. An airborne release fraction for particulates of 2E-4 is used to determine the fraction of material that would be airborne. The volume of the extraction hot cell is modelled as approximately 180 cubic feet. The leak path factor for the hot cell confinements is 1.25E-4. The yearly release rate to the RPF is based on the release rate per batch times 8 batches per cycle and 50 cycles per year. The yearly release rate into one extraction cell is based on the release per batch times 3 batches per extraction cell and 50 cycles per year. The concentration is the quotient of the release rate per year and the dilution volume.

Purification Hot Cell, Purification Hot Cell and RPF General Area
The purification cell airborne isotopic activity inventory per batch is provided in Sheet 3 of Table 11.1-5 of the FSAR. Three batches are assumed to be processed through the purification cells for calculation of the DAC within the hot cells. Eight batches are used to calculate the release to the RPF general area.

An airborne release fraction for particulates of 2E-4 is used to determine the fraction of material that would be airborne. The volume of the purification hot cell is modelled as approximately 125 cubic feet. The leak path factor for the hot cell confinements is 1.25E-4. The yearly release rate to the RPF is based on the release rate per batch times 8 batches per cycle and 50 cycles per year. The yearly release rate into one purification cell is based on the release per batch times 3 batches per purification cell and 50 cycles per year. The concentration is the quotient of the release rate per year and the dilution volume.

Table 11-2-1 – Isotope Mass, Primary System Boundary During Irradiation Cycle

Nuclide	Mass (g)
I-123	[ ]PROP/ECI
I-124	[ ]PROP/ECI
I-125	[ ]PROP/ECI
I-126	[ ]PROP/ECI
I-129	[ ]PROP/ECI
I-130	[ ]PROP/ECI
I-131	[ ]PROP/ECI
I-132	[ ]PROP/ECI
I-132m	[ ]PROP/ECI
I-133	[ ]PROP/ECI
I-134	[ ]PROP/ECI
I-135	[ ]PROP/ECI
Kr-81	[ ]PROP/ECI
Kr-83m	[ ]PROP/ECI
Kr-85	[ ]PROP/ECI
Kr-85m	[ ]PROP/ECI
Kr-87	[ ]PROP/ECI
Kr-88	[ ]PROP/ECI
Xe-127	[ ]PROP/ECI
Xe-131m	[ ]PROP/ECI
Xe-133	[ ]PROP/ECI
Xe-133m	[ ]PROP/ECI
Xe-135	[ ]PROP/ECI
Xe-135m	[ ]PROP/ECI
Xe-138	[ ]PROP/ECI

Table 11-2-2 – Isotope Mass Flow Rate Leaked from PVVS Below-Grade per Batch

Nuclide	Mass Flow Rate (g/hr)
I-123	[ ]PROP/ECI
I-124	[ ]PROP/ECI
I-125	[ ]PROP/ECI
I-126	[ ]PROP/ECI
I-129	[ ]PROP/ECI
I-130	[ ]PROP/ECI
I-131	[ ]PROP/ECI
I-132	[ ]PROP/ECI
I-132m	[ ]PROP/ECI
I-133	[ ]PROP/ECI
I-134	[ ]PROP/ECI
I-135	[ ]PROP/ECI
Kr-81	[ ]PROP/ECI
Kr-83m	[ ]PROP/ECI
Kr-85	[ ]PROP/ECI
Kr-85m	[ ]PROP/ECI
Kr-87	[ ]PROP/ECI
Kr-88	[ ]PROP/ECI
Xe-127	[ ]PROP/ECI
Xe-131m	[ ]PROP/ECI
Xe-133	[ ]PROP/ECI
Xe-133m	[ ]PROP/ECI
Xe-135	[ ]PROP/ECI
Xe-135m	[ ]PROP/ECI
Xe-138	[ ]PROP/ECI

c. Table 11.1-8 of the FSAR provides the estimated annual release from normal and maintenance operations of the SHINE facility. SHINE limited the nuclides in Table 11.1-8 of the FSAR to only nuclides with greater than 1 Ci/yr estimated release because the nuclides identified in Table 11.1-8 of the FSAR account for greater than 99.9 percent of the total annual estimated release activity. The SHINE Response to RAI 11-3a provides the estimated annual release from normal and maintenance operations for all nuclides.

RAI 11-3

FSAR Table 3.1-3 “SHINE Design Criteria,” Criterion 35 – “Control of releases of radioactive materials to the environment” specifies that the facility is designed to include means to suitably control the release of radioactive materials in gaseous and liquid effluents and to handle radioactive solid wastes produced during normal operation, including anticipated transients. Sufficient holdup capacity is provided for retention of radioactive gases.

The NRC staff noted that SHINE has performed a dose analysis to demonstrate that the off-site doses to individuals through direct exposure and potential environmental pathways, such as leafy vegetable ingestion, meat ingestion, and milk ingestion from the release of airborne radionuclides will not exceed the limits of 10 CFR Part 20, “Standards for Protection Against Radiation.” The results of this analysis are in Section 11.1.1.1, “Airborne Radioactive Sources,” of the SHINE FSAR, page 11.1-5 and state that the “estimated annual doses at the MEI and the nearest resident are 3.9 mrem and 0.3 mrem, respectively, which are less than the limit in 10 CFR Part 20.” The current version of the SHINE FSAR lacks enough information for the NRC staff to determine whether SHINE annual doses to the public comply with 10 CFR Part 20.

a. Provide a summary of the assumptions, methodology, and input parameters used to estimate annual doses to the MEI and nearest resident. Specifically, include FSAR tables with the input parameters used to perform the dose calculations that determined the above referenced doses to the MEI. This includes the base radionuclides (listed by isotope) used for analysis, along with those assumptions made in regard to radionuclide filtration, consumption rates, food yields, and any other inputs necessary for the staff to perform an independent evaluation.

The limits in 10 CFR Part 20 are specified in terms of TEDE and mrem because the regulations in 10 CFR Part 20 are based on the International Commission on Radiation Protection's (ICRP) recommendations in ICRP Reports 26 and 30. Dose conversion factors acceptable to the NRC staff are derived from data and methodologies provide in ICRP Publication 30, "Limits for Intakes of Radionuclides by Workers" and can be found in Federal Guidance Report (FGR) No. 11, "Limiting Values of Radionuclide Intake and Air Concentration and Dose Conversion Factors for Inhalation, Submersion, and Ingestion," and FGR No. 12, "External Exposure to Radionuclides in Air, Water, and Soil," for exposure to radionuclides in air, water, and soil.

As stated above the estimated annual doses at the maximum exposed individual and the nearest resident are 3.9 mrem and 0.3 mrem which are less than the limits in 10 CFR Part 20. However, the FSAR does not specify if the estimated doses are in terms of total effective dose equivalent (TEDE) which is defined as the sum of the effective dose equivalent (for external exposures) and the committed effective dose equivalent. It is not clear in the FSAR whether the dose coefficients used in the analysis are consistent with the regulations in 10 CFR Part 20 and ICRP Reports 26 and 30.

b. Clarify if the dose estimates were performed using the appropriate dose conversion factors based on ICRP 26/30 methods. If not, re-compute the radiological dose estimates to the MEI and nearest resident in terms of TEDE to be consistent with the regulations in 10 CFR Part 20.

Consistent with the evaluation findings in Section 11.1.1 of NUREG-1537, Part 2, the information requested in parts a. and b. of this RAI is necessary for the NRC staff to confirm that the FSAR identifies potential radiation safety hazards associated with the SHINE facility and conduct an independent review of the SHINE radiation protection program.

SHINE Response

a. As described in Subsection 11.1.1.1 of the FSAR, annual off-site doses due to the normal operation of the SHINE facility have been calculated using the computer code GENII2.

   The material at risk (MAR) for the dose analysis is based on normal gaseous waste emissions and estimated leakage from the various systems and processes in the SHINE facility.

   Table 11.1-8 of the FSAR provides the estimated annual release from normal and maintenance operations of the SHINE facility, summarizing the estimated release used in the dose analysis, as described in the SHINE Response to RAI 11-2c. Table 11-3-1 provides the complete listing of material released considered in the dose analysis, including the iodine particulate and gaseous values. Table 11-3-1 includes any reduction in radionuclide activity that occurs due to the normal operation of carbon adsorbers, as appropriate.

   The long-term, ground level atmospheric dispersion factors (χ/Q) and deposition factors (D/Q) for the maximally exposed individual (MEI) and nearest resident are provided in Table 11-3-2. The MEI is assumed to be on the site boundary, in the direction that has the maximum dispersion and deposition factors. The long-term χ/Q and D/Q values were input directly into GENII2. The nearest resident is located in the northwest direction at a distance of 0.49 miles.

   The hierarchy established for selecting input parameters into GENII1 was (1) site-specific data; (2) guidance from Regulatory Guide 1.109, “Calculation of Annual Doses to Man from Routine Releases of Reactor Effluents for the Purpose of Evaluating Compliance with 10 CFR Part 50, Appendix I” (Reference 8); and (3) GENII2 default parameters. As the GENII2 default parameters establish the foundation for the GENII2 code, Table 11-3-3 summarizes the input parameters obtained from site-specific data and Regulatory Guide 1.109 guidance. Remaining input parameters were maintained at their GENII2 default value.

   In accordance with Section B of Regulatory Guide 1.109, the exposed population is considered to be made up of four age groups: infants (0 to 1 year), children (1 to 11 years), teenagers (11 to 17 years), and adults (17 and older). The receptor intake input parameters are either applicable across the age groups or are age specific. Table 11-3-4 summarizes the receptor intake input parameters and is organized by applicability across age groups. As the GENII2 default parameters establish the foundation for the GENII2 code, Table11-3-4 summarizes the input parameters obtained from Regulatory Guide 1.109 guidance. Remaining input parameters were retained at the GENII2 default value. As the effluent

pathways do not include liquid releases, input parameters necessary to calculate dose to liquid pathways have been retained at the GENII2 default values.

The dose consequences are calculated based on the International Commission on Radiological Protection (ICRP) 30/48 option in GENII2. The ICRP 30/48 option implements dosimetry models recommended in ICRP 26, 30, and 48.

The resulting estimated doses at the MEI and the nearest resident are 4.6 mrem total effective dose equivalent (TEDE) and 0.3 mrem TEDE, respectively.

SHINE has revised Subsection 11.1.1.1 of the FSAR to provide reference to Table 11.1-8 of the FSAR as the estimated release of airborne nuclides considered in the dose analysis to estimate annual doses to the MEI and the nearest resident. SHINE has also revised Table 11.1-7 and Table 11.1-8 of the FSAR to correct administrative errors in the magnitude of activity considered in the normal yearly release calculation. A mark-up of the FSAR incorporating these changes is provided as Attachment 1.

Table 11-3-1 – Material Released

Gaseous (Ci/yr.)	Particulate (Ci/yr.)	Nuclide	Activity (Ci/yr.)	Nuclide	Activity (Ci/yr.)
		Ar-41	2.30E-02	Nd-147	9.79E-06
		As-77	1.24E-06	Np-239	1.42E-06
		Ba-140	1.00E-05	Pr-143	7.18E-07
		Br-82	1.36E-05	Rh-103m	1.97E-05
		Br-83	1.73E-02	Rh-105	1.48E-05
		Br-84	7.66E-06	Ru-103	1.51E-06
		Ce-141	7.17E-07	Sb-127	5.05E-05
		Ce-143	8.24E-07	Sb-128	6.03E-06
1.05E-19	4.50E-20	I-123	1.50E-19	Sb-129	6.14E-05
1.84E-24	7.88E-25	I-124	2.63E-24	Tc-99m	1.38E-02
1.70E-17	7.30E-18	I-125	2.43E-17	Te-127	5.88E-05
9.81E-14	4.21E-14	I-126	1.40E-13	Te-129	9.52E-05
1.07E-12	4.60E-13	I-129	1.53E-12	Te-129m	7.41E-06
2.56E-07	1.10E-07	I-130	3.66E-07	Te-131	5.92E-05
3.96E-03	1.70E-03	I-131	5.66E-03	Te-131m	2.26E-04
4.51E-03	1.93E-03	I-132	6.44E-03	Te-132	2.07E-03
1.29E-07	5.52E-08	I-132m	1.84E-07	Xe-127	9.88E-11
1.03E-02	4.41E-03	I-133	1.47E-02	Xe-131m	1.29E+03
2.39E-05	1.03E-05	I-134	3.42E-05	Xe-133	7.77E+03
4.51E-03	1.93E-03	I-135	6.44E-03	Xe-133m	1.05E+00
		Kr-81	2.11E-11	Xe-135	6.04E+00
		Kr-83m	5.90E+00	Xe-135m	9.96E+00
		Kr-85	1.23E+02	Xe-138	9.65E-02
		Kr-85m	5.03E+01	Y-91	2.62E-06
		Kr-87	1.16E-01	Y-93	2.95E-06
		Kr-88	2.12E+00	Zr-97	8.27E-06
		La-140	1.84E-05	HTO[1]	1.30E+02
		La-141	5.70E-06	HT[1]	0.00E+00
		Mo-99	1.43E-01	OBT[1]	0.00E+00
		Nb-97	4.17E-06		

1: HTO = tritiated water, HT = tritiated gas, OBT = organically-bound tritium

Table 11-3-2 – Dispersion and Deposition Factors

	Atmospheric Dispersion Factor, χ/Q (s/m^3)	Deposition Factor, D/Q ($1/m^2$)
MEI	7.1E-05	1.4E-07
Nearest Resident	5.3E-06	8.9E-09

Table 11-3-3 – Exposure Pathways Input Parameters

Parameter	Value	Parameter	Value
Exposure Pathways	Animal product ingestion Terrestrial food crop ingestion	Yield of milk animal forage	0.7 kg/m^2
Duration of Exposure Period	1.0 yr	Growing period for leafy vegetables	60 d
End of Release Period	30 yrs	Growing period for root vegetables	60 d
Time from start to exposure	15 yrs	Growing period for fruits	60 d
Absolute humidity	12.16 g/m^3	Growing period for grains	60 d
Average rain rate, when raining	34.79 in/yr	Yield of leafy vegetables	2 kg/m^2
Surface soil areal density	240.0 kg/m^2	Yield of root vegetables	2 kg/m^2
Surface soil layer thickness used for density	15 cm	Yield of fruits	2 kg/m^2
Surface soil density	1600 kg/m^3	Yield of grains	2 kg/m^2
Dry deposition interception fraction to plants	1	Time from harvest to ingestion of leafy vegetables	1 d
Weathering rate constant from plants	14 d	Time from harvest to ingestion of root vegetables	60 d
Milk animal feed consumption rate (hay) - for goats	1.5 kg/d	Time from harvest to ingestion of fruits	60 d
Milk animal forage consumption rate (grass) - for goats	4.5 kg/d	Time from harvest to ingestion of grains	60 d
Storage time for meat animal feed	90 d	Time from harvest to ingestion of meat	20 d
Storage time for meat animal forage	0 d	Ingestion	All available pathways selected
Storage time for milk animal forage	0 d	Inhalation	All available pathways selected
Growing period for milk animal feed	30 d	External	All available pathways selected
Yield of milk animal feed	0.7 kg/m^2		

Table 11-3-4 – Receptor Intake Input Parameters

Input Parameter	Value			
Input Parameter Applicable to All Age Groups				
Indoor shielding factor	0.7			
Input Parameter	**Value by Age**			
	Infant (0-1)	**Child (1-11)**	**Teen (11-17)**	**Adult (17-99)**
Food Crop Ingestion (kg/yr)				
Leaf vegetable consumption rate	0	26	42	64
Root vegetable consumption rate	0	280.8	340.2	280.8
Fruit consumption rate	0	114.4	138.6	114.4
Grain consumption rate	0	124.8	151.2	124.8
Animal Product Ingestion (kg/yr)				
Meat consumption rate	0	41	65	110
Poultry consumption rate	0	GENII2 default	GENII2 default	GENII2 default
Milk consumption rate	0.931 kg/d	0.931 kg/d	1.13 kg/d	0.874 kg/d
Egg consumption rate	0	GENII2 default	GENII2 default	GENII2 default
Air Inhalation (m^3/yr)				
Air inhalation rate	1400	3700	8000	8000
Indoor Inhalation (m^3/yr)				
Indoor inhalation rate	1400	3700	8000	8000

b. SHINE has re-computed the radiological dose to the MEI and nearest resident using the appropriate dose conversion factors based on ICRP 26/30 methods. SHINE has revised Subsection 11.1.1.1 of the FSAR to provide the resulting estimated annual doses at the MEI and nearest resident. A mark-up of the FSAR incorporating these changes is provided as Attachment 1.

RAI 11-4

NUREG-1537 Part 2, Section 11.1, states that acceptance criteria for applicant's analysis for doses in unrestricted areas should "contain conservative best estimates of the predicted annual total doses to at least the following in the unrestricted areas: (1) the maximum exposed individual, (2) the nearest permanent residence, and (3) any location of special interest, such as a classroom or campus dormitory."

Section 11.1.7.2.1, "Direct Radiation Monitoring," of the SHINE FSAR states that "dosimeters are stationed off site at special interest areas." However, no analysis for special interest areas appears in the SHINE FSAR which is inconsistent with the acceptance criteria in NUREG-1537.

Provide clarification as to whether the "special interest areas" referred to in Section 11.1.7.2.1 of the SHINE FSAR are meant to be "locations of special interest" as discussed in the acceptance criteria in NUREG 1537 Part 2. If so, provide the associated public dose analysis for these locations of special interest or a justification for why a separate dose analysis is not necessary.

SHINE Response

The "special interest areas" referred to in Subsection 11.1.7.2.1 of the FSAR are synonymous with the "locations of special interest" discussed in the acceptance criteria of Section 11.1.1 of NUREG-1537, Part 2 (Reference 7). These special interest areas are identified in Table 11.1-14 of the FSAR.

Table 11-4-1 provides the direction sector and distance to each special interest area, the maximally exposed individual, and the nearest full-time resident, as well as the corresponding χ/Q values for each.

The annual TEDE for a receptor is proportional to the χ/Q values between the receptor and the source. As shown in Table 11-4-1, the χ/Q values for the special interest areas are approximately two orders of magnitude lower than the χ/Q value for the MEI and approximately an order of magnitude lower than the χ/Q value for the nearest full-time resident. Therefore, the annual TEDE at each special interest area is bounded by the TEDE to the MEI or the nearest full-time resident provided in Subsection 11.1.1.1 of the FSAR. As such, a separate dose analysis for the special interest areas is not necessary.

Table 11-4-1 – Atmospheric Dispersion Values for Special Interest Areas, Maximally Exposed Individual, and Nearest Full-Time Resident

Location	Direction Sector	Distance	χ/Q (sec/m^3)
Rock County Christian Elementary School	SSW	1.30 miles	5.4E-7
Jackson Elementary School	NNE	1.45 mile	9.6E-7
University of Wisconsin – Rock County	NW	2 miles[1]	5.7E-7
Maximally Exposed Individual	E	0.14 miles	7.1E-5
Nearest Full-Time Resident	NW	0.49 miles	5.3E-6

1. The χ/Q value for University of Wisconsin – Rock County was not directly calculated. The χ/Q value assigned is equal to the χ/Q value calculated at 2 miles NW from the SHINE facility. Since University of Wisconsin – Rock County is more than 2 miles from the facility, this χ/Q value is conservative.

RAI 11-5

10 CFR 20.2003, “Disposal by release into sanitary sewerage,” and 10 CFR 20.2007, “Compliance with environmental and health protection regulations” as it relates to the applicant’s description of their potential liquid releases pathways and ensuring that liquid releases are done in accordance with NRC as well as other federal, state, and local regulations.

Section 11.1.7.2 of the SHINE FSAR states that “There are no routine radioactive liquid effluent discharges from the RCA” and that “There are no piped liquid effluent pathways from the RCA to the sanitary sewer.” Based on this information the staff understands that the only release of liquid radioactive material from the facility will be via the sewer.

If there are no piped liquid effluent pathways from the RCA to the sanitary sewer, clarify in the FSAR how SHINE intends to perform sewer discharges in accordance with 10 CFR 20.2003, “Disposal by release into sanitary sewerage,” and 10 CFR 20.2007, “Compliance with environmental and health protection regulations,” and how often the sewer discharges are expected to be performed. This additional information is necessary to ensure NRC staff and the public understand how SHINE intends to operate the facility while protecting the public and environment. In addition, section 11.1.4, “Radiation Monitoring and Surveying, of the SHINE FSAR states that “liquid effluent releases are collected and sampled prior to release.” In the context of this section, clarify if liquid effluent releases meant to be understood as liquid releases via sewer as discussed in 11.1.7.2.

SHINE Response

There are no liquid discharge connections from the RCA to the sanitary sewer. Liquids generated or located in the RCA that require disposal and do not exceed the criteria for release to the sanitary sewer may be discharged via collection in portable containers. Prior to discharge, the collected liquid is sampled, analyzed, and verified to meet the criteria for release to the sanitary sewer provided in 10 CFR 20.2003, 10 CFR 20.2007, and Janesville City Ordinance 40-170. Liquids meeting these criteria are transferred outside of the RCA in portable containers and released to the sanitary sewer via connections outside of the RCA.

If analysis determines that the collected liquids do not meet the criteria for release to the sanitary sewer, the contents will be disposed of as low-level radioactive waste. Liquid discharge volumes are estimated to be less than 40 gallons weekly. The SHINE Response to RAI WR-2 (Reference 9) provides additional details on the discharge of collected liquids to the sanitary sewer, including the identification of potential sources of these non-routine liquid discharges to the sanitary sewer.

The statement “liquid effluent releases are collected and sampled prior to release” in Subsection 11.1.4 of the FSAR is meant to be understood as liquid releases via the sanitary sewer described in Subsection 11.1.7.2 of the FSAR.

SHINE has revised Subsections 11.1.4.1 and 11.2.3 of the FSAR to clarify the manner and frequency in which these discharges to the sanitary sewer are expected to occur. A mark-up of the FSAR incorporating these changes is provided as Attachment 1.

RAI 11-6

NUREG-1537, Part 2, Chapter 11 of the standard review plan and acceptance criteria, as augmented by the interim staff guidance (ISG) for licensing radioisotope production facilities is applicable to reviewing a description of the radiation protection program and waste management for the licensing of a radioisotope production facility. The acceptance criteria in Chapter 11 of the ISG states that the application should identify a controlled area as defined in 10 CFR 20.1003. 10 CFR 20.1003 defines "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."

Section 11.1.5.1 of the SHINE FSAR defines "controlled access area" in the first sentence but later states that "Facility visitors include delivery people, tour guests, and service personnel who are transient occupants of the controlled area." Provide a definition for "controlled area" and provide clarification as to the difference between "controlled access area" and "controlled area" when used in Section 11.1.5 of the SHINE FSAR.

SHINE Response

Consistent with 10 CFR 20.1003, SHINE defines the "controlled area" as the area, outside of a restricted area but inside the site boundary, access to which can be limited by SHINE for any reason. At the SHINE facility, this is the area within the site boundary, and is referred to as the "owner controlled area," as described in Subsection 2.1.1.2 of the FSAR.

Consistent with 10 CFR 73.2, SHINE defines "controlled access area" as any temporarily or permanently established area which is clearly demarcated, access to which is controlled and which affords isolation of the material or persons within it. Controlled access areas at the SHINE facility are described in the Physical Security Plan.

SHINE has revised Subsection 11.1.5.1 of the FSAR to explicitly define "controlled area," and to remove the discussion of a controlled access area. A mark-up of the FSAR incorporating these changes is provided as Attachment 1.

RAI 11-7

FSAR Table 3.1-3 "SHINE Design Criteria," Criterion 38, "Monitoring Radioactivity Releases," specifies that means are provided for monitoring the primary confinement boundary, hot cell, and glovebox atmospheres to detect potential leakage of gaseous or other airborne radioactive material. Potential effluent discharge paths and the plant environs are monitored for radioactivity that may be released from normal operations, including anticipated transients, and from postulated accidents. In addition, NUREG-1537 Part 2, Section 11.1.1, "Environmental Monitoring," has acceptance criteria that states the environmental monitoring program should provide confidence that a significant radiological impact on the environment from the facility would be detected and the type and magnitude of the radiological impact would be determined.

Section 11.1.7.2, "Effluent Release Pathways," of the SHINE FSAR presents the pathways that represent plausible public exposure scenarios from airborne effluents, one of which is the "Ingestion exposure pathway." However, Section 11.1.7.2.3, "Ingestion Pathway (Biota Monitoring)," states that "biota monitoring is not routinely included in the REMP." However, given that Section 11.1.7.1 states that Regulatory Guide (RG) 4.1, "Radiological Environmental Monitoring for Nuclear Power Plants," was considered when developing the radiological

environmental monitoring program for the SHINE facility the staff requests the following information:

a. Provide a summary of the evaluation and data that was used to justify that routine biota monitoring is not necessary for SHINE and how this conclusion meets the objectives of RG 4.1, as applicable, and explain this in the SHINE FSAR.

b. If monitoring of this pathway will not be performed prior to start up or during operations, describe if evaluations will be performed in the future to determine if biota monitoring may become necessary after start-up.

SHINE Response

a. While the regulatory positions in Regulatory Guide 4.1, "Radiological Environmental Monitoring for Nuclear Power Plants" (Reference 10) are not applicable to non-power production and utilization facilities, SHINE considered these regulatory positions in selecting the monitoring locations of the radiological environmental monitoring program (REMP) described in Subsection 11.1.7 of the FSAR. SHINE determined that biota monitoring is not necessary as part of the REMP as nuclear power plants have long monitored the ingestion pathway and have seen neither appreciable dose nor upward trending of deposition, and because particulate and iodine radionuclides are not normally expected to be present in measurable quantities within airborne effluent releases from the SHINE facility.

   In evaluating the monitoring of the ingestion pathway at nuclear power plants, SHINE reviewed the historical effluent data from nuclear power plants documented in the annual volumes of NUREG/CR-2907, "Radioactive Effluents from Nuclear Power Plants" (Reference 11, et al.). The evaluation showed a steady decline in the activities of noble gases discharged in gaseous effluents, a long-standing, historical measure of the licensee's ability to control gaseous effluents. Additionally, SHINE evaluated the resulting median maximum annual organ doses from gaseous releases from nuclear power plants and determined these doses did not represent an appreciable dose contribution to the public. As stated in Subsection 11.1.7.2.3, the SHINE source term is expected to be several orders of magnitude lower than that of a nuclear power plant.

   Although biota monitoring is not included in the REMP, SHINE performs environmental airborne sampling to identify and quantify particulates and radioiodine in airborne effluents, as described in Subsection 11.1.7.2.2 of the FSAR.

   The objectives of the REMP described in Section B of Regulatory Guide 4.1 do not explicitly identify exposure pathways to be monitored. Rather, the objectives generally focus on the establishment of pre-operational baseline data, appropriate characterization of radiation or radioactive materials in the local environment, reporting monitoring results, and maintenance of the REMP. The environmental monitoring program for the SHINE facility, described in Subsection 11.1.7 of the FSAR, meets the objectives of the REMP described in Section B of Regulatory Guide 4.1 without necessitating biota monitoring.

   SHINE has revised Subsection 11.1.7.2.3 of the FSAR to provide additional justification that biota monitoring is not needed in the REMP. A mark-up of the FSAR incorporating these changes is provided as Attachment 1.

b. As described in Subsection 11.1.7.8 of the FSAR, annual reviews of the REMP are conducted to examine the adequacy and effectiveness of the REMP. This programmatic review evaluates the need to expand or reduce the monitoring program based on the results of collected data, including whether biota monitoring may be necessary after start-up.

RAI 11-8

FSAR Table 3.1-3 “SHINE Design Criteria,” Criterion 38, “Monitoring Radioactivity Releases,” specifies that means are provided for monitoring the primary confinement boundary, hot cell, and glovebox atmospheres to detect potential leakage of gaseous or other airborne radioactive material. Potential effluent discharge paths and the plant environs are monitored for radioactivity that may be released from normal operations, including anticipated transients, and from postulated accidents. In addition, NUREG-1537 Part 2, Section 11.1.1, “Environmental Monitoring,” has acceptance criteria that states the environmental monitoring program should provide confidence that a significant radiological impact on the environment from the facility would be detected and the type and magnitude of the radiological impact would be determined.

However, Section 11.1.7.2.4, “Groundwater Monitoring,” states that surface waters are not expected to accumulate detectable levels of radioactivity as such surface water will not be included in the radiological environmental monitoring program.

Provide a summary of the evaluation and assumptions that support the justification that surface water sampling is not needed in the radiological environmental monitoring program and provide this discussion in the SHINE FSAR. If monitoring of surface water will not be performed prior to start up or during operations, clarify if evaluations will be performed in the future to determine if surface water sampling may become necessary after start-up.

SHINE Response

There are no surface water features on the SHINE site, nor are there any surface waters immediately adjacent to the SHINE site. As described in Subsection 2.4.1.1 of the FSAR, the nearest surface water to the SHINE site is an unnamed tributary, approximately one mile southeast of the SHINE site, and the Rock River, located approximately two miles west of the SHINE site. Since SHINE does not utilize surface water in any production process, and SHINE does not discharge to surface water, surface water is not considered to be a route of exposure for ingestion or direct radiation for the SHINE facility. Therefore, SHINE determined that surface water sampling is not needed to meet the objectives of the REMP.

Groundwater sampling for the presence of radionuclide contaminants, as part of the REMP, is performed as described in Subsection 11.1.7.2.4 of the FSAR.

As described in Subsection 11.1.7.8 of the FSAR, annual reviews of the REMP are conducted to examine the adequacy and effectiveness of the REMP. This programmatic review evaluates the need to expand or reduce the monitoring program based on the results of collected data, including whether surface water sampling may be necessary after start-up.

SHINE has revised Subsection 11.1.7.2.4 of the FSAR to provide additional justification that surface water sampling is not needed in the REMP. A mark-up of the FSAR incorporating these changes is provided as Attachment 1.

RAI 11-9

10 CFR 50.34(b)(3) states that the FSAR will include the kinds and quantities of radioactive material expected to be produced in the operation of the facility as it relates to tracking the amount of radioactive material contained in the various facility components as well as the anticipated radionuclides inside and outside the plant environment. Section 11.2 of the FSAR states that "the material storage building is used for interim storage of wastes for decay and for preparation for shipment. Wastes are not stored for more than five years." Section 11.2.2.2 states, "At least 5,600 square feet (ft2) of the material staging building is for temporary storage to allow for decay." Section 11.2.2.2 also states that "drums may be stored in multiple layers." However, the SHINE FSAR does not provide anticipated activities of waste stored within the material storage building.

Provide additional detail regarding the anticipated activities of waste stored within the material storage building so that the staff can determine compliance with 10 CFR 50.34(b)(3) in defining the annual amount of radionuclides generated as waste at the SHINE facility. In addition, this source term information will be used by the staff to perform an independent evaluation for direct dose to determine compliance with 10 CFR 20.1301 dose limits. Specifically, provide information on the activity by isotope, the volumes of the barrels that the applicant plans to use for stored waste, shielding thickness credited in the analysis and any other assumptions the applicant used to determine that "The material staging building design evaluated the shielding provided by the building to ensure 10 CFR 20 site dose limits are met and ALARA principles are followed," as described in FSAR section 11.2.2.

SHINE Response

The kinds and quantities of radioactive materials expected to be processed through the material staging building (MATB) are the estimated annual wastes identified in Table 11.2-1 of the FSAR, excluding laboratory waste. The SHINE Response to RAI WM-2 (Reference 9) provides a process flow diagram for waste treatment and disposal pathways, including waste quantities expected to be processed through the MATB.

The MATB is designed to store solidified liquid waste in four below-grade shielded concrete pits. Other radioactive waste, activated materials, or contaminated materials may be stored in the MATB; however, radiation levels in the MATB are expected to be bounded by radiation from the waste drums stored in the below-grade pits containing solidified waste.

Drums in the below-grade pits are stored in racks which allow for vertical stacking of up to three drums. Removable concrete panels shield the top opening of the below-grade pits and provide access to the storage racks and drums. Drums have the approximate dimensions of a standard 55-gallon drum. The MATB drum storage area has a capacity of approximately 1500 drums. Simplifying assumptions are made about the waste drums and storage configuration in evaluating MATB shielding, which assure a conservative analysis with respect to radiation levels, such as:

- The MATB is modeled with materials and approximate dimensions of:
  - Exterior walls made of concrete 1 foot thick and 30 feet high (roof geometry has negligible contribution to the dose analysis).
  - The drum storage area is modelled as one concrete below grade pit, 15 feet wide, 12 feet deep, with 18-inch thick concrete foundation walls.
  - The shield panels that cover the pit are constructed of 1-foot thick concrete, 15 feet long, 4-1/2 feet wide, with half inch gaps between panels.

- The waste drums are modeled with waste decayed 40 weeks after target irradiation ends. Considering main production facility processing times and the potential for waste decay periods of up to five years within the MATB, 40 weeks is a conservative isotope inventory to apply to the dose analysis.
- The analysis modeled 252 drums, in the center of a below-grade pit, positioned nearest to the east exterior wall, configured in a six by fourteen array, and stacked three high. This number of drums produces conservative dose estimates considering the 40 week decay period and considering that modeling additional drums would have less dose effect due to additional distance.
- The model does not account for shielding provided by the drum construction material.

The isotope inventory modeled for the drums is a nominal source term derived from best estimate partitioning information and accounting for removal of [ ]$^{PROP/ECI}$ of all strontium, cesium, and cerium based on SHINE liquid waste extraction processes. The isotope inventory provided in Table 11-9-1 at 40 weeks accounts for greater than 99 percent of the activity in the isotope inventory modeled in the SHINE dose analysis.

Table 11-9-1 – Solidified Waste Drum Source Terms for Material Staging Building Radiation Analysis

Isotope	Activity at 40 weeks (Ci)	Isotope	Activity at 40 weeks (Ci)
Ag-110m	[ ]PROP/ECI	Pr-144	[ ]PROP/ECI
Ba-137m	[ ]PROP/ECI	Pr-144m	[ ]PROP/ECI
Ba-140	[ ]PROP/ECI	Pu-238	[ ]PROP/ECI
Bi-212	[ ]PROP/ECI	Pu-239	[ ]PROP/ECI
Cd-113m	[ ]PROP/ECI	Pu-240	[ ]PROP/ECI
Cd-115m	[ ]PROP/ECI	Ra-224	[ ]PROP/ECI
Ce-141	[ ]PROP/ECI	Rh-103m	[ ]PROP/ECI
Ce-144	[ ]PROP/ECI	Rh-106	[ ]PROP/ECI
Cs-134	[ ]PROP/ECI	Ru-103	[ ]PROP/ECI
Cs-137	[ ]PROP/ECI	Ru-106	[ ]PROP/ECI
Eu-152	[ ]PROP/ECI	S-35	[ ]PROP/ECI
Eu-154	[ ]PROP/ECI	Sb-125	[ ]PROP/ECI
Eu-155	[ ]PROP/ECI	Sm-151	[ ]PROP/ECI
Eu-156	[ ]PROP/ECI	Sn-119m	[ ]PROP/ECI
H-3	[ ]PROP/ECI	Sn-121	[ ]PROP/ECI
Kr-85	[ ]PROP/ECI	Sn-121m	[ ]PROP/ECI
La-140	[ ]PROP/ECI	Sn-123	[ ]PROP/ECI
Nb-95	[ ]PROP/ECI	Sn-126	[ ]PROP/ECI
Nb-95m	[ ]PROP/ECI	Sr-89	[ ]PROP/ECI
Nd-147	[ ]PROP/ECI	Sr-90	[ ]PROP/ECI
P-32	[ ]PROP/ECI	Tc-99	[ ]PROP/ECI
P-33	[ ]PROP/ECI	Te-125m	[ ]PROP/ECI
Pa-234m	[ ]PROP/ECI	Te-127	[ ]PROP/ECI
Pb-212	[ ]PROP/ECI	Te-127m	[ ]PROP/ECI
Pm-147	[ ]PROP/ECI	Te-129	[ ]PROP/ECI
Pm-148	[ ]PROP/ECI	Te-129m	[ ]PROP/ECI
Pm-148m	[ ]PROP/ECI	Th-228	[ ]PROP/ECI
Pr-143	[ ]PROP/ECI		

RAI 11-10

Subpart H, “Respiratory Protection and Controls to Restrict Internal Exposure in Restricted Areas,” of 10 CFR Part 20, “Standards for Protection Against Radiation,” specifies the conditions under which respiratory protection equipment may be used and lists the procedural requirements that must be met by a licensee when using respirators to limit intakes of radioactive material or to take credit for the protection assigned to a respirator in limiting and estimating intakes of airborne radioactive materials.

Section 11.3 of the SHINE FSAR briefly describes the respiratory protection program and states that the program is “in accordance with 10 CFR 20, Subpart H.” Regulatory Guide 8.15, “Acceptable Programs for Respiratory Protection” describes a respiratory protection program that is acceptable to the NRC staff. Regulatory Guide 8.15 also provides guidance on performing evaluations to determine whether the use of respirators optimizes the sum of internal and external dose and other risks. The NRC staff uses RG 8.15 to evaluate respiratory protection programs for compliance with Subpart H of 10 CFR Part 20, except in those cases where an applicant proposes an acceptable alternative method for complying with specific portions of the regulations.

Clarify if SHINE intends to conform their program to RG 8.15 or if SHINE intends to apply their own methods when using respiratory protection. If SHINE’s program will not conform with RG 8.15, further describe SHINE’s methods in the FSAR. For example, describe:

- the respiratory protection training program,
- industry standards used as basis for program,
- TEDE-ALARA evaluations,
- fit-testing program,
- types of respirators to be used,
- use of assigned protection factors,
- air sampling bioassays,
- qualifications of respirator wearers,
- safety precautions,
- breathing air quality management,
- maintenance, repair, testing, QA and storage of respiratory equipment, and
- whether respirators will be used for mixed hazards (airborne radioactive material and nonradioactive hazardous material).

SHINE Response

The SHINE respiratory protection program conforms to the guidance of Regulatory Guide 8.15, “Acceptable Programs for Respiratory Protection” (Reference 12).

RAI 11-11

Section 5.8.1, “Operating Reports,” of the SHINE Technical Specifications requires SHINE to submit an annual report providing “A summary of exposures received by facility personnel and visitors where such exposures are greater than 25 percent of that allowed or recommended.” However, 10 CFR 20.2206, “reports of individual monitoring,” requires licensees who fall under one of the categories listed in 10 CFR 20.2206(a) to submit an annual report of the results of

individual monitoring carried out by the licensee for each individual for whom monitoring was required by 10 CFR 20.1502.

Justify why SHINE does not meet the category described in 10 CFR 20.2206(a)(7) and why the current technical specification annual report requirement is appropriate for SHINE.

SHINE Response

SHINE meets the license category described in 10 CFR 20.2206(a)(7). SHINE has revised Subsection 12.5.1 of the FSAR and Section 5.8.1 of the technical specifications to clarify the annual reporting requirements related to individual monitoring. A mark-up of the FSAR incorporating these changes is provided as Attachment 1. A mark-up of the technical specifications incorporating these changes is provided as Attachment 2.

RAI 11-12

Section 11.1.2, “Radiation Protection Program” describes the SHINE radiation protection program which is intended to protect the radiological health and safety of workers and members of the public. Section 5.1.3, “Facility Staffing Required” of the SHINE technical specifications provides the required minimum staffing for when the facility is not secured. The staffing plan does not specifically include radiation safety personnel but states that radiation safety personnel should be included on a telephone number list readily available in the control room.

Considering the types, quantities and concentrations of radioactive material that will be handled at the SHINE facility, describe the staffing expectations for radiation safety personnel when the facility is secured and not secured. In addition, confirm if radiation safety personnel work will still be handled by the radiation safety personnel, as defined in Chapter 11, during a secured condition.

This information is necessary for the NRC staff to verity that SHINE has developed, documented, and implemented a radiation protection program commensurate with the scope and extent of licensed activities and sufficient to ensure compliance with the provisions of 10 CFR Part 20.

SHINE Response

In addition to the minimum operations staffing requirements described in Section 5.1.3 of the technical specifications, the SHINE Emergency Plan requires an individual to be present on-site to fulfill the role of Radiation Safety Coordinator whether the facility is secured or not secured. The Radiation Safety Coordinator is responsible for making on-site and off-site dose assessments and projections, recommending protective actions, and identifying exposed personnel and determining their radiation dose. This individual is capable of monitoring radiation dose rates and contamination levels. The requirement for the Radiation Safety Coordinator may be met by the presence of a member of the radiation protection staff or another individual who has received training and is qualified to perform the required duties.

When the SHINE facility is secured, implementation of the radiation protection program functions will be handled by radiation protection staff, as described in Subsection 11.1.2 of the FSAR.

References

1. NRC letter to SHINE Medical Technologies, LLC, “Issuance of Request for Additional Information Related to the SHINE Medical Technologies, LLC Operating License Application (EPID No. L-2019-NEW-0004),” dated January 27, 2021

2. SHINE Medical Technologies, LLC letter to the NRC, “SHINE Medical Technologies, LLC Application for an Operating License,” dated July 17, 2019 (ML19211C143)

3. American Society of Mechanical Engineers, “Rules for Construction of Overhead and Gantry Cranes (Top Running Bridge, Multiple Girder),” NOG-1-2004, 2004

4. U.S. Nuclear Regulatory Commission, “Control of Heavy Loads at Nuclear Power Plants,” NUREG-0612, July 1980

5. American National Standards Institute, “Radioactive Materials - Special Lifting Devices for Shipping Containers Weighing 10,000 Pounds (4500 kg) or More,” ANSI N14.6-1993, 1993

6. American Society of Mechanical Engineers, “Slings,” B30.9-2018, 2018

7. U.S. Nuclear Regulatory Commission, "Guidelines for Preparing and Reviewing Applications for the Licensing of Non-Power Reactors: Standard Review Plan and Acceptance Criteria," NUREG-1537, Part 2, February 1996 (ML042430048)

8. U.S. Nuclear Regulatory Commission, “Calculation of Annual Doses to Man from Routine Releases of Reactor Effluents for the Purpose of Evaluating Compliance with 10 CFR Part 50, Appendix I,” Regulatory Guide 1.109, Revision 1, October 1977

9. SHINE Medical Technologies, LLC letter to NRC, dated March 13, 2020, “SHINE Medical Technologies, LLC Application for an Operating License Response to Environmental Requests for Additional Information” (ML20073E880)

10. U.S. Nuclear Regulatory Commission, “Radiological Environmental Monitoring for Nuclear Power Plants,” Regulatory Guide 4.1, Revision 2, June 2009

11. U.S. Nuclear Regulatory Commission, “Radioactive Effluents from Nuclear Power Plants, Annual Report 2018,” NUREG/CR-2907, Volume 24, November 2020

12. U.S. Nuclear Regulatory Commission, “Acceptable Programs for Respiratory Protection,” Regulatory Guide 8.15, Revision 1, October 1999 (ML003739528)

ENCLOSURE 2
ATTACHMENT 1

SHINE MEDICAL TECHNOLOGIES, LLC

SHINE MEDICAL TECHNOLOGIES, LLC OPERATING LICENSE APPLICATION
RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION AND SUPPLEMENT NO. 7

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION
PUBLIC VERSION

FINAL SAFETY ANALYSIS REPORT CHANGES
(MARK-UP)

33 pages follow

Manufacturer's Association of America (CMAA) 70, Specifications for Top Running Bridge & Gantry Type Multiple Girder Electric Overhead Traveling Cranes (CMAA, 2004); and ASME NOG-1, Rules for Construction of Overhead and Gantry Cranes (Top Running Bridge, Multiple Girder) (ASMSE, 20154).

The IF overhead crane is designed to the following criteria:

- Meet seismic requirements and prevent failures of the crane that could damage safety-related equipment such that the equipment would be prevented from performing its safety function.
- Meet the single-failure-proof design criteria and construction of ASME NOG-1, Type I cranes and be designed to perform as a Service Level B – Light Service crane as described in CMAA 70.
- Secure its load in place upon a loss of power and any fault condition. The hoisting machinery and wire rope reeving system, in addition to other affected components, is designed to withstand the most severe potential overload, including two-blocking and load hang-up.

Radioisotope Production Facility Overhead Crane

The radioisotope production facility (RPF) overhead crane is designed to meet the applicable requirements of ASME B30.2 (ASME, 2011a), CMAA 70 (CMAA, 2004), and ASME NOG-1 (ASME, 20154).

The RPF overhead crane is designed to the following criteria:

- Meet seismic requirements and prevent failures of the crane that could damage safety-related equipment such that the equipment would be prevented from preforming its safety function.
- Meet the design criteria and construction of ASME NOG-1, Type II cranes and be designed to perform as a Service Level B – Light Service crane as described in CMAA 70.
- Remain in place with or without a load during a seismic event.

9b.7.2.2 System Description

Irradiation Facility Overhead Crane

The IF overhead crane is a 40-ton, double girder, bridge style crane designed for the handling of shield cover plugs and equipment such as neutron drivers and process skids inside the IF. The IF overhead crane is designed to span the width of the IF and travel the length of the IF.

The use of a single-failure-proof crane with rigging and procedures that implement the requirements of NUREG-0612, Control of Heavy Loads at Nuclear Power Plants (USNRC, 1980) assures that the potential for a heavy load drop is extremely small, and therefore, analysis of the potential effects of heavy load drops are not required.

The IF overhead crane is designed and constructed such that it will remain in place and support the critical load during and after an aircraft impact but is not required to be operational after this

9b.8 REFERENCES

ANSI, 1993. Radioactive Materials – Special Lifting Devices for Shipping Containers Weighing 10,000 Pounds (4500 kg) or More, ANSI N14.6-1993, American National Standards Institute, 1993.

ANSI/ANS, 1998. Guide for Nuclear Criticality Safety in the Storage of Fissile Materials, ANSI/ANS 8.7-1998 (R2007), American National Standards Institute/American Nuclear Society, 1998.

ASME, 2004. Rules for Construction of Overhead and Gantry Cranes (Top Running Bridge, Multiple Girder), NOG-1-2004, American Society of Mechanical Engineers, 2004.

ASME, 2009. Code on Nuclear Air and Gas Treatment, AG-1-2009, American Society of Mechanical Engineers, 2009.

ASME, 2010. Rules for Construction of Pressure Vessels, Boiler and Pressure Vessel Code, Section VIII, Division 1, American Society of Mechanical Engineers, 2010.

ASME, 2011a. Overhead and Gantry Cranes (Top Running Bridge, Single or Multiple Girder, Top Running Trolley Hoist), B30.2-2011, American Society of Mechanical Engineers, 2011.

ASME, 2011b. Building Services Piping, B31.9-2011, American Society of Mechanical Engineers, 2011.

ASME, 2013. Process Piping, B31.3-2012, American Society of Mechanical Engineers, 2013.

~~**ASME, 2015.** Rules for Construction of Overhead and Gantry Cranes (Top Running Bridge, Multiple Girder), NOG-1-2015, American Society of Mechanical Engineers, 2015.~~

ASME, 2018. Slings, B30.9-2018, American Society of Mechanical Engineers, 2018.

CMAA, 2004. Specifications for Top Running Bridge & Gantry Type Multiple Girder Electric Overhead Traveling Cranes, CMAA 70-2004, Crane Manufactures Association of America, Inc., 2004.

USNRC, 1980. Control of Heavy Loads at Nuclear Power Plants, NUREG-0612, U.S. Nuclear Regulatory Commission, July 1980.

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

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

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

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

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

11.1.1.1 Airborne Radioactive Sources

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

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

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

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

The dose analysis considered the estimated release of airborne radionuclides provided in Table 11.1-8 and exposure to off-site individuals through direct exposure and potential environmental pathways, such as leafy vegetable ingestion, meat ingestion, and milk ingestion. The analysis considered variations in consumption and other parameters by age group. The estimated annual doses at the MEI and the nearest resident are ~~3.9~~4.6 mrem total effective dose equivalent (TEDE) and 0.3 mrem TEDE, respectively, which are less than the limit in 10 CFR 20.

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

11.1.1.2 Liquid Radioactive Sources

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

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

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

11.1.1.3 Solid Radioactive Sources

Solid radioactive sources exist in several locations in the SHINE facility. Fresh, low enriched uranium is received at the facility in the form of uranium metal or uranium oxide that has been enriched to a nominal 19.75 percent by weight in uranium-235 (U-235). If uranium metal is received, it is converted to uranium oxide and then to a liquid uranyl sulfate solution. Other solid

radioactive sources are listed in Table 11.2-1 and include spent extraction columns from the molybdenum extraction process, glassware, spent filters, and solidified liquid waste.

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

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

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

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

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

11.1.1.4 Technical Specifications

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

11.1.2 RADIATION PROTECTION PROGRAM

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

11.1.2.1 Commitment to Radiation Protection Program Implementation

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

The radiation protection program meets the requirements of 10 CFR 20, Subpart B, Radiation Protection Programs, and is consistent with the guidance provided in Regulatory Guide 8.2, Revision 1, Administrative Practices in Radiation Surveys and Monitoring (USNRC, 2011), and ANSI/ANS 15.11-2016, Radiation Protection at Research Reactor Facilities (ANSI/ANS, 2016).

c. Continuous Tritium Detectors

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

d. Gaseous Effluent Monitoring

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

- Continuous monitoring of radioactive stack releases for noble gases.
- Generating real time data for control room display and recording.
- Allowing periodic collection of filters to allow for laboratory analysis for particulate and iodine.

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

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

There are no piped radioactive liquid effluent discharges from the facility; therefore, there are no installed liquid effluent monitors. However, infrequent liquid effluent releases are collected ~~and~~, sampled, and verified to meet the criteria for release provided in 10 CFR 20.2003 and 10 CFR 20.2007 prior to ~~release~~discharge to the sanitary sewer.

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

f. Radiation Area Monitors

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

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

g. Control Point Monitoring

Monitor stations are located at the access points for restricted areas. Monitors are provided to detect radioactive contamination of personnel. Monitoring station locations

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

- Regulatory Guide 8.2, Guide for Administrative Practice in Radiation Monitoring (USNRC, 2011)
- Regulatory Guide 8.7, Instructions for Recording and Reporting Occupational Radiation Exposure Data (USNRC, 2018)
- Regulatory Guide 8.9, Acceptable Concepts, Models, Equations, and Assumptions for a Bioassay Program (USNRC, 1993)
- Regulatory Guide 8.24, Health Physics Surveys During Enriched Uranium-235 Processing and Fuel Fabrication (USNRC, 2012) (applicable to target solution preparation processes)
- Regulatory Guide 8.34, Monitoring Criteria and Methods to Calculate Occupational Radiation Doses (USNRC, 1992)
- ANSI N323AB-2013, American National Standard for Radiation Protection Instrumentation Test and Calibration, Portable Survey Instruments (ANSI/ANS, 2014)

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

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

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

11.1.4.3 Technical Specifications

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

11.1.5 RADIATION EXPOSURE CONTROL AND DOSIMETRY

11.1.5.1 Controlled ~~Access~~ Area

~~The area of the SHINE site within the security fence, including within the main production facility physical structure beyond the main reception area, but outside any restricted area is part of the controlled access area. Due to the presence of administrative and physical barriers, members of the public do not have direct access to this controlled access area of the site and must be processed by security and authorized to enter the facility. Training for access to a controlled access area is provided commensurate with the radiological hazard.~~Consistent with 10 CFR 20.1003, the controlled area is the area outside of a restricted area but inside the site boundary, access to which can be limited by SHINE for any reason. At the SHINE facility, this is the area within the site boundary, and is referred to as the “owner controlled area,” as described in Subsection 2.1.1.2.

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

11.1.5.1.1 Radiological Zones

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

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

a. Unrestricted Area

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

b. Restricted Area

   10 CFR 20.1003 defines a restricted area as an area where access is limited by the licensee for the purpose of protecting individuals against undue risks from exposure to radiation and radioactive materials. Access to and egress from a restricted area at the facility site is through a radiation protection control point. Monitoring equipment is located at these control points.

   Most restricted areas are located within the physical structure of the main production facility and locations in the material staging building where radioactive material is normally stored. Radioactive material may be temporarily stored in outdoor areas during transfer between areas. These temporary areas may require that a restricted area be established with the controls described in this section.

c. Radiologically Controlled Area

   The RCA is a restricted area. The RCA is an area within the restricted area posted for the purpose of protecting individuals against undue risks from exposure to radiation and radioactive materials. Only individuals who have successfully completed training in

activity. Table 3.12-1 of NUREG-1301 (USNRC, 1991) suggests an additional air sample location in the vicinity of a community having the highest calculated annual average ground level deposition factor, D/Q. This CAS requirement is combined with the air sample location at the site boundary location in the north direction (refer to Table 11.1-14). A description of air sample locations and the rationale for air sample locations are provided in Table 11.1-14.

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

11.1.7.2.3 Ingestion Pathway (Biota Monitoring)

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

11.1.7.2.4 Groundwater Monitoring

There are no surface water features on the SHINE site, nor are there any surface waters immediately adjacent to the SHINE site. Surface waters of the rivers in the vicinity of the ~~plant~~SHINE site (e.g., the Rock River and its tributaries) are not expected to accumulate detectable levels of radioactivity. SHINE does not utilize surface water in any production process, and SHINE does not discharge to surface water; therefore, surface water is not considered to be a route of exposure for ingestion or direct radiation for the SHINE facility. As such, surface water sampling is not included in the REMP. Similarly, marine life in the rivers is not expected to accumulate detectable levels of radioactivity and thus sampling of fish or other marine creatures for the ingestion pathway is not included in the REMP.

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

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

Table 11.1-5 – Airborne Radioactive Sources
(Sheet 1 of 4)

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

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

Table 11.1-5 – Airborne Radioactive Sources
(Sheet 2 of 4)

TOGS, Off-gas Piping, Zeolite Beds, TOGS Shielded Cell (Conservative Best Estimate Activity)

Isotope	Activity (Ci)
I-124	[ ]PROP/ECI
I-125	[ ]PROP/ECI
I-126	[ ]PROP/ECI
I-129	[ ]PROP/ECI
I-130	[ ]PROP/ECI
I-131	[ ]PROP/ECI
I-133	[ ]PROP/ECI
Kr-81	[ ]PROP/ECI
Kr-83m	[ ]PROP/ECI
Kr-85	[ ]PROP/ECI
Kr-85m	[ ]PROP/ECI
Kr-87	[ ]PROP/ECI
Kr-88	[ ]PROP/ECI
Xe-127	[ ]PROP/ECI
Xe-131m	[ ]PROP/ECI
Xe-133	[ ]PROP/ECI
Xe-133m	[ ]PROP/ECI
Xe-135	[ ]PROP/ECI
Xe-135m	[ ]PROP/ECI
Xe-138	[ ]PROP/ECI

Table 11.1-5 – Airborne Radioactive Sources
(Sheet 3 of 4)

RVZ1, Supercell Atmosphere, Supercell Gloveboxes (Conservative Best Estimate Activity)

Extraction Cell	
Isotope	**Activity (Ci)**
Br-83	PROP/ECI
I-130	PROP/ECI
I-131	PROP/ECI
I-132	PROP/ECI
I-133	PROP/ECI
I-134	PROP/ECI
I-135	PROP/ECI
Mo-99	PROP/ECI
Tc-99m	PROP/ECI
Xe-133	PROP/ECI
Xe-133m	PROP/ECI
Xe-135	PROP/ECI
Xe-135m	PROP/ECI

Purification Cell	
Isotope	Activity(Ci)
I-130	PROP/ECI
I-131	PROP/ECI
I-132	PROP/ECI
I-133	PROP/ECI
I-134	PROP/ECI
I-135	PROP/ECI
Mo-99	PROP/ECI
Tc-99m	PROP/ECI
Xe-133	PROP/ECI
Xe-133m	PROP/ECI
Xe-135	PROP/ECI
Xe-135m	PROP/ECI

PVVS Cell	
Isotope	**Activity (Ci)**
I-123	PROP/ECI
I-124	PROP/ECI
I-125	PROP/ECI
I-126	PROP/ECI
I-129	PROP/ECI
I-130	PROP/ECI
I-131	PROP/ECI
I-132	PROP/ECI
I-132m	PROP/ECI
I-133	PROP/ECI
I-134	PROP/ECI
I-135	PROP/ECI
Kr-81	PROP/ECI
Kr-83m	PROP/ECI
Kr-85	PROP/ECI
Kr-85m	PROP/ECI
Kr-87	PROP/ECI
Kr-88	PROP/ECI
Xe-127	PROP/ECI
Xe-131m	PROP/ECI
Xe-133	PROP/ECI
Xe-133m	PROP/ECI
Xe-135	PROP/ECI
Xe-135m	PROP/ECI
Xe-138	PROP/ECI

Table 11.1-5 – Airborne Radioactive Sources
(Sheet 4 of 4)

PVVS and VTS, PVVS and VTS Piping, Pipe Trenches, Valve Pits, and PVVS Hot Cell
(Conservative Best Estimate Activity)

Isotope	Activity (Ci)
I-123	[ ] PROP/ECI
I-124	[ ] PROP/ECI
I-125	[ ] PROP/ECI
I-126	[ ] PROP/ECI
I-129	[ ] PROP/ECI
I-130	[ ] PROP/ECI
I-131	[ ] PROP/ECI
I-132	[ ] PROP/ECI
1-132m	[ ] PROP/ECI
I-133	[ ] PROP/ECI
I-133m	[ ] PROP/ECI
I-134	[ ] PROP/ECI
I-135	[ ] PROP/ECI
Kr-81	[ ] PROP/ECI
Kr-83m	[ ] PROP/ECI
Kr-85	[ ] PROP/ECI
Kr-85m	[ ] PROP/ECI
Kr-87	[ ] PROP/ECI
Kr-88	[ ] PROP/ECI
Xe-122	[ ] PROP/ECI
Xe-123	[ ] PROP/ECI
Xe-127	[ ] PROP/ECI
Xe-131m	[ ] PROP/ECI
Xe-133	[ ] PROP/ECI
Xe-133m	[ ] PROP/ECI
Xe-135	[ ] PROP/ECI
Xe-135m	[ ] PROP/ECI
Xe-138	[ ] PROP/ECI

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

Parameter	PVVS Pathway	Hot Cells	Primary Confinement Boundary	General Area
Primary Nuclide Inventory Constituents	Kr, Xe, I	Kr, Xe, I, particulates	Kr, Xe, Ar 41, N-16	Kr, Xe, I, H-3
Type of Radiation Emitted	Beta and Gamma	Beta and Gamma	Beta and Gamma	Beta and Gamma
Total Curies	9.~~0~~7E+05	320	430	13~~2~~0
Primary Constituents Released	Kr, Xe	Kr, Xe	Kr, Xe	Kr, Xe, H-3
Type of Radiation Emitted	Beta and Gamma	Beta and Gamma	Beta and Gamma	Beta and Gamma
Total Curies	9.~~3~~2E+03	16	~~9.5~~8.2	13~~2~~0
Delay Time Credited for Decay	1.7 days (Kr) 40 days (Xe)	None	1 minute	None
Iodine Removal Mechanisms	Carbon Guard Bed Carbon Delay Beds	Carbon Filter on Hot Cell RVZ1 Exhaust Carbon Filter on Facility RVZ1 Exhaust	Carbon Filter on Facility RVZ1 Exhaust	Carbon Filter on Facility RVZ1 Exhaust

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

Radionuclide	Annual Release (Ci)
Kr-83m	5.9E+00
Kr-85	1.2E+02
Kr-85m	5.0E+01
Kr-88	2.~~2~~1E+00
Xe-131m	1.3E+03
Xe-133	7.8E+03
Xe-133m	1.1E+00
Xe-135	6.~~2~~0E+00
Xe-135m	1.0E+01
H-3	~~7~~1.3E+0~~1~~2

Table 11.1-9 – Liquid Radioactive Sources
(Sheet 1 of 8)

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

Table 11.1-9 – Liquid Radioactive Sources
(Sheet 2 of 8)

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

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

Table 11.1-9 – Liquid Radioactive Sources
(Sheet 3 of 8)

SCAS, Target Solution in TSV and TSV Dump Tank (Shutdown), IU Cell, and
SCAS, Target Solution in TSV (Operating), IU Cell (Conservative Best Estimate Activity)

Isotope	Activity (Ci)	Isotope	Activity (Ci)	Isotope	Activity (Ci)
Ag-109m	PROP/ECI	Nd-149	PROP/ECI	Sn-128	PROP/ECI
Ba-137m	PROP/ECI	Np-239	PROP/ECI	Sr-89	PROP/ECI
Ba-139	PROP/ECI	P-32	PROP/ECI	Sr-90	PROP/ECI
Ba-140	PROP/ECI	Pd-109	PROP/ECI	Sr-91	PROP/ECI
Ba-141	PROP/ECI	Pm-147	PROP/ECI	Sr-92	PROP/ECI
Br-83	PROP/ECI	Pm-149	PROP/ECI	Tc-101	PROP/ECI
Br-84	PROP/ECI	Pm-151	PROP/ECI	Tc-104	PROP/ECI
Ce-141	PROP/ECI	Pr-143	PROP/ECI	Tc-99m	PROP/ECI
Ce-143	PROP/ECI	Pr-144	PROP/ECI	Te-127	PROP/ECI
Ce-144	PROP/ECI	Pr-145	PROP/ECI	Te-129	PROP/ECI
Cs-137	PROP/ECI	Pr-146	PROP/ECI	Te-131	PROP/ECI
Cs-138	PROP/ECI	Rb-88	PROP/ECI	Te-131m	PROP/ECI
I-131	PROP/ECI	Rb-89	PROP/ECI	Te-132	PROP/ECI
I-132	PROP/ECI	Rh-103m	PROP/ECI	Te-133	PROP/ECI
I-133	PROP/ECI	Rh-105	PROP/ECI	Te-133m	PROP/ECI
I-133m	PROP/ECI	Rh-105m	PROP/ECI	Te-134	PROP/ECI
I-134	PROP/ECI	Rh-106	PROP/ECI	U-237	PROP/ECI
I-135	PROP/ECI	Rh-107	PROP/ECI	U-239	PROP/ECI
Kr-83m	PROP/ECI	Ru-103	PROP/ECI	Xe-133	PROP/ECI
Kr-85m	PROP/ECI	Ru-105	PROP/ECI	Xe-133m	PROP/ECI
Kr-87	PROP/ECI	Ru-106	PROP/ECI	Xe-135	PROP/ECI
Kr-88	PROP/ECI	Sb-127	PROP/ECI	Xe-135m	PROP/ECI
La-140	PROP/ECI	Sb-128m	PROP/ECI	Xe-138	PROP/ECI
La-141	PROP/ECI	Sb-129	PROP/ECI	Y-90	PROP/ECI
La-142	PROP/ECI	Sb-130	PROP/ECI	Y-91	PROP/ECI
La-143	PROP/ECI	Sb-131	PROP/ECI	Y-91m	PROP/ECI
Mo-101	PROP/ECI	Se-81	PROP/ECI	Y-92	PROP/ECI
Mo-99	PROP/ECI	Se-83	PROP/ECI	Y-93	PROP/ECI
Nb-97	PROP/ECI	Sm-153	PROP/ECI	Y-94	PROP/ECI
Nb-97m	PROP/ECI	Sm-155	PROP/ECI	Zr-95	PROP/ECI
Nb-98m	PROP/ECI	Sn-127	PROP/ECI	Zr-97	PROP/ECI
Nd-147	PROP/ECI				

Table 11.1-9 – Liquid Radioactive Sources (Sheet 4 of 8)

PCLS, Primary Cooling Water in Pump and Piping, IU Cell and Primary Cooling Room (Conservative Best Estimate Activity)

Isotope	Activity (Ci)
Ag-108	6.39E-04
Ag-110	1.25E-03
Ag-110m	6.14E-05
C-14	4.18E-05
Cl-38	1.01E-05
Co-60	4.64E-05
Co-60m	2.54E-04
Cr-51	7.22E-06
Cu-64	1.74E-04
Cu-66	3.54E-05
H-3	3.71E-03
N-16	4.98E-02
Na-24	7.98E-05
S-35	3.07E-05

Table 11.1-9 – Liquid Radioactive Sources (Sheet 5 of 8)

MEPS, Mo Eluate in Mo Eluate Hold Tank, Supercell (Conservative Best Estimate Activity)

Isotope	Activity (Ci)	Isotope	Activity (Ci)
[ ]PROP/ECI	[ ]PROP/ECI	[ ]PROP/ECI	[ ]PROP/ECI
[ ]PROP/ECI	[ ]PROP/ECI	[ ]PROP/ECI	[ ]PROP/ECI
[ ]PROP/ECI	[ ]PROP/ECI	[ ]PROP/ECI	[ ]PROP/ECI
[ ]PROP/ECI	[ ]PROP/ECI	[ ]PROP/ECI	[ ]PROP/ECI
[ ]PROP/ECI	[ ]PROP/ECI	[ ]PROP/ECI	[ ]PROP/ECI
[ ]PROP/ECI	[ ]PROP/ECI	[ ]PROP/ECI	[ ]PROP/ECI
[ ]PROP/ECI	[ ]PROP/ECI	[ ]PROP/ECI	[ ]PROP/ECI
[ ]PROP/ECI	[ ]PROP/ECI	[ ]PROP/ECI	[ ]PROP/ECI
[ ]PROP/ECI	[ ]PROP/ECI	[ ]PROP/ECI	[ ]PROP/ECI
[ ]PROP/ECI	[ ]PROP/ECI	[ ]PROP/ECI	[ ]PROP/ECI
[ ]PROP/ECI	[ ]PROP/ECI	[ ]PROP/ECI	[ ]PROP/ECI
[ ]PROP/ECI	[ ]PROP/ECI	[ ]PROP/ECI	[ ]PROP/ECI
[ ]PROP/ECI	[ ]PROP/ECI	[ ]PROP/ECI	[ ]PROP/ECI
[ ]PROP/ECI	[ ]PROP/ECI	[ ]PROP/ECI	[ ]PROP/ECI
[ ]PROP/ECI	[ ]PROP/ECI	[ ]PROP/ECI	[ ]PROP/ECI
[ ]PROP/ECI	[ ]PROP/ECI	[ ]PROP/ECI	[ ]PROP/ECI
[ ]PROP/ECI	[ ]PROP/ECI	[ ]PROP/ECI	[ ]PROP/ECI
[ ]PROP/ECI	[ ]PROP/ECI	[ ]PROP/ECI	[ ]PROP/ECI
[ ]PROP/ECI	[ ]PROP/ECI	[ ]PROP/ECI	[ ]PROP/ECI
[ ]PROP/ECI	[ ]PROP/ECI	[ ]PROP/ECI	[ ]PROP/ECI
[ ]PROP/ECI	[ ]PROP/ECI	[ ]PROP/ECI	[ ]PROP/ECI
[ ]PROP/ECI	[ ]PROP/ECI	[ ]PROP/ECI	[ ]PROP/ECI
[ ]PROP/ECI	[ ]PROP/ECI	[ ]PROP/ECI	[ ]PROP/ECI
[ ]PROP/ECI	[ ]PROP/ECI	[ ]PROP/ECI	[ ]PROP/ECI
[ ]PROP/ECI	[ ]PROP/ECI	[ ]PROP/ECI	[ ]PROP/ECI

Table 11.1-9 – Liquid Radioactive Sources (Sheet 6 of 8)

TSSS, Target Solution in Target Solution Hold Tank, Tank Vault
(Conservative Best Estimate Activity)

Isotope	Activity (Ci)	Isotope	Activity (Ci)
Ag-109m	PROP/ECI	Pr-143	PROP/ECI
Ag-111	PROP/ECI	Pr-144	PROP/ECI
Ag-112	PROP/ECI	Pr-144m	PROP/ECI
Ba-137m	PROP/ECI	Pr-145	PROP/ECI
Ba-139	PROP/ECI	Rb-88	PROP/ECI
Ba-140	PROP/ECI	Rh-103m	PROP/ECI
Br-83	PROP/ECI	Rh-105	PROP/ECI
Cd-115	PROP/ECI	Rh-105m	PROP/ECI
Ce-141	PROP/ECI	Rh-106	PROP/ECI
Ce-143	PROP/ECI	Ru-103	PROP/ECI
Ce-144	PROP/ECI	Ru-105	PROP/ECI
Cs-137	PROP/ECI	Ru-106	PROP/ECI
I-131	PROP/ECI	Sb-129	PROP/ECI
I-132	PROP/ECI	Sm-153	PROP/ECI
I-133	PROP/ECI	Sm-156	PROP/ECI
I-134	PROP/ECI	Sn-121	PROP/ECI
I-135	PROP/ECI	Sn-127	PROP/ECI
In-115m	PROP/ECI	Sr-89	PROP/ECI
La-140	PROP/ECI	Sr-90	PROP/ECI
La-141	PROP/ECI	Sr-91	PROP/ECI
La-142	PROP/ECI	Sr-92	PROP/ECI
Nb-97	PROP/ECI	Tc-99m	PROP/ECI
Nb-97m	PROP/ECI	Te-132	PROP/ECI
Nd-147	PROP/ECI	U-237	PROP/ECI
Nd-149	PROP/ECI	Y-90	PROP/ECI
Np-239	PROP/ECI	Y-91	PROP/ECI
P-32	PROP/ECI	Y-91m	PROP/ECI
Pd-109	PROP/ECI	Y-92	PROP/ECI
Pd-112	PROP/ECI	Y-93	PROP/ECI
Pm-147	PROP/ECI	Zr-95	PROP/ECI
Pm-149	PROP/ECI	Zr-97	PROP/ECI
Pm-151	PROP/ECI		

Table 11.1-9 – Liquid Radioactive Sources (Sheet 7 of 8)

RLWS, Liquid Waste in Annular Waste Tank, Tank Vault
(Conservative Best Estimate Activity)

Isotope	Activity (Ci)	Isotope	Activity (Ci)
PROP/ECI	PROP/ECI	PROP/ECI	PROP/ECI

Table 11.1-9 – Liquid Radioactive Sources
(Sheet 8 of 8)

RLWS, Liquid Waste in RLWS Collection Tank, Tank Vault
(Conservative Best Estimate Activity)

Isotope	Activity (Ci)	Isotope	Activity (Ci)
[ ]PROP/ECI	[ ]PROP/ECI	[ ]PROP/ECI	[ ]PROP/ECI
[ ]PROP/ECI	[ ]PROP/ECI	[ ]PROP/ECI	[ ]PROP/ECI
[ ]PROP/ECI	[ ]PROP/ECI	[ ]PROP/ECI	[ ]PROP/ECI
[ ]PROP/ECI	[ ]PROP/ECI	[ ]PROP/ECI	[ ]PROP/ECI
[ ]PROP/ECI	[ ]PROP/ECI	[ ]PROP/ECI	[ ]PROP/ECI
[ ]PROP/ECI	[ ]PROP/ECI	[ ]PROP/ECI	[ ]PROP/ECI
[ ]PROP/ECI	[ ]PROP/ECI	[ ]PROP/ECI	[ ]PROP/ECI
[ ]PROP/ECI	[ ]PROP/ECI	[ ]PROP/ECI	[4]PROP/ECI
[ ]PROP/ECI	[ ]PROP/ECI	[ ]PROP/ECI	[ ]PROP/ECI
[ ]PROP/ECI	[ ]PROP/ECI	[ ]PROP/ECI	[ ]PROP/ECI
[ ]PROP/ECI	[ ]PROP/ECI	[ ]PROP/ECI	[ ]PROP/ECI
[ ]PROP/ECI	[ ]PROP/ECI	[ ]PROP/ECI	[ ]PROP/ECI
[ ]PROP/ECI	[ ]PROP/ECI	[ ]PROP/ECI	[ ]PROP/ECI
[ ]PROP/ECI	[ ]PROP/ECI	[ ]PROP/ECI	[ ]PROP/ECI
[ ]PROP/ECI	[ ]PROP/ECI	[ ]PROP/ECI	[ ]PROP/ECI
[ ]PROP/ECI	[ ]PROP/ECI	[ ]PROP/ECI	[ ]PROP/ECI
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[ ]PROP/ECI	[ ]PROP/ECI	[ ]PROP/ECI	[ ]PROP/ECI
[ ]PROP/ECI	[ ]PROP/ECI	[ ]PROP/ECI	[ ]PROP/ECI
[ ]PROP/ECI	[ ]PROP/ECI		

Table 11.1-10 – Solid Radioactive Sources
(Sheet 1 of 5)

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

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

Table 11.1-10 – Solid Radioactive Sources
(Sheet 2 of 5)

TOGS, TOGS Components, IU Cell and TOGS Cell
(Conservative Best Estimate Activity)

Isotope	Activity (Ci)
Ba-137m	4.00E+01
Ba-139	3.61E+03
Ba-140	1.09E+03
Ba-141	1.09E+02
Ce-141	7.90E+01
Cs-137	4.22E+01
Cs-138	6.76E+03
La-140	1.01E+03
La-141	1.07E+02
La-142	2.70E+01
Rb-88	3.86E+03
Rb-89	4.48E+03
Sr-89	2.38E+03
Sr-90	2.37E+01
Sr-91	1.10E+03
Sr-92	1.57E+02
Y-90	2.30E+01
Y-91	5.59E+02
Y-91m	6.72E+02
Y-92	1.56E+02
Y-93	3.10E+01

Table 11.1-10 – Solid Radioactive Sources
(Sheet 3 of 5)

MEPS, Spent Columns, Supercell
(Conservative Best Estimate Activity)

Isotope	Activity (Ci)	Isotope	Activity (Ci)
[ ]PROP/ECI	[ ]PROP/ECI	[ ]PROP/ECI	[ ]PROP/ECI
[ ]PROP/ECI	[ ]PROP/ECI	[ ]PROP/ECI	[ ]PROP/ECI
[ ]PROP/ECI	[ ]PROP/ECI	[ ]PROP/ECI	[ ]PROP/ECI
[ ]PROP/ECI	[ ]PROP/ECI	[ ]PROP/ECI	[ ]PROP/ECI
[ ]PROP/ECI	[ ]PROP/ECI	[ ]PROP/ECI	[ ]PROP/ECI
[ ]PROP/ECI	[ ]PROP/ECI	[ ]PROP/ECI	[ ]PROP/ECI
[ ]PROP/ECI	[ ]PROP/ECI	[ ]PROP/ECI	[ ]PROP/ECI
[ ]PROP/ECI	[ ]PROP/ECI	[ ]PROP/ECI	[ ]PROP/ECI
[ ]PROP/ECI	[ ]PROP/ECI	[ ]PROP/ECI	[ ]PROP/ECI
[ ]PROP/ECI	[ ]PROP/ECI	[ ]PROP/ECI	[ ]PROP/ECI
[ ]PROP/ECI	[ ]PROP/ECI	[ ]PROP/ECI	[ ]PROP/ECI
[ ]PROP/ECI	[ ]PROP/ECI	[ ]PROP/ECI	[ ]PROP/ECI
[ ]PROP/ECI	[ ]PROP/ECI	[ ]PROP/ECI	[ ]PROP/ECI
[ ]PROP/ECI	[ ]PROP/ECI	[ ]PROP/ECI	[ ]PROP/ECI
[ ]PROP/ECI	[ ]PROP/ECI	[ ]PROP/ECI	[ ]PROP/ECI
[ ]PROP/ECI	[ ]PROP/ECI	[ ]PROP/ECI	[ ]PROP/ECI
[ ]PROP/ECI	[ ]PROP/ECI	[ ]PROP/ECI	[ ]PROP/ECI
[ ]PROP/ECI	[ ]PROP/ECI	[ ]PROP/ECI	[ ]PROP/ECI
[ ]PROP/ECI	[ ]PROP/ECI	[ ]PROP/ECI	[ ]PROP/ECI
[ ]PROP/ECI	[ ]PROP/ECI	[ ]PROP/ECI	[ ]PROP/ECI
[ ]PROP/ECI	[ ]PROP/ECI	[ ]PROP/ECI	[ ]PROP/ECI
[ ]PROP/ECI	[ ]PROP/ECI	[ ]PROP/ECI	[ ]PROP/ECI
[ ]PROP/ECI	[ ]PROP/ECI	[ ]PROP/ECI	[ ]PROP/ECI
[ ]PROP/ECI	[ ]PROP/ECI		

Table 11.1-10 – Solid Radioactive Sources
(Sheet 4 of 5)

RLWI, Solidified Waste Drum, Liquid Waste Solidification Cell
(Conservative Best Estimate Activity)

Isotope	Activity (Ci)
Ba-137m	3.80E+00
Ce-141	7.59E-01
Ce-144	3.67E+01
Cs-137	4.01E+00
Eu-155	1.32E-01
H-3	1.20E-01
Nb-95	2.48E-01
Pm-147	1.01E+01
Pr-144	3.67E+01
Pr-144m	3.51E-01
Rh-103m	2.04E+00
Rh-106	4.04E+00
Ru-103	2.07E+00
Ru-106	4.04E+00
S-35	1.08E-01
Sm-151	8.99E-02
Sr-89	1.24E+00
Sr-90	5.30E-01
U-234	3.86E-02
Y-90	5.30E-01
Y-91	1.71E+01
Zr-95	1.23E-01

Table 11.1-10 – Solid Radioactive Sources (Sheet 5 of 5)

Solid Radwaste, Spent Filters, Supercell
(Conservative Best Estimate Activity)

Isotope	Activity (Ci)
I-131	3.82E-01
I-132	5.13E-02
I-133	1.42E-01
I-135	2.02E-02
Mo-99	4.62E-04
Tc-99m	4.10E-04

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

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

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

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

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

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

11.2.2.2.12 Radioactive Gaseous Waste

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

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

11.2.2.3 Technical Specifications

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

11.2.3 RELEASE OF RADIOACTIVE WASTE

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

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

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

discharge from the RCA are sampled and analyzed prior to discharge. Liquids that are not within limits for discharge are instead disposed of as low-level radioactive waste, while those that are acceptable are manually discharged to the sanitary sewer. Liquid discharge volumes are estimated to be less than 40 gallons weekly.

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

11.2.3.1 Solid Wastes

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

11.2.3.1.1 Irradiation Units

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

11.2.3.1.2 Spent Columns

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

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

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

When a set of columns are to be transferred out of the hot cell, they are remotely loaded into an export waste drum within a shielded cask. Dose rates from the cask and contamination levels are confirmed to be within limits, then the cask is remotely transported to a bore hole for interim

12.5 REPORTS

12.5.1 OPERATING REPORTS

An annual report covering the operation of the facility during the previous calendar year will be submitted to the NRC Document Control Desk within 30 days of the end of the calendar year providing the following information:

- A narrative summary of operating experience including the energy produced by each irradiation unit (IU) or the hours each IU was operating, or both.
- The unscheduled shutdowns including, where applicable, corrective action taken to preclude recurrence.
- Tabulation of major preventative and corrective maintenance operations having safety significance.
- Tabulation of major changes in the facility and procedures, including a summary of the evaluation leading to the conclusions that they are allowed without prior NRC approval.
- A summary of the nature and amount of radioactive effluents released or discharged to environs beyond SHINE's effective control, as determined at, or before, the point of such release or discharge. The summary will include to the extent practicable an estimate of individual radionuclides present in the effluent. If the estimated average release after dilution or diffusion is less than 25 percent of the concentration allowed or recommended, a statement to this effect is sufficient.
- A summarized result of environmental surveys performed outside the facility.
- ~~A summary of exposures received by facility personnel and visitors where such exposures are greater than 25 percent of that allowed or recommended.~~Results of individual monitoring carried out by SHINE for each individual for whom monitoring was required by 10 CFR 20.1502.

12.5.2 SPECIAL REPORTS

Special reports are used to report unplanned events as well as planned major facility and administrative changes. Special reports will follow the schedule below.

There will be a report not later than the following working day by telephone and confirmed in writing by facsimile or similar conveyance to the NRC Operations Center, to be followed by a written report to the NRC Document Control Desk that describes the circumstances of the event within 14 days of any of the following:

- Violation of a safety limit;
- Release of radioactivity from the site above allowed limits;
- Operations with actual safety system settings for required systems less conservative than the limiting safety system settings specified in the technical specifications;
- Operation in violation of limiting conditions for operation established in the technical specifications unless prompt remedial action is taken;
- A safety system component malfunction that renders or could render the safety system incapable of performing its intended safety function, as described in the technical specifications;
- An unanticipated or uncontrolled change in reactivity greater than one dollar;
- Abnormal and significant degradation of the primary system boundary (excluding minor leaks); and

ENCLOSURE 2
ATTACHMENT 2

SHINE MEDICAL TECHNOLOGIES, LLC

SHINE MEDICAL TECHNOLOGIES, LLC OPERATING LICENSE APPLICATION
RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION AND SUPPLEMENT NO. 7

RESPONSE TO REQUEST FOR ADDITIONAL INFORMATION
PUBLIC VERSION

TECHNICAL SPECIFICATION CHANGES
(MARK-UP)

1 page follows

extent practicable an estimate of individual radionuclides present in the effluent. If the estimated average release after dilution or diffusion is less than 25 percent of the concentration allowed or recommended, a statement to this effect is sufficient;

6. A summarized result of environmental surveys performed outside the facility; and

7. ~~A summary of exposures received by facility personnel and visitors where such exposures are greater than 25 percent of that allowed or recommended~~Results of individual monitoring carried out by SHINE for each individual for whom monitoring was required by 10 CFR 20.1502.

5.8.2 Special Reports

Special reports are used to report unplanned events as well as planned major facility and administrative changes. Special reports will follow the schedule below:

1. There will be a report not later than the following working day by telephone and confirmed in writing by facsimile or similar conveyance to the NRC Operations Center, to be followed by a written report to the NRC Document Control Desk that describes the circumstances of the event within 14 days of any of the following:

   a. Violation of a safety limit;

   b. Release of radioactivity from the site above allowed limits;

   c. Operations with actual Safety System settings for required systems less conservative than the limiting safety system settings specified in Section 2.2;

   d. Operation in violation of limiting conditions for operation established in Section 3, unless prompt remedial action is taken as permitted in Section 3;

   e. A Safety System component malfunction that renders or could render the Safety System incapable of performing its intended safety function. If the malfunction or condition is caused by maintenance, then no report is required;

   f. An unanticipated or uncontrolled change in reactivity greater than one dollar;

   g. Abnormal and significant degradation of the PSB (excluding minor leaks); and

   h. An observed inadequacy in the implementation of administrative or procedural controls such that the inadequacy causes or could have caused the existence or development of an unsafe condition with regard to operations.

2. There will be a written report within 30 days to the NRC Document Control Desk of the following:

   a. Permanent changes in the facility organization involving Level 1 or Level 2 management, and