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7.0 RADIATION PROTECTION

7.1 ENSURING THAT OCCUPATIONAL RADIATION EXPOSURES ARE AS LOW AS REASONABLY ACHIEVABLE

Idaho Spent Fuel (ISF) Facility components, equipment, systems, and buildings have been designed to minimize personnel exposure to radiological hazards. Foster Wheeler Environmental Corporation (FWENC) has endeavored to eliminate radiological hazards where possible and to develop engineered features to minimize worker risk where possible.

Facility and equipment layout affects the potential for radiation exposures. The ISF Facility design uses fixed and portable shielding to reduce exposure by isolating radiation sources. Where equipment components constitute a substantial radiation source that cannot be effectively reduced in place, features that permit the removal of such components for maintenance at remote locations were considered for reducing exposures. The ISF Facility design also includes the use of remote-handling features to reduce exposure of facility personnel during fuel handling operations consistent with U.S. Nuclear Regulatory Commission (NRC) Regulatory Guides 8.8 and 8.10 (Refs. 7-1 and 7-2).

The ISF Facility Health Physics Program follows the As Low As Reasonably Achievable (ALARA) guidelines of Regulatory Guide 8.8 and 8.10 and the requirements of Title 10 Code of Federal Regulations (CFR) Part 20, *Standards for Protection Against Radiation* (Ref. 7-3). The following methods will be used to achieve ALARA objectives:

- maintaining a program to monitor occupational radiation doses ALARA
- personnel and organizational level ALARA goals, tracking personnel exposure, and maintaining associated records
- allocating the appropriate technical, administrative, and supervisory resources
- ensuring individual and collective exposures will not exceed the limits recommended for the appropriate circumstances
- controlling access to radiation and radioactive contamination areas
- using procedures and engineered controls (e.g., ventilation, remote handling, and shielding) and monitoring equipment (e.g., continuous air monitors [CAM] and remote area monitors)
- requiring an ALARA review of procedures and work packages for work activities that involve radiological work resulting in 50-millirem (mrem) individual exposure per day, 500-mrem collective exposure per day, or entries into high radiation areas where the general area radiation levels exceed 100 mrem/hr
- ISF Facility design that ensures ALARA by minimizing required maintenance operations, minimizing radiation levels and operating times, and providing contamination control during handling, transfer, and storage of radioactive material

The effectiveness of the ISF Facility Health Physics Program will be evaluated by annual facility audits.

7.1.1 Policy Considerations

ISF Facility management ensures that radiation exposure to personnel and the public and releases of radioactivity to the environment will be ALARA.

The ISF site at the Idaho National Engineering and Environmental Laboratory (INEEL) is a controlled area. The fence surrounding the facility buildings (shown in Figure 4.1-1) establishes the restricted area boundary. Entrance to the restricted area is either through the guard house for personnel, or through the adjacent vehicle access gate. The purpose is to limit access by non-ISF personnel to protect them from undue risk from exposure to radiation and radioactive materials. Personnel will be provided radiation exposure monitoring devices before entering and working in restricted areas of the facility. Visitors will be monitored in accordance with radiation protection procedures before entering the restricted area or any radiologically controlled area in the facility. Visitors are not allowed in areas of airborne radioactivity or contamination.

It is ISF Facility management policy to maintain occupational radiation exposure ALARA. Personnel will be aware of this commitment through an integrated program of employee training, procedural reviews, administrative exposure controls, engineering controls, and annual exposure reviews. Facility personnel will be committed to maintaining personnel exposures within regulatory limits and ALARA. Work processes that have the potential to expose workers to radiation or radioactive materials will be reviewed by the ISF Facility Environmental Safety and Health (ES&H) Manager to ensure that any exposures or releases are ALARA.

ISF Facility management responsibilities include the following:

- Establish and administer a program of occupational radiation controls and safety to minimize the exposure of facility personnel to ionizing radiation.
- Ensure that personnel responsible for performing radiological work activities are appropriately trained in the facility ALARA Program. Standards are established to ensure the technical competency of personnel through implementation of standardized and mandated radiological training and development programs, as appropriate. Section 9.3 discusses the ISF Facility training program.
- Ensure the technical competence of personnel responsible for implementing and overseeing the radiological control program. An appropriate level of technical competence gained through education, experience, and job-related technical and professional training is a critical component for achieving the goals of radiological control. Qualification requirements commensurate with this objective are established for technical and professional radiological control program positions. The requirements are consistent with applicable industry standards and promote professional development and excellence in radiological performance.
- Establish and maintain, from the lowest to the highest levels, line management involvement and accountability for departmental radiological performance and goals. The responsibility for compliance with radiological protection requirements, and for minimizing personnel radiation exposure, starts at the worker level and broadens as it progresses upward through the line organization. The ISF Facility managers will be fully responsible for radiological performance

within their programs and will take necessary actions to ensure that requirements are implemented and that performance is monitored and corrected as necessary.

- Ensure that radiological measurements, analyses, personnel monitoring results, and estimates of public exposures are accurate and appropriately made. The capability to accurately measure and analyze radioactive materials and workplace conditions, and determine personnel radiation exposure is fundamental to the safe conduct of radiological operations.
- Conduct radiological operations in a manner that controls the spread of radioactive materials and reduces radioactive exposure to personnel and the public, and uses a process that seeks to maintain exposure levels ALARA. Radiological operations and activities will be preplanned to allow for the effective implementation of dose and contamination reduction and control measures. Operations and activities include reasonable controls directed towards reducing exposure, preventing the spread of radiological contamination, and minimizing the generation of contaminated wastes and the release of effluents.
- Conduct oversight to ensure strict compliance with regulatory requirements regarding procedures, radiation exposures, and release of radioactive materials.

7.1.2 Design Considerations

The ISF Facility design incorporates design features and improvements from previously constructed independent spent fuel storage installations such as the Fort St. Vrain facility in Colorado. Section 4.2 discusses the ISF design features and layout. The ISF Facility design ensures ALARA by minimizing required maintenance operations, minimizing radiation levels and operating times, and providing contamination control during handling, transfer, and storage of radioactive material. Specific design features of the ISF Facility for ensuring ALARA include:

- concrete shield walls in the Transfer Tunnel and Fuel Processing Area (FPA) that minimize the onsite and offsite dose
- concrete shield plugs used to reduce the dose to personnel during canister transfer, storage, and maintenance operations
- heavy shielded transfer trolley for handling and transfer operations to ensure that the dose to personnel and the general public is minimized
- remote handling technology to minimize direct and scattered radiation exposure from fuel handling operations
- fixed and portable shielding to minimize direct and scattered radiation exposure
- passive storage system design for long-term storage that requires minimal maintenance
- proven procedures and experience to control contamination during canister handling and transfer operations
- automatic welding system for canister closure weld
- shielded canister handling machine (CHM) for loading ISF canisters into storage tubes
- shielded area for fuel handling machine (FHM) maintenance

- high efficiency particulate air (HEPA) filters in the FPA that are remotely replaceable

During facility design the ALARA program plan provided direction to project design engineers and cognizant reviewers for proper implementation of radiological safety principles and as ALARA philosophy into the facility design. A series of checklists specific to engineering and design discipline were compiled to address the ALARA design guidelines in these Regulatory Guides 8.8 and 8.10. Design features were evaluated against the applicable checklist(s) to help identify choices between otherwise comparable alternatives affected by ALARA considerations for reduction of collective doses from direct radiation. Choices included shielding thickness of facility walls, shielding of the canister trolley, and shielding of the canister handling machine. One example is the thickness of the storage vault wall adjacent to the transfer tunnel. Initial shielding calculations showed that controlling the placement of loaded canisters in the storage vault with a vault wall of three foot of concrete will provide an equivalent reduction in collective dose from direct radiation through the vault wall as a four foot concrete wall.

The ISF Facility has areas designed specifically for decontamination of equipment (e.g., cask decontamination area). Section 4.4, *Decontamination Systems*, summarizes the expected equipment decontamination operations. Concrete floors and other surfaces that have the potential to be contaminated will be sealed to minimize entrainment of contamination and to facilitate decontamination. General facility areas found to have removable contamination activity greater than the limits identified in Section 7.5 will be controlled as contamination areas or decontaminated using standard techniques.

Regulatory Position 2 of Regulatory Guide 8.8 is incorporated into design considerations, as described below (Ref. 7-1):

- Regulatory Position 2a on access control is met by use of a fence with a controlled access gate that surrounds the ISF site area and barriers within the ISF Facility, which prevent unauthorized access.

- Regulatory Position 2b on radiation shielding is met by the heavy shielding of the transfer cask, shielded canister trolley, and shielded CHM, which minimizes personnel exposures during transfer cask reception, canister transfer, and placement into canister storage. The designs of the Storage Area air inlet and outlet ducts prevent direct radiation streaming. The FPA, Storage Area, and Transfer Tunnel have thick concrete walls designed to shield personnel in adjacent areas from high dose rates that could be associated with spent fuel transfer, packaging, and storage operations.

- Regulatory Position 2c on process instrumentation is met using an integrated data collection system (IDCS). The IDCS consists of a computer-based system capable of monitoring, recording, and reporting data provided by systems throughout the facility. The purpose of the IDCS is to provide a centralized collection point for data generated by systems, instrumentation, and monitors throughout the ISF Facility.

- Regulatory Position 2d on control of airborne contaminants is met by providing protection against airborne radioactive material using engineering controls. The ventilation system design incorporates HEPA filters within the ventilation ducting to minimize the spread of contamination through the ventilation system. Maintaining air pressure gradients and airflow from areas of low potential airborne contamination to areas of higher potential contamination minimizes the spread of airborne contamination. A confinement boundary is established to minimize the spread of airborne contamination from facility areas where fuel is directly handled to adjacent areas.

- Regulatory Position 2e on crud control is not applicable because there are no systems that produce crud.
- Regulatory Position 2f on decontamination is met by including redundant ventilation capabilities to allow isolation of components for servicing. Surfaces of the FPA, Transfer Tunnel, Canister Closure Area (CCA), and Solid Waste Processing Area (SWPA) walls and floors are painted with a special paint that is easily decontaminated.
- Regulatory Position 2g on radiation monitoring is met with the use of area radiation monitors (ARM) for monitoring general area dose rates. Continuous air monitoring (CAM) will be located throughout the facility to provide prompt warning of airborne radioactivity. A criticality monitoring system will be located around the FPA to provide prompt warning of a criticality event.
- Regulatory Position 2h on resin treatment systems is not applicable to the ISF Facility because there will not be any radioactive systems containing resins.
- Applicable portions of Regulatory Position 2i concerning other miscellaneous ALARA items is met through ISF Facility design features that provide a favorable working environment to promote work efficiency (paragraph 2i[13]). These include adequate lighting, ventilation, working space, and accessibility.
- Regulatory Position 2i(15) is met because the emergency lighting system is adequate to permit prompt egress from any high radiation areas if the facility lighting fails.

7.1.3 Operational Considerations

ISF Facility management's policy is to keep occupational radiation exposures ALARA. Therefore, ISF Facility operations personnel will follow specific plans and procedures to ensure that ALARA goals are achieved consistent with the intent of NRC Regulatory Guides 8.8 and 8.10 and the requirements of 10 CFR 20, Subpart C (Refs. 7-1, 7-2, and 7-3). Section 5.2, *Spent Fuel Handling Systems*, discusses the operations and maintenance at the ISF Facility.

Operations and maintenance activities that could involve significant radiation exposure of personnel will be carefully planned. They will use previous operating experience and will be carried out using trained personnel and proper equipment. Where applicable, formal ALARA reviews will be prepared that specify radiation exposure reduction techniques, such as those set out in NRC Regulatory Guide 8.8.

Regulatory Position 4 of Regulatory Guide 8.8 will be met by the following:

- a low-radiation background counting room with the following counting room instrumentation available to support facility personnel protection and to ensure that both exposures and releases are ALARA: multi-channel gamma pulse height analyzer (e.g., high-purity germanium [HPGe] for material protection measurements, low-background alpha-beta radiation proportional or scintillation counter, end-window G-M counters, and liquid scintillation counter (Table 7.1-1)
- portable instrumentation available to radiation protection personnel measuring dose rates and radiation during facility operations (Table 7.1-2)

- personnel monitoring instrumentation available including G-M friskers, direct reading pocket dosimeters, alarming dosimeters, thermoluminescent dosimeter (TLD) badges, hand and foot monitors and portal monitors (Table 7.1-2)
- protective equipment available, including anti-contamination clothing and respirators, for facility personnel

Regulatory Guide 8.10 will be incorporated into the ISF Facility operational considerations as described below:

- Facility personnel will be aware of management's commitment to keep occupational exposures ALARA.
- Periodic formal reviews will be performed to determine how exposures might be lowered.
- There is a well-supervised radiation protection staff with well-defined responsibilities.
- Facility workers receive sufficient training.
- The facility Radiation Safety Officer has authority to enforce safe facility operation.
- Operating and maintenance procedures and equipment and facilities will be modified where they substantially reduce exposures at a reasonable cost.
- The radiological control staff understands the origins of radiation exposures in the facility and seeks ways to reduce exposures.
- Adequate equipment and supplies for radiation protection work will be provided.
- Tradeoff between the frequency of monitoring or maintenance activities (and the exposures that would result) and the potential hazards associated with reduced frequency of these activities will be evaluated to minimize unnecessary exposure to workers and maintain occupational exposures ALARA.
- The loading sequence (location of repackaged fuel within the storage vault) will be evaluated to ensure exposures through the vault walls are minimized and associated occupation exposures are ALARA.
- Practices involving radiation exposure will not be undertaken unless it produces a net benefit.
- ALARA exposures will consider technological and economic factors.

7.2 RADIATION SOURCES

The major source of radiation at the ISF Facility is the spent nuclear fuel (SNF) that will be transferred to the facility for packaging, interim storage, and eventual retrieval for transport to a long-term repository. The three basic fuel types to be handled at the facility are Peach Bottom fuel elements, TRIGA fuel elements, and Shippingport reflector modules. Each type of spent fuel is discussed below.

Section 3.1.1, Materials to be Stored, discusses the spent fuel types to be handled at the ISF Facility.

Other sources of radiation will include solid radioactive waste and liquid radioactive waste. Chapter 6 discusses the generation and management of solid and liquid radioactive waste at the ISF Facility. Table 6.3-1, *Estimated Concentrations of Principal Radionuclides in Liquid Waste*, and Table 6.4-1, *Characterization and Estimated Annual Activity of Principal Radionuclides in Solid Waste*, provide the estimated total annual activity for each radionuclide.

7.2.1 Characterization of Sources

The ORIGEN2 (Oak Ridge Isotope Generation) computer code was used to derive the radioisotope activity, photon energy spectra, decay heat, and neutron generation rate associated with each fuel type (Ref. 7-4). The U.S. Department of Energy (DOE) provided original radionuclide source term data for each fuel type to FWENC. This source term accounts for activation products associated with the activation of non-fuel components of the fuel assembly. Induced radioactivity by interactions involving neutrons originating in the fuel is not a significant source term due to the low fluence rate of neutrons originating in the fuel. This data served as the baseline reference data inserted into the ORIGEN2 program and then decayed to obtain the desired output. Data was decayed to July 1, 2004, the projected ISF Facility operational start date.

The source term is summarized below in three categories: (1) radionuclide composition, (2) photon production rate, and (3) neutron production rate.

Radionuclide Composition. Table 7.2-1 through Table 7.2-5 summarize the isotopic activity for the activation products, actinides and daughters, and fission products for each of the fuel types as of July 1, 2004. This data is the result of ORIGEN2 decay analyses. These radionuclide activities are used as the basis for solid and liquid waste volumes discussed in Chapter 6 and the shielding and dose analyses presented in this chapter.

Photon Production Rate. The ORIGEN2 photon database contains and displays the number of photons per decay in an 18-energy group format.

Table 7.2-6 provides the photon production rate (photons/second-assembly) resulting from the ORIGEN2 analysis. Photon production rates are tabulated by energy group number for each of the fuel types expected to be shipped to the ISF Facility. These photon production rates are used as inputs for the facility dose rate calculations discussed in Section 7.4.

Neutron Production Rate. Neutron production tables associated with each fuel type (neutrons/second-cm^3) are provided in Table 7.2-7. The neutron production rates are summarized for both (alpha, n) reactions and spontaneous fission events. These tables are generated for actinides and their daughter products because only these radionuclides emit significant numbers of spontaneous neutrons or alpha particles. The neutron spectrum presented in each table represents the bounding intensity for each fuel type. These neutron production rates are used as inputs for the facility dose rate calculations discussed in Section 7.4.

7.2.2 Airborne Radioactive Material Sources

The design and layout of the ISF Facility minimize the number of areas easily accessible to or normally occupied by operating personnel that have the potential for airborne radioactive material. The two areas with the greatest potential for airborne radioactive materials are the fuel handling machine (FHM) Maintenance Area and the SWPA.

Airborne contamination is expected to be present in the FPA. The FHM Maintenance Area is adjacent to the FPA and is separated by a retractable shield door. Airflow is designed to cascade from the Maintenance Area into the FPA. Facility personnel do not normally occupy the Maintenance Area. Facility personnel will enter the Maintenance Area to perform maintenance work on the FHM as needed. Facility personnel will not normally enter the FPA while fuel is present. Access may be required during change-out operations in preparation for other fuel types. The FPA is designed for remote operations and any maintenance of equipment will be performed remotely. Before anyone enters the FHM Maintenance Area, the airborne contamination concentrations will be evaluated and the information will be used to select the appropriate personnel protective equipment (PPE).

Airborne releases from handling waste stream materials are possible in the SWPA. Process-generated waste (paper, plastic PPE, spent filters, and empty fuel canisters) is size reduced, consolidated, and compacted for disposal in the SWPA. Table 7.2-8 identifies the nuclides and calculated concentrations of airborne radioactivity expected. The concentration of airborne radioactivity is based on the calculated annual waste inventory per isotope, multiplied by an airborne release factor from American Nuclear Society/American National Standard (ANSI/ANS)-5.10, *Airborne Release Fractions at Non-Reactor Nuclear Facilities*. Airborne release factors of 0.00004 for particulates and 1.0 for gases are used (Ref. 7-5).

7.3 RADIATION PROTECTION DESIGN FEATURES

Equipment and installation features of the ISF Facility are provided in Chapter 4. This section discusses key design features used for ensuring that occupational exposures to radiation are ALARA.

One of the ISF radiological design goals was to ensure that personnel exposures are ALARA. Various methods were used to evaluate installation design features ensure personnel exposures are ALARA. For the ISF, the design goal was achieved by following:

- Perform initial calculations for the various fuel configurations to determine initial doses
- Modify the shielding thickness and fuel configurations (i.e., placement of fuel in storage vault) as necessary to maintain personnel exposures ALARA
- Reevaluate the design and results, as new information becomes available.

Following this process resulted in the design shielding thickness identified below:

Area/Equipment	Design Shielding Thickness	Shielding Material
Transfer Tunnel (east wall)	3'	Concrete
Transfer Tunnel (west/south wall)	3'	Concrete
Fuel Packaging Area	4'	Concrete
Storage Vault (walls)	3'	Concrete
Storage Vault (charge face)	30"	Concrete/steel
Canister Trolley	11" (radial shielding)	Carbon steel
Canister Shield plugs	5" to 10"	Stainless steel
Canister Handling Machine	12"/4"	Steel/Jabroc

This process works well for average doses and dose rates. There is still the potential for locally higher levels, and periodic changes in work patterns. The calculations generally accounted for much of this, and are considered conservative. As an added precaution, temporary shielding is available to further reduce exposures as necessary.

Figure 6.3-2 provides a scaled layout showing:

- liquid waste storage tank area
- solid waste processing area
- solid waste storage area

Figure 7.3-1 through Figure 7.3-2 provides a layout showing:

- controlled access areas
- radiological control area designations
- location of health physics office, counting room, and decontamination shower
- location of spent fuel during storage/fuel packaging

Figure 7.3-3 through Figure 7.3-8 show:

- location of CAMs, ARMs, and criticality monitors
- location of frisking stations and change areas

Figure 7.3-9 and Figure 7.3-10 show the shield wall dimensions for the facility.

7.3.1 Installation Design Features

The design considerations listed in Section 7.1.2 ensure that occupational exposures are ALARA and that a high degree of integrity is obtained for the confinement of radioactive material. Key design features for the facility are briefly discussed in the following subsections.

7.3.1.1 Access Control of Radiation Areas

Access to the ISF Facility is controlled in accordance with 10 CFR 72.106 (Ref. 7-6). The INEEL site boundary establishes the controlled area boundary. The ISF site area fence (owner controlled fence) establishes the restricted area to protect non-ISF personnel from undue risk from exposure to radiation and radioactive materials in the ISF Facility.

In accordance with 10 CFR 72.104, the annual dose to an individual beyond the INEEL controlled area boundary is less than 25 mrem. Access to radiologically controlled areas within the ISF Facility site area fence, where there is potential for radiation fields to exceed 2 mrem/hr, is controlled by physical or administrative controls. Physical controls include warning signs, beacons, access door interlocks, or key locks. Administrative controls include the use of radiation work permits, ALARA job reviews, or stay-time monitoring.

Radiologically controlled areas may require further designation as high or very high radiation areas according to 10 CFR 20.1003. Access to these areas is controlled to preclude inadvertent and unnecessary exposure. Access control features may include physical barriers, locked entryways, and audible and visible alarm signals. Access to high and very high radiation areas is controlled in accordance with regulatory position 2.4 of Regulatory Guide 8.38 (Ref. 7-7). Figure 7.3-1 and Figure 7.3-2 provide the facility layout identifying the expected restricted areas and their classification (i.e., Radiological Control Area, Radiation Area).

Figure 4.1-1 provides a site layout showing the ISF Facility restricted area. A fence to preclude ready access to the radiologically controlled area surrounds the ISF Facility. Vehicle gates and personnel gates are also identified in this figure.

7.3.1.2 Radiation Shields and Geometry

Fixed radiation shielding is integral to the ISF Facility design for reducing personnel exposure to radiation. The main shielding features of the facility are described in the following paragraphs. Section 8.2.4.2 discusses off-normal and accident conditions associated with loss of shielding.

The Cask Receipt Area does not provide shielding. SNF is contained within the shielded Peach Bottom transfer casks when being handled inside the Cask Receipt Area. Workers in the Cask Receipt Area will be shielded from radiation during fuel transfer operations inside the Transfer Tunnel by shielding in the

Transfer Tunnel and on the shielded canister trolley. The CHM and Storage Area vaults provide shielding during canister transfer and storage.

The Transfer Tunnel provides a shielded route for the cask trolley to safely travel between the Cask Receipt Area and the FPA, via the Transfer Tunnel decontamination and maintenance area. It also provides the route for the canister trolley to safely travel between the FPA, the CCA, and the Storage Area. The Transfer Tunnel east wall adjacent to the FPA is approximately 3 feet thick. The east wall adjacent to the Storage Area vault is approximately 3 feet thick to provide the necessary radiation shielding in case personnel need to access a trolley while SNF is in the FPA bench vessels or Storage Area vault. The west wall of the Transfer Tunnel is 3 feet thick. During fuel transfer operations from the cask into the FPA there is a small (~2 inches) gap between the FPA port and the cask. The tunnel ceiling and walls minimize the dose rate in adjacent occupied areas by shielding radiation streaming through this gap during fuel transfer. The tunnel is controlled as a high radiation area during fuel transfers.

The FPA and the FHM Maintenance Area are designed to minimize radiation exposure to operations and maintenance personnel. The walls of the FPA are 4 feet thick, the north and south walls of the FHM are 4 feet thick, and the east and west walls are 3 feet thick to provide protection from gamma and neutron radiation.

The Storage Area vaults provide radiation shielding from the SNF stored in this structure. The 3 foot walls and 30 inch charge face are designed with thick concrete sections and design features to minimize the radiation dose rates in adjacent areas that can be occupied by personnel at the ISF Facility.

A shielded canister trolley is used to minimize operator exposure during canister closure operations and canister trolley recovery maintenance in the Transfer Tunnel. The canister trolley has three primary operating positions: CCA port, ISF canister port (below the FPA), and Storage Area load/unload port. At each of these positions the onboard jacking system will jack the nose of the shielded cask into a recess in the floor of each of the operational areas. This provides two benefits: 1) it alleviates the radiation streaming that would occur if the SNF transfers were done in an unshielded position, and 2) it reduces the open free area of the transfer port aperture and therefore minimizes the disturbance to the ventilation system. The canister cask walls are 11 inch thick steel. The trolley shielding is designed to reduce maximum exterior exposure rates at 1 foot to less than 25 mR/hr for recovery/repair operations. ITS components located on the canister trolley or that may come near the cask are either radiation tolerant or shielded to ensure adequate availability of the canister trolley and the control system.

The CHM is designed to ensure that dose rates to the operator and personnel on the floor of the Storage Area remain ALARA during canister handling operational phases. The main cask body contains the cavity for the ISF canister and provides radiation protection using steel (for gamma shielding), clad with a layer of Jabroc for neutron shielding. The shield skirt and storage tube shield plug ensure that streaming effects are minimized during transfer operations over the open fuel tube.

7.3.1.3 Control of Airborne Contaminants and Gaseous Radiation Sources

ISF Facility ventilation design features use engineered control to protect against airborne radioactive material. Section 3.3.2.2, *Ventilation and Off-Gas Systems*, and Section 4.3.1, *Ventilation and Off-Gas Systems*, provide details of the ventilation design.

The ISF ventilation system is designed to confine contamination by filtration, maintain differential pressures between confinement zones, and ensure that air flows from areas of low potential contamination toward areas of high potential contamination. Heating and air conditioning components are designed to provide environmental control that ensure that structures, systems, and components operate within designed temperature parameters and interior temperature control for personnel comfort.

The system is designed to:

- prevent the accidental release of radiological hazards to the environment
- keep personnel exposure to radiological hazards ALARA
- control the spread of radioactive materials and controlling contamination between areas
- limit the spread of radioactive materials within the ventilation system

The ventilation system is designed to meet the 40-year operational life expectancy of the facility. Components with a potentially shorter life are designed and installed to permit replacement with minimal impact on operations and maintain personnel exposure ALARA. Provisions are made for the routine maintenance of HVAC components to maximize their operational life.

7.3.2 Shielding

Fixed radiation shielding constitutes the primary method of reducing occupational exposure. Radiation shielding design is based on the results of shielding analysis performed using a shielding calculation methodology known to provide reliable and accurate results. The calculation method used was Monte Carlo based analysis (MCNP4B2 Monte Carlo N-Particle). Shielding models were evaluated for the key facility features; each model is discussed in the following subsections.

MCNP (Ref. 7-8) was selected as the primary code for evaluating shielding models associated with the ISF, because of the need to accurately represent the complex geometric arrangements and the wide range of materials that were needed to describe the models. It was determined that a Monte Carlo type of code met this need, and MCNP is the most acceptable widely available code. It can perform these calculations and provide the flexibility to accurately represent the conditions. Monte Carlo approaches also provide the capability to accurately assess streaming problems typical of equipment handling with high radiation sources. The other option, a point kernel type of calculation approach, does not provide the ability to account for the complexity of the model. The point kernel type of code cannot model the combination of complex geometry coupled with many different materials that are present in the models and it also cannot perform streaming calculations. As a result of these calculational requirements, MCNP was chosen as the primary code for performing the shielding analysis associated with the ISF project. Section 3.3.4.3.2 discusses the validation of the MCNP code relative to analysis performed in support of this SAR.

A loaded ISF TRIGA canister source term consisting of two baskets containing 54 elements each (a total of 108 elements) was used as the source term for the shielding models evaluated. The TRIGA source term was selected because it provided the bounding photon flux for the three fuel types. Table 7.3-1 provides a comparison of the photon spectrum for the various fuel types to be received at the facility. TRIGA has the highest photon flux per canister due to the number of fuel assemblies handled per canister (108) versus 10 assemblies per canister for Peach Bottom fuels and one assemble per canister for Shippingport reflector modules and its characteristic radial peaking factor of 2.62 (Ref. 7-9).

Each fuel assembly in the basket is modeled by MCNP to determine the number of photons emitted from the assembly that reach the shielding surface (i.e. self-shielding in the fuel).

The MCNP was used for the neutron dose rate calculations. Neutron capture provided by neutron absorbing material was used when modeling the canister handling machine. Four inches of Jabroc 'N' were included in the model to provide additional neutron shielding. The material composition of shielding material used is provided in Table 7.3-2. MCNP uses the Monte Carlo method to theoretically duplicate a statistically significant process, such as the interaction of nuclear particles with materials. Neutron fluence-to-dose conversion factors were conservatively taken from ANSI/ANS6.1.1, *Neutron and Gamma-ray Fluence-to-dose Factors*. For the design temperature ranges presented in Chapter 3, the shielding properties of the materials used for shielding will not be affected.

7.3.2.1 Cask Receipt Area

The Peach Bottom transfer cask will be used to transport the SNF to the ISF Facility. The cask was originally designed to shield irradiated and unirradiated fuel such that the radiation levels on contact do not exceed 100 mrem/hr. Shielding analysis was performed to determine the dose rates for handling TRIGA fuel. The results of the shielding analysis show that the expected surface dose rates are well below the Peach Bottom cask design criteria of 100 mrem/hr on contact. The peak combined neutron/gamma, dose rate at 1foot will be about 3 mrem/hr.

Calculated Dose Rates for Peach Bottom Cask with TRIGA Fuel

Axial from Top/Bottom		Radially from Side	
Distance (ft)	(mrem/hr)	Distance (ft)	(mrem/hr)
0.2	$1.0x10^{-02}$	0.2	7.6
1	$7.5x10^{-03}$	1	3.4
6	$1.1x10^{-03}$	5	0.6

The actual contact dose rate is expected to be less because the incoming shipments will contain only 90 TRIGA fuel rods.

The TRIGA fuel was modeled in an ISF canister with a 0.375 inch (0.95cm) thick, 316 stainless steel canister around it inside the Peach Bottom cask to represent the buckets and cans used during the Peach Bottom cask transfer of TRIGA fuel. The Peach Bottom cask cavity wall of the inner shell is 0.25 inch thick stainless steel. The lead shielding contained between the inner and outer shell is 6.25 inches thick in the 110 inch central section, and 5.25 inches thick at the cask ends (30 inch length at each end). The outer shell is 1.5 inch-thick mild steel clad with a 0.25 inch layer of stainless steel. The calculations were made assuming the stainless steel lids will be used for the fuel transfers to the ISF Facility. The top lid comes in two separated sections; one is 8.47 inches thick and the other is 7 inches thick, for a combined top shield thickness of 15.47 inches of stainless steel. Figures 7.3-11 and 7.3-12 depict the radial and axial geometry odeled.

The following table provides the source and shielding geometry used to estimate the dose rates.

Peach Bottom Cask Shielding Geometry/Source Term

Source Geometry	**Cylindrical volume**	**56" length**	**18" diameter**
Source Strength	108 TRIGA assemblies (2 canisters)		
Shields	**Thickness (inches)**	**Material**	**Source Shield Distance (inches)**
Fuel canister	0.4	Stainless steel	0
Air gap	4	Air	0.4
Inner cask wall	0.3	Stainless steel	4.4
Lead shielding	6.3	Lead	4.7
Outer cask wall	1.5	Steel	11
	0.3	Stainless steel	12.5
Cask Exterior			12.8

7.3.2.2 Fuel Packaging Area

The purpose of these analyses was to determine the dose rate on the far side of an FPA cell wall from a TRIGA canister filled with 108 TRIGA fuel elements. The dose rates were calculated from the center of the canister to a dose point 206 cm away (which is reasonably representative of the operator position relative to the canister) through 2, 3, and 4-foot thick concrete walls. Based on the dose rates calculated, a wall thickness of 4 feet was selected for the design of the FPA walls. The results of the calculations are shown in the following table.

Dose Rates from TRIGA Fuel

Concrete Thickness (ft)	**Dose Rate mrem/hr**
2	183
3	2.82
4	0.044

The TRIGA fuel was modeled in a canister with a 0.375 inch thick sidewall. The shield wall thickness was varied from 2 to 4 feet of concrete. Figures 7.3-13 and 7.3-14 depict geometry modeled.

The following table provides the source and shielding geometry used to estimate the dose rates.

Fuel Packaging Area Shielding Geometry/Source Term

Source Geometry	**Cylindrical volume**	**28" length**	**18" diameter**
Source Strength	54 TRIGA assemblies (1 canister)		
Shields	**Thickness (inches)**	**Material**	**Source Shield Distance (inches)**
Fuel canister	0.4	Stainless steel	0
Air gap	12	Air	0.4
Shield wall	48	Concrete	12.4
Exterior			60.4

7.3.2.3 Canister Closure Area

The streaming analysis of the CCA covers the evaluation of the dose rates associated with work directly above and around the canister closure welding and inspection operations. The primary objective is to determine the dose rate as a function of the fabrication tolerances that can be used to fabricate the shield plug and canister while still meeting the ALARA requirements associated with the operation. This evaluation presents the streaming as a function of gap size. The TRIGA fuel elements were selected for the analysis because they represent the bounding photon flux for the three fuel types.

There are two gaps of concern: 1) the gap between the canister and the canister collet; and 2) the gap between the shield plug inside the canister and the inner wall of the canister.

The MCNP analysis has been performed for the canister centered in the canister collet with an effective radial gap of 0.050 inch between the collet and the outside surface of the canister. The dose rate 1 inch above and in line with the 0.050 inch outer canister/collet gap was 26 mrem/hour. The 0.050-inch gap is the maximum expected with the collet fully compressing the canister.

The minimum expected shield plug/canister gap of 0.1 inch gives a calculated dose rate of 50 mrem/hr. The MCNP analysis was performed for the shield plug centered in the canister with an effective radial gap of 0.10 inch between the shield plug and the inside surface of the canister. A maximum gap of 0.2 inch gap is expected with the shield plug offset in the canister. The dose rate with the 0.20 inch gap was calculated to be 143 mrem/hour. The dose rates at 1 inch above and directly over the canister/shield plug of various gap sizes with the lid-off and the lid-on are provided as follows:

Canister Shield Plug Gap Dose Rates

Canister/Shield Plug Gap Width	Dose Rate with Canister Lid
(inches)	(mrem/h)
0.1250	3
0.2000	6
0.3125	9
0.6250	30

The CCA was also evaluated for streaming dose before the canister lid is placed on the canister. The following table shows the dose rates calculated at various locations around the canister.

Dose Rates at Various Locations Around Canister

Location	Dose Rate (mrem/h)
1 inch above plug/canister gap	194
1 inch above canister/wall gap	190
6.5 feet radially out from can axis & in plane of top of lid	0.1
In CCA below plane of top of lid	0.03
In CCA above plane of top of lid	0.2

The basic layout of the canister, the cap, and the collet are shown in Figure 7.3-15. This configuration has been modeled in detail with the MCNP code. Figure 7.3-16 and Figure 7.3-17 provide the top portion of the canister and the complete canister respectively. The TRIGA fuel was modeled in the canister with a 0.375 inch thick sidewall.

The following table provides the source and shielding geometry used to estimate the dose rates.

Canister Closure Area Shielding Geometry/Source Term

Source Geometry	**Disk**	**18" diameter**	
Source Strength	54 TRIGA assemblies (1 canister)		
Shields	**Thickness (inches)**	**Material**	**Source Shield Distance (inches)**
Cover plate	1.5	Stainless steel	0
Void	2	Air	1.5
Shield plug	10	Lead	3.5
Void	4	Air	13.5
Impact limiter/lid	3	Stainless steel	17.5
Exterior			20.5

7.3.2.4 Canister Trolley

The canister trolley includes a steel cask that holds the ISF canister and provides shielding during transfer operations. The TRIGA fuel was modeled with a 0.375 inch thick canister sidewall around it. The canister was placed in an 11 inch thick steel cask with an inside diameter of 28 inches. The canister contains 108 TRIGA fuel elements, in two baskets containing 54 elements each. Figures 7.3-18 and 7.3-19 provide the geometry modeled. The following table provides the source and shielding geometry used to estimate the dose rates.

Shielded Canister Trolley Shielding Geometry/Source Term

Source Geometry	**Cylindrical volume**	**56" length**	**18" diameter**
Source Strength	108 TRIGA assemblies (2 canisters)		
Shields	**Thickness (inches)**	**Material**	**Source Shield Distance (inches)**
Fuel canister	0.4	Stainless steel	0
Air gap	5	Air	0.4
Shielding	11	Stainless steel	5.4
Exterior			16.4

The dose rates were calculated radially outward from the side of the cask on the mid-plane of the upper basket of fuel. The maximum dose rate at 1 foot from the shielded canister trolley is 14 mrem/hr. The dose rates calculated for various distances from the shielded canister trolley are provided in the following table.

Dose Rates from Canister Trolley

Radial Distance from Cask Surface (inches)	Dose Rate (mrem/h)
1	31
12	14
60	3
120	1

7.3.2.5 Storage Area

The dose rate with a 5 by 5 grid of storage tubes containing TRIGA fuel was modeled to determine the dose rate through various thicknesses of concrete. Figure 7.3-20 provides the geometry modeled. The table below provides the calculated dose rates at 1 foot from a concrete wall of given thickness.

The following table provides the source and shielding geometry used to estimate the dose rates.

Storage Area Shielding Geometry/Source Term

Source Geometry	**Slab**	**190" length**	**190" width**
Source Strength	2700 TRIGA assemblies (25 canisters)		
Shields	**Thickness (inches)**	**Material**	**Source Shield Distance (inches)**
Fuel canister	0.4	Stainless steel	0
Air gap	12	Air	0.4
Shielding	36	Stainless steel	12.4
Exterior			48.4

The number of canisters to include in the model was determined by staring with a single canister 1 foot from the interior storage vault wall (closes storage location) and calculating the dose rate 1 foot from exterior of the storage vault wall. Additional canisters were added forming a row until the contribution to the dose point by additional canisters was minimal. A row of five canisters was found to provide the maximum dose. Next a second row was added to the model making a 5 x 2 array and the same process followed until a 1 foot dose was reached where additional rows did not increase the calculated dose. A 5 x 5 array of canister was found to be the limiting number of canisters. The addition of additional rows did not significantly contribute to the calculated dose.

Dose Rate 1 Foot from a Concrete Wall with 25 Storage Tubes Holding TRIGA Fuel

Wall Thickness (feet)	Dose Rate (mrem/hr)
1	59,886
3	15

The wall thickness of the storage area is designed to be approximately 3 feet of concrete. Initial shielding calculations show that, by controlling the placement of loaded canister in the storage vault, an equivalent reduction in the dose rate of that from a 4-foot wall will be achieved.

7.3.2.6 Canister Handling Machine

The CHM is a fully shielded crane used to transfer loaded ISF canisters from the canister trolley to the storage vault tubes. The CHM will be positioned over the storage vault tube location and the shield skirt lowered to provide shielding during placement of the ISF canister into the storage tube. The CHM will also be used in the future to remove ISSF canisters from the storage vault and transfer them to a transport cask for offsite shipment.

The CHM was modeled using an ISF storage canister loaded with 108 TRIGA assemblies in two baskets of 54 assemblies each. The dose rates were calculated radially outward from the side of the CHM on the mid-plane of the upper basket of fuel. The following table provides the calculated dose rates at radial distances from the CHM. Figures 7.3-21 and 7.3-22 depict the axial and radial geometry modeled.

Radial Dose Rates from CHM

Distance from CHM Surface (cm)	Dose Rate (mrem/h)
2.54	5.02
30.48	2.39
152.44	0.52
304.80	0.21

The following table provides the source and shielding geometry used to estimate the dose rates.

Canister Handling Machine Shielding Geometry/Source Term

Source Geometry	Cylindrical volume	56" length	18" diameter
Source Strength	108 TRIGA assemblies (2 canisters)		
Shields	**Thickness (inches)**	**Material**	**Source Shield Distance (inches)**
Fuel canister	0.4	Stainless steel	0
Air gap	4	Air	0.4
Metal Shielding	12	Stainless steel	4.4
Jabroc	4	Jabroc	16.4
Exterior			20.4

7.3.2.7 Waste Processing Area

Processing irradiated fuel results in the generation of radioactive waste materials that will ultimately be processed in the SWPA. This area is designed to receive waste directly from the FPA, either through ceiling waste ports or the hatch into the Crane Maintenance Area portion of the FPA, and from other areas of the facility. Waste generated in the facility other than in the FPA is expected to dose rates less than 5 mrem/hr and are not expected to pose personnel exposure concern. Waste generated in the FPA (e.g., empty fuel canister) have the potential to pose an exposure concern to personnel due to the possible presence of hot particles, fuel fleas, or residual crud in the empty fuel canister that can produce high dose rates. Fuel canisters and other waste generated in the FPA will be surveyed prior to transfer to the waste processing area. An administrative limit of 50 mrem/hr for waste being transferred to the waste processing will be used to minimize personnel exposure during waste processing activities. This limit along with the use of temporary shielding/shadow shielding will maintain worker exposure ALARA.

7.3.3 Ventilation

Section 3.3.2.2, *Ventilation and Off-Gas Systems,* and Section 4.3.1, *Ventilation and Off-Gas Systems,* provide details of the ventilation design. These sections discuss:

- areas and equipment serviced by each system
- design limits selected for operation and the performance limits
- major components and operating characteristics
- room controls and confinement zones
- testing criteria
- criteria for filter change-out

The ventilation is designed to prevent spread of radioactive material and control contamination between areas. Building air flows from areas of lower contamination potential to areas of higher contamination potential. Filters (both roughing and HEPA filters) will be used to limit the spread of radioactive materials in the ventilation system. Section 7.2.2 identifies the sources of radioactive material that may become airborne.

7.3.4 Area Radiation and Airborne Radioactivity Monitoring Instrumentation

Radiological monitoring and contamination control at the ISF Facility will be performed to ensure that radiation exposure and release limits in 10 CFR 20.1301 are not exceeded (Ref. 7-3). Monitoring involves the use, as appropriate, of fixed ARMs and CAMs. Fixed radiological instrumentation will be serviced by the facility's uninterruptible power supply system and ensures operability of instrumentation if building power is lost.

Criticality Monitors

Criticality monitors are solid-state area detectors that provide fast, accurate monitoring of total gamma exposure rates and instantaneous rise times of exposure rates. These instruments are robust and can monitor relatively high exposure rates for extended periods without damage to the instrument. Dedicated criticality monitors are in the FPA. The criticality monitoring system will have set points that will alarm on a gamma rate of rise or a neutron dose threshold. This will allow the criticality monitoring system to energize clearly audible alarms and visual signals if accidental criticality occurs. The guidance provided in ANSI/ANS8.3, *Criticality Alarm Systems,* will be used to determine coverage, placement, response time, failure warning, detection criteria, sensitivity, and testing requirements of the system. The CMS will not be seismically qualified.

Table 7.1-2 provides an estimate of the type and number of fixed monitors for the facility. Any reduction or addition in the overall number of fixed instruments will be evaluated to ensure that fixed instrumentation provides the monitoring necessary to maintain exposure ALARA. Placement of fixed instrumentation will be optimized by modeling and evaluation of the physical surroundings (e.g., airflow patterns, shielding features). Figure 7.3-3 through Figure 7.3-8 identify the location of the facility CAS, CAMs and ARMs.

Personnel Monitors

Personnel contamination monitors measure the exposure of personnel to radioactive contamination. Handheld monitors and half-body contamination monitors and/or walk-through detectors are used. In areas of frequent occupation, permanent detectors will be installed. In areas of infrequent occupation, portable detectors are used as required.

Area Monitors

Fixed ARMs will be located in key areas of the facility, in frequently occupied locations with potential for unexpected increases in dose rates, and in remote locations that require local indication of dose rates before personnel enter the area. Alarm setpoints will be established by evaluating the nominal area background dose rate. The setpoints will be set to a value greater than the nominal background dose rate that will trigger an alarm if exceeded. A typical setpoint could be twice the nominal background dose rate, or it may be a fixed area dose rate that, if exceeded, will trigger an alarm to notify personnel. The alarms will be visual and audible locally, with a corresponding signal included in the IDCS. ARMs can trigger local and facility interlock alarms and support configurable setpoints for personnel protection monitoring.

Air Monitoring

Air sampling and monitoring is required by 10 CFR 20.1703(a)(3)(i) to evaluate airborne hazards whenever respiratory protective equipment is used to limit intakes pursuant to 10 CFR 20.1702. Air sampling and monitoring will also be performed in situations where respiratory protective equipment is not required but the airborne radioactivity level can fluctuate and early detection of airborne radioactivity could prevent or minimize intake of radioactivity. Air sampling and monitoring will be conducted in accordance with the guidance in Regulatory Guide 8.25 (Ref. 7-10).

CAMs will be installed in occupied areas where facility personnel without respiratory protection could be exposed to airborne radioactivity concentrations exceeding 12 derived air concentration (DAC) hours in a week, or where there is a need to alert potentially exposed workers to unexpected increases in the airborne radioactivity levels. CAMs also will be used to detect breakthrough of the Fuel Packaging Area HEPA filters downstream of the FPA.

Each CAM will be configured with a setpoint appropriate to its primary function. For CAMs that monitor occupied work areas, the setpoint is a level of activity above the established background. Typical alert and alarm setpoints are 10 percent and 33 percent of DAC, respectively. A CAM alarm in a work area will instigate an evacuation of the immediate area per administrative procedures. Interlocks allow for emergency egress for life safety purposes. Response to the alarm is determined by administrative procedures.

For CAMs that monitor the discharge air downstream of the HEPA filter from the Fuel Transfer Area, a setpoint has been determined that indicates breakthrough of the FPA HEPA filters, with the response of instigating a maintenance cycle. Typical alert and alarm setpoints are 50 percent and 100 percent of the 10 CFR 20, Appendix B, Table 2 effluent concentration, respectively.

A beta-gamma CAM monitors the facility stack effluents. Typical alert and alarm setpoints are 50 percent and 100 percent of the 10 CFR 20, Appendix B, Table 2 effluent concentration, respectively, for representative radionuclides. Record sampling is also performed at the exhaust stack. If analysis indicates above-normal activity, administrative procedures will determine the appropriate response actions.

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7.4 ESTIMATED ONSITE COLLECTIVE DOSE ASSESSMENT

The ISF Facility is designed to minimize or eliminate radiation areas and high radiation areas that are normally occupied by facility personnel. Radiation areas may be present during cask receipt, fuel transfer, canister closure, canister storage, and waste processing activities. Because of the potential presence of radiation areas during these activities operational aspects are designed to ensure that exposure to facility personnel is ALARA. Dose rates will be kept ALARA by utilizing shielding and engineered controls. Exposures will be kept ALARA by performing most operations remotely, as follows:

- Transfer and fuel packaging operations are performed remotely.
- Personnel access to the Transfer Tunnel during fuel transfer is controlled by warning signs, beacons, and interlocks.
- Movement of fuel canister is performed inside a heavily shielded trolley system.
- Decontamination tools and services are strategically located throughout the facility to allow equipment to be decontaminated before maintenance activities.

In addition, many engineered design features minimize occupational personnel exposure during placement of SNF in dry storage, as well as offsite dose to the nearest neighbor during storage. The resulting dose at the ISF site boundary is well within the limits specified by 10 CFR 72.104 and 10 CFR 72.106 (Ref. 7-6).

As discussed in Section 7.3, the ISF Facility design incorporates shielding features that greatly reduce the expected dose rates resulting from fuel packaging and storage activities. The design objectives for the facility are to maintain doses to facility workers ALARA. The area design dose rates are based on expected occupancy and the administrative limits presented in Section 7.5.3.1. Each area design dose rate is established so that the area occupancy multiplied by the design dose rate does not exceed the annual administrative limits in Section 7.5.3.1. Table 7.4-1 provides the design dose rates for each area classification.

7.4.1 Operational Dose Assessment

The process flow for receipt, repackaging, and storage of a fuel canister received at the ISF Facility follows.

Fuel Receipt

Irradiated fuel will be transported to the Cask Receipt Area by truck and trailer using the Peach Bottom shipping cask. This cask has been used for INEEL onsite movement of Shippingport (LWBR) Type I modules from the Naval Reactor Facilities to the Idaho Chemical Processing Plant Facility. When the INEEL fuel was first moved, the fuel was out of the reactor approximately 3 years. The fuel will be more than 20 years old when shipped to the ISF Facility. The calculated highest anticipated dose rate for contact with the side of the Peach Bottom cask is 3.4 mrem/hr at 1 foot for the bounding TRIGA fuel.

In the Cask Receipt Area the Peach Bottom cask will be uprighted (it will be transported horizontally) and moved on to the cask trolley using an overhead crane. There is no shielding of this area other than the shipping cask. Time, distance, and shielding will be used to maintain facility personnel exposure ALARA.

Fuel Repackaging

The loaded cask trolley will enter a thick concrete-shielded Transfer Tunnel through a steel door at the west end of the south wall of the Storage Area. The Transfer Tunnel runs beneath the operating deck of the Storage Area and FPA. The Transfer Tunnel is the connector for movement of the cask trolley from fuel receipt to fuel repackaging, fuel canister closure, and to storage. The tunnel has 3 foot thick east wall (storage vault west wall) and west wall, and ceiling. The Transfer Tunnel east wall (FPA west wall) is 4 feet thick. The north wall is approximately 30 inches thick. The east wall of the Transfer Tunnel is the west shield wall of the Storage Area. The bottom of the tunnel is 3 to 5 feet below grade. In the FPA, the east Transfer Tunnel wall is the west wall of the ground floor electrical and HVAC rooms and the FPA. The north wall is common to the CCA. The west Transfer Tunnel wall is common to the SWPA, and the liquid waste storage tank area of the Transfer Area. A short section of the east and approximately the southern half of the west Transfer Tunnel walls are approachable from the facility exterior. This shielding is necessary to protect the public, adjacent offsite ISF Facility personnel, and ISF Facility personnel from high dose rates encountered when irradiated fuel is being removed from the Peach Bottom shipping cask at the FPA cask port. Personnel will be excluded from this northern area during spent fuel transfers. The southern part of the Transfer Tunnel will be used for preparing the transfer cask for either fuel removal at the FPA or for preparing the transfer cask for return to DOE. The shutter door isolates the northern section and is interlocked so that the door must be closed and personnel access prevented before fuel may be removed from the cask. The Transfer Tunnel also acts as a buffer to control and minimize the potential spread of radioactive contamination when performing decontamination operations on a transfer cask or maintenance or repair activities related to the CHM, cask trolley, and canister trolley.

The SNF will be transferred from the cask through the FPA cask port to the FPA by the FHM. The FPA is a large remote-handling area surrounded by 4 foot thick concrete walls, floor, and shielded windows. The floor, walls, and windows separate the FPA from the FPA operating galleries, maintenance shop, offices, change room, CCA, SWPA, and the lower level HVAC and electrical rooms. In addition to shielding, the walls provide a contamination control boundary during the processing and handling of irradiated fuel, potentially contaminated fuel baskets and other components, and during maintenance and repair of FPA equipment. The FPA is where spent fuel will be repackaged into ISF baskets and placed in the ISF storage canisters. The FPA has four floor transfer ports, two to the Transfer Tunnel and two to the SWPA. These ports are closed by stepped shield plugs.

Canister Closure

After the fuel has been repackaged in the FPA, the loaded fuel baskets and the canister internal shield plug will be sequentially loaded into a storage canister at the FPA canister port for transfer to the CCA via the Transfer Tunnel by the shielded canister trolley. The dose rate from the loaded canister is controlled to acceptable levels by the 11 inches of steel trolley shielding and the internal steel shield plug, which rests on top of the fuel basket closure plate.

After the canister trolley has been moved and located in its CCA port, the fuel basket funnel will be removed and the canister prepared for closure head installation and welding. Following welding, a nondestructive examination ultrasonic inspection of the welds performed and the canister connected to the vacuum drying/helium backfill system. During the vacuum drying, the trolley heaters will be energized to aid in the removal of any moisture from the fuel and the canister interior. The CCA has 3 foot thick concrete walls, floor, and ceiling. There are two floor ports in the CCA; one allows access to a vestibule area for importing new canister and components, and the other accesses the Transfer Tunnel and the canister trolley. Personnel radiation exposure in the CCA is a function of the canister loading. Anticipated dose rates from a sealed canister of TRIGA fuel are 6 to 10 mrem/hr at the top of the canister. The dose rates for Peach Bottom and Shippingport reflectors will be lower, as the active fuel region is farther from the closure head and has more axial self shielding due to fuel assembly construction. Significant radiation streaming could occur at the outer surface of the canister when the circularity collet is released. After canister closure completion, the canister trolley will be lowered and moved to the Storage Area/Transfer Tunnel access load/unload port.

Canister Storage

On the Storage Area operating deck, the Transfer Tunnel load/unload port plug will be removed and the CHM will be positioned to retrieve a loaded canister from the canister trolley. The CHM grapples the canister, raises it to the desired traveling elevation, rotates to close the canister chamber, raises the shield skirt, and travels to the storage vault tube designated for the canister. The CHM will be positioned over the storage vault tube location and the shield skirt will be lowered, the storage tube shield plug removed, and the canister will be lowered into the storage tube. The CHM will then install the shield plug. After the shield plug has been installed and the grapple raised, the skirt will be raised and the CHM moved to a standby location so that personnel can complete the storage tube closure. The CHM operating platform is above the top of the active fuel region of a loaded canister to reduce personnel exposure.

Storage tube closure will be accomplished by bolting a tube closure plate to the top of the tube, leak checking the closure plate seal, and connecting the tube to the evacuation/backfill system via a valve and connection on the closure plate. Upon completion of the helium backfill, the backfill system will be disconnected and a seal plate placed over the connection location and its seals leak checked. The final operation relating to canister storage is to install a steel tornado-missile protection cover over the top of the tube assembly.

The low-level wastes generated as a result of facility operations will be initially accumulated in the SWPA for processing and packaging. The SWPA is bounded by concrete walls with two shielding wing walls dividing it into north and south areas. The southern area is isolated by a contamination control barrier as it is designed to receive wastes from the FPA via two circular ceiling ports and support maintenance of the FHM via rectangular ceiling opening. This portion of the SWPA contains size reduction equipment for non-compactable wastes. The northern section has two personnel access doors (west and north walls) and a north wall rollup door for large item shipping and receiving. Compactable wastes will be segregated from non-compactable wastes and processed accordingly. The waste compactor facility is located in the southeast corner of this northern section and is connectable to the southern portion via a passthrough in the wing wall so that compactible wastes from the FPA and/or the CHM may be processed. Section 6.4, *Solid Waste*, describes waste transfer and survey in the FPA to the Solid Waste Processing System.

Table 7.4-1 identifies potential radiation areas during fuel movement, storage, and waste processing activities and estimated occupancy times. The time estimates are based on receiving an average of two cask shipments from the DOE each week.

Table 7.4-2 provides the estimated occupational dose from these operations. The general formula described below was used for dose estimate calculations.

Person-rem dose for task = (#workers)(frequency)(duration)(dose rate)

where,

#workers – the number of workers performing the task

frequency – how often the task will be performed (e.g., daily, weekly, etc.)

duration – the amount of time taken to accomplish the task

dose rate – the dose rate at the worker location. (In some cases an occupancy factor correction has been made, so that even though the worker is required for the operation, the worker is not considered to be present full time. This factor is expressed as a reduced dose rate.)

The Cask Receipt Area, FPA Operating Gallery, and Storage Area operations dose rates are based on values calculated for having a Peach Bottom cask loaded with TRIGA fuel or TRIGA fuel being processed or handled in the other areas. The cask return values are based on returning a contaminated Peach Bottom cask liner and basket assembly in the Peach Bottom cask. In this case, the dose rate from the cask is negligible when compared to the shield leakage from TRIGA fuel stored in the storage vault on the other side of the east tunnel wall. The CCA operations values are based on closing canisters of TRIGA fuel, as this fuel has the fuel meat much closer to the top of the canister than either Peach Bottom or Shippingport loaded canisters. SWPA size reduction operations are evaluated using the 50mrem/hr maximum allowable contact dose rate for waste items being exported from the FPA. The shield walls of the Operating Gallery and the Transfer Tunnel are assumed to be 4 and 3 feet thick, respectively.

The following assumptions were used in the preparation of the dose estimates associated with the operation of the ISF.

- All tasks assume TRIGA fuel is being handled or stored (bounding photon flux per canister).
- The dose estimates are conservatively based on processing the maximum (bounding) dose rate material for one year of operation under equilibrium conditions.
- The maximum fuel inventory in the FPA.
- Loaded storage vault tubes adjacent to the tunnel wall.
- Loaded storage vault tubes adjacent to an empty tube.
- SWPA work with a partially loaded waste box present.
- Dose rates in the SWPA are assumed to be principally from the material being processed and its accumulation in filled or partially filled waste containers.
- Source (S) sizes are assumed to be large enough that a 1/R adjustment for distances out to S/2 is appropriate and $1/R^2$ after that if not provided by the calculation.
- The frequency of operation is based on the assumption of receipt of five DOE fuel shipments per month (1.25 shipments per week).

- It is assumed that the shielding efficiency of the FPA viewing windows is the same as that of FPA walls. This is based on the assumption that FPA operating gallery personnel will be spending at least 90% of their work time in front of or in close proximity to a window.

7.4.2 Site Dose Assessment

The ISF Facility fence serves as the restricted area boundary in accordance with 10 CFR 20.1502 (Ref. 7-3) for personnel monitoring of external and internal occupational dose. The INEEL site boundary serves as the controlled area boundary in accordance with 10 CFR 20.1301 (Ref. 7-3), and 10 CFR 72.104 and 10 CFR 72.106 (Ref. 7-6).

Because the fuel handling and storage operations are 8 miles from the INEEL-controlled area boundary, the dose at the boundary will be well below the 10 CFR 20.13012 requirement of 100 mrem/yr to a member of the public, and below the 10 CFR 72.104 limit of 25 mrem/yr to any real individual. The maximum dose to the maximally exposed individual is estimated to be 0.00003 mrem/yr from ISF facility operations at the controlled area boundary. The contribution from nearby facilities is estimated to be less than 0.32 mrem/yr. The total estimated dose at the controlled area boundary (ISF Facility dose and the dose contributed from nearby facilities) is less than 0.32 mrem/yr and is well below the 10 CFR 20.1302 and 10 CFR 72.104 limits. The greatest source of radiation to the controlled area boundary is from air emissions during ISF Facility operations. Direct and scattered radiation from the ISF Facility does not contribute a measurable dose at the controlled area boundary. Shielding provided by natural and manmade earth barriers between the ISF Facility and the controlled area boundary are not used for shielding to reduce the dose at the controlled area boundary.

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7.5 HEALTH PHYSICS PROGRAM

The ISF Facility Health Physics Program is established to control and safely minimize the exposure of personnel to ionizing radiation. Corporate and facility personnel will be committed to maintaining personnel exposures within regulatory limits and ALARA.

7.5.1 Organization

This section describes the administrative organization of the ISF Facility Health Physics Program including the authority and responsibility of each position identified. The administrative organization consists of the Environmental Health and Safety (ES&H) Manager, the Radiation Safety Officer (RSO), administrative and technical support, and radiation protection technicians. The ISF Facility ES&H Manager will be responsible for developing and implementing the ISF Facility Health Physics Program. The RSO, administrative and technical support personnel, and radiation protection technicians will assist the ES&H Manager with the implementation and oversight of the program. The ES&H Manager has authority to enforce safe plant operations and cessation of work to prevent unsafe practices. This authority is passed down through the organization.

The ISF ES&H Manager reports directly to the FWENC Director of Environmental Health, Safety, and Quality (ESQ). However, he takes day-to-day ISF project direction from the ISF Project Manager. This direct line to the Director of ES&Q ensures an appropriate level of independence in safety functions from ISF line management, including sufficient independence from cost and schedule issues. The ISF ES&H Manager will be responsible for environmental, health and safety, emergency planning, security, and administers radiation safety at the ISF Facility. Chapter 9 discusses the responsibilities of the ISF ES&H Manager.

The RSO reports to the ISF ES&H Manager and will be responsible for the implementation of the Radiological Protection Program as directed by the ISF ES&H Manager. Chapter 9 discusses the responsibility of the RSO.

Radiation protection technicians advise the RSO and the Shift Supervisor of any abnormal radiological condition that could result in an unusual hazard. In the absence of the RSO, radiation protection technicians assume responsibility for radiation monitoring and control functions during emergency conditions.

Radiation protection personnel will be selected, trained, and qualified to ensure that they have sufficient knowledge and practical abilities to implement the radiation protection program effectively. Qualification criteria and job descriptions have been developed for technical positions within the radiation protection organization.

Radiation protection technicians will be required to participate in classroom and specific on-the-job training. The radiation protection program and implementing procedures ensure that radiation protection personnel, who will be selected, trained, and qualified, have the knowledge and practical skills necessary to perform their work

7.5.1.1 Selection

The ES&H Manager ensures that personnel have sufficient education and/or experience in the job functions to which they are assigned. Radiation protection technicians will be required to meet the education and experience levels specified in ANSI/ANS 3.1 (Ref. 7-11).

7.5.1.2 Classroom Training

Classroom training includes:

- training in emergency response duties
- training in radiation protection procedures, the operation and limitations of survey and count room equipment, and methods to ensure proper record documentation and traceability
- reviewing major work activities and potential radiological hazards that may be encountered
- reviewing revisions to 10 CFR 20 and their impact on radiation protection activities (Ref. 7-3)
- testing radiation protection technicians to verify appropriate knowledge level in radiation protection theory, equipment, basic mathematics, and recognizing unusual situations involving radioactivity

Annual refresher training for radiation protection technician, using a structured program approved by the ES&H Manager, will be conducted. This training will be documented and may include a written examination.

7.5.1.3 On-the-Job Training

Upon completion of required classroom training, radiation protection personnel complete on-the-job training in assigned duties. The responsible supervisor documents successful completion of these duties on the radiation protection personnel's qualification card. The responsible supervisor ensures that training has been adequate by observation of on-the-job performance.

7.5.1.4 Other Training

Additional training will be provided to radiation protection personnel if significant changes occur in radiation protection policy, requirements, techniques, procedures, or equipment, and as required by other programs. This information will be disseminated to affected personnel or organizations through periodic awareness presentations and/or required reading. Section 9.3, *Training Programs*, discusses the ISF Facility Training Program.

7.5.2 Equipment, Instrumentation, and Facilities

The ISF Facility health physics organization implements the operational radiological surveillance program. This program ensures the health and safety of facility personnel, the public, and protection of the environment. The health physics organization utilizes a low background counting room and shift change room to perform counting operations and store health physics equipment and supplies. The low background counting room has alpha/beta scalers, a HPGe assay system, and a liquid scintillation counter. Additional portable health physics instrumentation is also stored in the counting room.

Protective clothing used by facility personnel is located in the facility change rooms. Respiratory protective equipment is also located in the facility change rooms.

The ISF Facility has a personnel decontamination shower and several eye wash stations located throughout the facility. Additional equipment for personnel decontamination and other contamination control equipment, including spill control materials, will be available.

The ISF Facility does not have any fixed facilities for internal radiation monitoring, such as whole-body counters, thyroid counters, or bioassay sample analysis equipment. If these types of services are required ISF Facility personnel will utilize the same services as those used by INEEL personnel.

7.5.2.1 Requirements for Instruments

Instruments for measuring radiation will be used to:

- monitor radiation exposure levels
- monitor contamination levels and concentrations of airborne radioactivity to characterize workplace conditions
- verify the effectiveness of physical design features and engineering, and administrative controls
- identify areas requiring postings

Radiation detection instrumentation has a sensitivity suitable for the required measurement (e.g., instruments used to measure leak tests of a sealed source are able to detect concentrations less than 0.005 microcuries of the kind of radioactive material in the source).

Instruments used to measure radiation dose rates, levels of contamination, and concentrations of airborne radioactivity in the field are given in Table 7.1-1. Equivalent instrumentation may be substituted.

Counting Instrumentation

The ISF Facility health physics program uses the following counting room instruments to support facility personnel protection and to ensure that both exposures and releases will be ALARA:

- HPGe analyzer for material protection measurements
- low-background alpha-beta radiation proportional or scintillation counter
- end-window GM counters
- a liquid scintillation counter for low energy beta (e.g., tritium) analysis

Instruments used in the low-background are provided in Table 7.1-1.

Instrument Calibration

Portable equipment used to perform radiological surveillance will be calibrated at 6month intervals or following repair, whichever occurs sooner. Stationary equipment used to perform radiological surveillance will be calibrated annually or following repair, whichever occurs sooner. Other measuring equipment will be maintained following manufacturer's recommendations.

Airflow or volume-metering devices used with air samplers will be calibrated annually. Lapel air samplers will be calibrated before use or after media change.

Instrument Checks

Instruments used to perform radiation surveys will be readily available and response-checked daily or before operation by radiation protection technicians. When response checks are not feasible, such as with instruments used to measure neutrons or tritium, compensatory actions have been established to ensure proper instrument performance. Instruments that do not respond properly will be taken out of service until they have been repaired and recalibrated. Battery checks will be performed each time an instrument is used.

Daily checks of fixed instrumentation will be performed and ChiSquare evaluations will be performed as applicable for the type of instrument. Fixed monitoring instrumentation will be included on the facility maintenance schedule.

Instrument Selection

The ES&H Manager determines the type of instruments used at the facility. The ES&H Manager ensures there is a sufficient number of the proper type instruments available in the event of instrument failure or during periods of instrument calibration. The ES&H Manager maintains and controls the instruments in use at the facility.

Instrument Storage, Calibration, and Maintenance Facilities

Health physics instruments will be stored in the health physics' office when they are not in service. Portable instruments will be returned to the health physics office at the end of each shift. Minor instrument maintenance (e.g., replacing batteries) is performed by the health physics organization. Instruments requiring major maintenance or calibration will be removed from service and shipped to the instrument vendor for service and calibration.

7.5.2.2 Radiological Surveys

Radiological surveys will be conducted for verification and documentation of radiation and contamination levels to ensure personnel exposure is ALARA. Surveys document radioactive contaminant concentrations and dose rates within controlled process areas and will be used to ensure that appropriate protective measures are included in task planning. Information gained in such surveys is the basis for preparing radiation work permits.

Radiological monitoring of radiation exposure levels, levels of contamination, and concentrations of airborne radioactivity will be conducted to characterize workplace conditions, to verify the effectiveness of physical design features and engineering and administrative controls, identify required PPE, and to identify areas requiring postings.

Only trained and qualified personnel, using instruments that are properly calibrated and routinely tested for operability, perform monitoring. Surveys for radiation, contamination, and airborne radioactive materials will be performed as specified in implementing procedures and radiological work permits.

A central monitoring system that gives information on the dose rate and concentration of airborne radioactive material is used. The following features were considered in the design, selection, and installation of the central monitoring system:

- readout capability at the main radiation protection access control point
- placement of detectors for optimum coverage of areas
- circuitry that indicates component failure
- remote and local alarms and readouts
- ranges adequate to ensure readout of the highest anticipated radiation levels and to ensure positive readout at the lowest anticipated levels
- capability to record the readout of systems

Survey Frequency

The frequency of routine surveys depends on the nature, quantity, and use of radioactive material, as well as the specific protective facilities, equipment, and procedures that are designed to protect personnel from external and internal exposure.

Job-specific monitoring will be conducted whenever operation or maintenance includes accessing unknown contamination or radiation conditions or known high contamination or radiation conditions. Airborne activity, surface contamination, including alpha contamination and area surveys including beta radiation, may be required whenever the radiological conditions are unknown in a work area. In cases where the need for a survey is not clear, the ES&H Manager makes the determination. Non-radiological areas will be surveyed periodically to ensure that radiation and radioactive material is adequately controlled.

Table 7.5-1 and Table 7.5-2 identify the minimum survey requirements of the ISF Facility. Additional surveys will be performed as necessary to properly assess radiological conditions.

Survey Responsibilities

The ES&H Manager maintains routine survey status and provides daily, weekly, and monthly schedules for the performance of routine surveys. The RSO ensures surveys will be performed as scheduled. At the end of shift, the schedule will be checked for completeness and the status system updated.

Any scheduled surveys not performed by the designated shift will be performed by the next shift, or reasons for not performing the survey will be documented. Routine surveys will not be performed in high radiation areas or high contamination areas except as directed by the RSO.

Radiological survey follow-up includes evaluation of the data to determine the radiological controls necessary for work areas.

Survey Documentation

Job-specific surveys and results will be documented at the time they are performed. Other survey data will be documented by the end of each workday. Survey data includes the area surveyed, the measured results including instrument background, and the location of the readings. These surveys also indicate instrument serial numbers, calibration due dates, and the name and signature of the person conducting the survey. Survey documentation will be accurately and legibly completed. Blanks will be filled in or marked not applicable (N/A). Negative data will be recorded.

An alteration or change to survey records, either existing or being generated, will be made neatly by drawing a single line through the incorrect entry and recording the correction/alteration adjacent to the incorrect entry. Correction fluids or other correction media/techniques that obliterate the original entry will not be used. The original entry must remain legible. The person making the change initials and notes the date of the correction. Only the person making the data entry can change or alter the survey data. The ES&H Manager may make administrative information corrections (i.e., corrections that do not involve the survey results). The ES&H Manager maintains the survey records and reviews them as deemed necessary.

7.5.3 Procedures

Procedures for ISF Facility operations that expose facility personnel to radiation will be reviewed and approved by the ES&H Manager before use to ensure adherence to the facility radiological control requirements. These reviews include the following objectives:

- avoid unnecessary exposure to radiation
- maintain doses to individuals ALARA
- maintain the collective facility dose ALARA

Radiation protection requirements for radiological work will be implemented through the use of procedures including, but not limited to, the following:

- performing badging functions for access authorization
- issuing personnel dosimetry, monitoring, recording, and tracking individual exposures
- performing radiological safety training and refresher training
- performing ALARA reviews of plant procedures and monitoring of operations
- issuing, revising, and terminating radiation work permits
- roping off, barricading, and posting radiologically controlled areas or zones
- decontaminating personnel, equipment, and areas
- performing radiation surveys
- smear/swab sampling and sample counting
- quantifying airborne radioactivity
- maintaining records of the radiation protection program, including audits and other reviews of program content and implementation; radiation surveys; instrument calibrations; individual monitoring results; and records required for decommissioning

7.5.3.1 Occupational Dose Limits

The ISF Facility Manager and ES&H Manager ensure radiation exposures are maintained ALARA and within regulatory and administrative limits. Levels shall not exceed the occupational dose limits set for individual adults in 10 CFR 20. The ES&H Manager obtains and maintains records of prior exposure history of personnel who may be exposed to radiation at the facility.

Dose Limits and Administrative Control Levels

Applicable dose limits, administrative control levels, and responsibilities for authorization to exceed the administrative control levels are listed below (Ref. 7-3). The persons identified in the table below authorize exposure extensions in writing. Extended exposures do not exceed the limits in 10 CFR 20.

	Total Effective Dose Equivalent	Dose to Lens of Eye	Shallow Dose	Extremity Dose	Permission to Exceed (Documented)
Regulatory Limit/Year	5000 mrem	15,000 mrem	50,000 mrem	50,000 mrem	Planned Special Exposure (10 CFR 20.1206)
Administrative Control Limits					
Day	50 mrem				Shift Supervisor
Week	200 mrem				Operations Manager
Quarter	500 mrem				ES&H Manager/RSO
Year	1000 mrem	3000 mrem	10,000 mrem	10,000 mrem	ES&H Manager/RSO and ISF Facility Manager

The ES&H Manager will investigate internal exposure monitoring results when radionuclides in the worker's body could deliver more than 5 percent of the annual limit on intake (ALI) as specified in 10 CFR 20, Appendix A. A report of this investigation shall be maintained in the worker's dose records.

Declared Pregnant Woman

A declared pregnant woman is any woman who has voluntarily informed her employer of her pregnancy and estimated date of conception in writing. The ISF Facility administrative dose limits for declared pregnant women are the same as those listed in 10 CFR 20.

- The occupational exposure of the declared pregnant woman shall not exceed 500 mrem for the entire gestation period.
- The exposure rate to the declared pregnant woman will be maintained at a level such that the dose to the embryo/fetus does not exceed 50 mrem/month.

Should occupational exposure to the declared pregnant woman have exceeded 450 mrem before declaration, she shall be immediately removed from further exposure. A declared pregnant woman is not eligible for a planned special exposure.

Upon declaration of pregnancy, or intent to become pregnant, steps may be taken to further reduce the woman's exposure to radioactive materials. Some of these actions may include:

- limiting work assignments in radiation or contamination areas
- forbidding work assignments in high radiation, high contamination, or airborne radioactivity areas

Dose Limits for Individual Members of the Public and Minors

Exposure of any individual outside a restricted area or any minor is limited so that the total dose from ISF Facility operations does not exceed the 10 CFR 20.1301 dose limit of 100 mrem/year for individual members of the public.

7.5.3.2 Internal Personnel Radiation Monitoring (Bioassay)

Internal radiation monitoring (bioassay) will be performed on facility employees to verify the effectiveness of the radiation protection program. Additionally, internal radiation monitoring will be used to assess any potential uptakes by facility employees and to monitor the elimination of contaminants from an affected individual. Monitoring of the intake of radioactive material by a facility employee is required by 10 CFR 20.1502(b) if the employee is likely to receive an intake that exceeds 10 percent of the ALI in one year. The internal monitoring program is based on Regulatory Guide 8.9, Regulatory Guide 8.11, and Regulatory Guide 8.26 (Refs.7-12, 7-13, and 7-14).

Initial Employment

An initial baseline is established for personnel whose job involves the potential for significant occupational exposure to airborne radioactivity and for personnel authorized to use respiratory protection equipment. A whole body count will be performed, usually within the first month of employment. A urinalysis will be conducted before work on a job involving potential exposure to airborne radioactivity. The analysis is for ^{3}H, gamma emitting radionuclides, gross alpha, and gross beta (minus ^{40}K).

Periodic Monitoring

Personnel assigned to this facility on a long-term basis have a routine whole body count and/or urinalysis on an annual (once/year) basis in accordance with Regulatory Guide 8.9 (Ref. 7-12). Additionally, personnel will be selected each quarter for whole body counts and/or urinalysis; selection is either random or based on the potential for uptake. The number selected will be increased if the number of radiation personnel on site increase.

Method sensitivities will be based on International Commission on Radiological Protection (ICRP) 54 and will generally be based on 10 percent of the ALI (Ref. 7-15). However, for some alpha-emitting radionuclides, current bioassay methods are not sensitive enough to confirm intakes of 10 percent of the ALI. Therefore, the monitoring program may use personal air samplers to determine intake.

Special Monitoring

Internal radiation monitoring will be performed at a minimum for the following circumstances (Reg. Guide 8.9 [Ref. 7-12]):

- an individual's exposure exceeds 40 DAC hours since the most recent bioassay measurement
- an individual, wearing a respiratory protection device, is exposed to an environment that would create an intake exceeding 520 DAC hours if no protection was used
- an individual is exposed to a radiological airborne contaminant and the average concentration or duration of exposure is not known
- skin contamination in the facial/nasal area
- following the contamination of an open wound or damaged skin, as needed
- evidence indicates measurable unknown ingestion or absorption of radioactive material
- evidence of damage to or failure of respiratory protective devices

Internal radiation monitoring will be evaluated as needed for other circumstances as they arise.

Termination Monitoring

When employees who have been occupationally exposed while working at the facility leave employment they will be given a termination whole body count. If a terminated employee refuses to have the count conducted, a statement to that effect will be prepared by the ES&H Manager and placed in the employee's dosimetry file.

Records

The ES&H Manager will be responsible for maintaining records for personnel dosimetry and internal monitoring. Records containing information specified in 10 CFR 20.2106 will be maintained for occupationally exposed personnel and will be available for inspection.

7.5.3.3 External Radiation Monitoring Dosimetry

Whole Body Monitoring

Each employee who is occupationally exposed to radiation will be provided with, and required to wear, either a whole body TLD and a self-reading dosimeter. These dosimeters will be worn together on the front of the body, unless otherwise specified by radiation protection.

Visitors and personnel not assigned to the facility will be assigned a self-reading dosimeter, or equivalent monitoring, before entering any radiologically controlled area.

Extremity Monitoring

Extremity dosimetry will be issued and worn in accordance with ISF Facility health physics program requirements.

Quality Control of Dosimetry

TLDs will be provided and processed by INEEL or by an organization currently accredited by the National Voluntary Laboratory Accreditation Program (NVLAP) for Personnel Dosimetry Processors of the National Institute of Science and Technology. Personnel and extremity dosimeters will be processed quarterly unless conditions dictate more frequent processing. Self-reading dosimeters will be checked for accuracy and drift at least every six calendar months.

7.5.3.4 Signs and Labels

The ES&H Manager will be responsible for implementation and compliance with the radiological controls posting requirements of 10 CFR 20 (Ref. 7-3).

7.5.3.5 Instructions and Notices to Personnel

The ISF Facility Manager will be responsible for complying with posting and notification requirements to personnel as required by 10 CFR 19 (Ref. 7-16).

7.5.3.6 Notifications and Reports

Radiation exposure data for an employee and the results of any measurements, analyses, and calculations of internally deposited radioactive material must be reported to the employee. Data and results required by applicable regulations or license conditions will be included in notification reports.

7.5.3.7 Notifications to Workers

Notification reports issued by the ES&H Manager must be in writing as described in 10 CFR 19. Notification reports will be issued annually and include:

- company name
- individual's name
- individual's social security number
- individual's lifetime exposure totals calculated for his/her Occupational Radiation Exposure History Record
- individual's facility radiation exposure information contained in the updated Occupational Radiation Exposure History Record

The following statement will be included in the Notification Report: "This report is furnished to you under the provisions of the Nuclear Regulatory Commission's rules and regulations for radiation protection. You should preserve this report for further reference."

7.5.3.8 Reports to NRC

When the ISF Facility is required to report any employee's exposure to the NRC in accordance with 10 CFR 20, a written report of the exposure data will also be issued to the affected individual. Such reports shall be transmitted no later than the transmittal to the NRC.

7.5.3.9 Terminating Employees

At the request of an employee terminating employment that received occupational exposure at the ISF Facility, the ES&H Manager shall:

- provide a written report at termination to the individual or the individual's designee
- report the radiation dose received during the current year or fraction thereof
- provide a written estimate if the most recent individual monitoring results are not available at the time of request

7.5.3.10 Reports to Former Employees

An employee formerly engaged in activities controlled by the ISF Facility can request a report of the employee's exposure for each year they were required to be monitored. The report shall be furnished within 30 days from the time the request is made or within 30 days after the exposure of the employee, whichever is later. The report covers the period of time the employee's activities involved radiation exposure at the ISF Facility and includes the dates and locations of activities the worker participated in during the period.

7.5.3.11 Instrument Check Sources

Security and Posting

Calibration and daily check sources that are not exempted from labeling requirements of 10 CFR 20.1905 and licensing requirements of 10 CFR 31.5 will be controlled in accordance with the source labeling, control, and leak testing requirements in 10 CFR 31.5 (Refs. 7-3 and 7-17). When these sources are not in use they will be kept in a source storage locker that is locked when not attended. This locker will be under the control of the ES&H Manager and only authorized personnel will be allowed access. The source storage locker will be posted in accordance with 10 CFR 20.

Radiation Levels

Contact radiation levels at accessible exterior surfaces of the source storage lockers will be less than 2 mrem/hour. A radiation survey of the locker will be performed routinely and immediately after receipt or disposal of nonexempt sources.

Inventory

A physical inventory of nonexempt sources will be conducted within 12 months from the last inventory. This inventory will be updated as sources are transferred, received, or disposed of.

Transfer or Disposal

When a source is no longer needed, it will be returned to the manufacturer or disposed of as radioactive waste. When transfer will be to an employee or organization outside of the ISF Facility, the transfer can be made only to an employee licensed to possess the source. The ES&H Manager maintains written verification of the employee's legal authorization to possess the source. Transfers will be documented and a copy maintained.

7.5.3.12 External Contamination

Personnel must survey themselves, or be surveyed by a person qualified to perform personnel surveys, before leaving a radiologically controlled area. If contamination exceeding established limits is detected the employee will remain in place and immediately contact (or have someone contact) the health physics department. The ES&H Manager implements decontamination measures as needed.

Areas and Equipment

Surface contamination will be controlled to minimize inhalation or ingestion of radioactive material to prevent the spread of contamination and to minimize buildup of radioactive material in the work environment. Radioactive material will be controlled and the number and extent of contamination areas minimized to maintain personnel exposure ALARA. Work areas with surface contamination levels greater than the limits listed below will be controlled as Contamination Areas or decontaminated using standard techniques. Surface contamination limits for unrestricted release of materials and equipment are listed below unless a more restrictive limit is directed by the ES&H Manager.

Contaminant	Removable	Total (fixed plus removable)
Alpha	20 dpm/100 cm^2	100 dpm/100 cm^2
Beta/Gamma	1000 dpm/100 cm^2	5000 dpm/100 cm^2

Work areas with airborne radioactive material in concentrations greater than 0.1 DAC or such that a worker would be exposed to greater than 12 DAC hours in a week will be controlled as Airborne Radioactivity Areas.

Prevention

Plans, training, and work instructions emphasize the need to minimize radioactive contamination of personnel and areas not controlled for radioactive surface contamination. Occurrences of skin contamination will be documented and reviewed by the ES&H Manager.

7.5.3.13 Respiratory Protection

The ISF Facility is designed to utilize process or engineering controls to minimize to the extent practical the concentration of radioactive material in the air. Process or engineering controls that may be evaluated include the use of local filtered ventilation systems, decontamination of equipment before performing maintenance, control of access, limitation of exposure time, and use of other types of exposure controls. Respiratory protection equipment will only be used after these or other similar measures to limit worker intakes of radioactive materials have been considered.

When respiratory protective equipment is used, air sampling will be performed sufficient to identify the potential hazard, permit proper equipment selection, and estimate dose. The air sampling equipment used may include CAMs, portable work place air samplers, and lapel breathing-air zone samplers. The air sampling recommendations in NRC Regulatory Guide 8.25, *Air Sampling in the Workplace*, will be used to establish the ISF Facility workplace air sampling program requirements (Ref. 7-10). Recommendations for air sampling from NUREG1400 will be used to determine location of air samplers and sampling methods (Ref. 7-18).

Policy

It is ISF Facility policy to minimize the inhalation of air contaminated with dusts, fumes, mists, gases, vapors, and radionuclides. The primary means to implement this policy is prevention or mitigation of contamination at the source. This is achieved using engineering controls such as containment, ventilation and process modification.

The ISF Facility health physics program implements a comprehensive respiratory protection program to minimize exposure and comply with requirements and industry standards. The respiratory protection program is based on Regulatory Guide 8.14 (Ref. 7-19). Quantitative fit tests will be performed for negative-pressure respirators and consideration of exposure versus increased work time due to respiratory equipment is considered to ensure that personnel exposure is ALARA. Only respiratory protection equipment that is tested and certified, or had certification extended by the National Institute for Occupational Safety and Health will be used.

Respirator Use

The use of respirators is less desirable than the use of engineering controls. A respirator subjects the wearer to additional stress and increases the risk of injury by interfering with vision, limiting motion, and impairing communication. Routine and special tasks will be planned such that potential sources of airborne contaminants are managed by engineering controls. Respirators may be prescribed for radiation protection while engineering controls are being instituted or evaluated or where effective controls are impractical.

It is impractical to set specific time limits for respirator use under all conditions. Breaks and limitations on the total time of continuous use of respirators will be encouraged. Personnel using respirators will be required to leave the area of required usage and remove their respirators in instances of equipment malfunction, undue physical or psychological stress, procedural or communication failure, significant deterioration of operational conditions, or any other condition that might require such relief.

Program Administration

The respirator program will be administered as follows:

ES&H Manager Responsibilities

- Administer overall respiratory protection program development, administration, technical direction, and the evaluation of program effectiveness.
- Authorize provision for cessation of work in cases where respiratory program noncompliance exists.
- Ensure program compliance with respiratory policy and applicable requirements.
- Evaluate oxygen deficient environments or environments containing industrial contaminants and prescribe any necessary and approved respiratory protection devices.
- Ensure the program at each location is appropriate and promptly identify adverse trends.

- Ensure that personnel who may be required to use respiratory protective devices satisfactorily complete a physiological examination and a fit test before working in a respirator.
- Ensure that an employee has annual approval by a physician indicating that the employee is physically and mentally able to wear a respirator in accordance with established facility standards.
- Ensure that an actual evaluation of the sealing efficiency is performed as part of the fit test. This evaluation provides a quantitative fit test for each make and/or type of facepiece. Fit tests will not be required for hoods, helmets, or suits.

ISF Facility Manager Responsibilities

- Implement engineering controls to minimize the need for respiratory protective equipment.
- Implement procedures designed to minimize the inhalation of airborne radioactive materials to levels within regulatory limits and ALARA.
- Control exposure to airborne radiological materials and interface with the ES&H Manager for technical guidance relative to respiratory protection devices for the control of industrial airborne contaminants.
- Plan and conduct work with the potential for inhalation of contaminants or in the presence of an oxygen deficient atmosphere to prevent or mitigate the adverse environment.
- Obtain ES&H Manager review of work plans.
- Ensure that personnel supervising tasks requiring the use of respiratory protection devices comply with the respiratory protection program.

Personnel Responsibilities

- Use prescribed respiratory devices provided in accordance with the instructions and training received.
- Guard against damage to respiratory protection equipment.
- Report any malfunction of respiratory protection equipment to health physics.
- Ensure personnel are clean-shaven, because facial hair can cause facepiece leakage. For purposes of the respiratory protection program, clean-shaven is defined as no facial hair between the sealing surface of the facepiece and the face.

7.5.3.14 Records

Any alteration or change to health physics records, either existing or being generated, will be made neatly by drawing a single line through the incorrect entry and recording the correction/alteration adjacent to the incorrect entry. Correction fluids or other correction media/techniques that obliterate the original entry will not be used. The original entry must remain legible. The person making the change shall initial and note the date of the correction. Only the person making the data entry can change or alter the survey data. The ES&H Manager may make administrative information corrections (e.g., spelling). Records will be maintained as required by 10 CFR 20 Subpart L, as discussed in Chapter 9.

7.6 ESTIMATED OFFSITE COLLECTIVE DOSE ASSESSMENT

The purpose of the environmental monitoring program is to demonstrate compliance with the dose limits for offsite personnel. The dose limits are given in 10 CFR 72.104 as 25 mrem/year to the whole body, 75 mrem/year to the thyroid, and 25 mrem/year to any other critical organ (Ref. 7-6). The limits refer to exposure from planned discharges, direct radiation, and radiation from other fuel cycle operations in the region. The Idaho Department of Environmental Quality regulates the INEEL site in accordance with 40 CFR 61 Subpart H, *National Emission Standards for Emissions of Radionuclides Other than Radon from Department of Energy Facilities*, which requires the ISF Facility to comply with the dose limit of 10 mrem/yr to any member of the public from facility air emissions (Ref. 7-20).

The priority parameters monitored as part of the environmental program are the identity and concentration of particulate radionuclides in the building ventilation air (i.e., exhausted through the stack). Other parameters are the identity and concentration of radionuclides in ambient onsite air, direct exposure in the outdoor portions of the site, and the identity and concentrations of radionuclides in site soil.

Environmental Measurements

Doses from stack exhaust will be determined by measuring the amounts of radionuclides emitted and meteorological modeling. The effluent concentrations will be determined using an isokinetic sampler in the facility stack and analyzing the activity of the various radionuclides collected. Modeling is based on information from the National Oceanic and Atmospheric Administration (NOAA) Air Resources Laboratory. The calculated offsite concentrations will be converted to doses based on dose conversion factors developed in accordance with ICRP 30, *Limits for Intakes of Radionuclides by Workers*, and standard inhalation rates (Ref. 7-15). Radiation dose rates at the perimeter of the site will be measured directly using environmental TLDs.

Doses from ground level releases, if any, will be determined directly by measuring the concentrations of radionuclides in the air at the site boundary. The calculated concentrations at the site boundary will be converted to doses based on dose conversion factors developed in accordance with ICRP 30 and standard inhalation rates.

The analysis of soil samples may be necessary to assess the effects of a spill or the efficacy of cleanup activities. Soil samples also will be taken as part of the eventual decontamination and dismantling of the facility.

7.6.1 Effluent and Environmental Monitoring Program

INEEL Monitoring Program

The ISF Facility specific environmental monitoring program, discussed below, will continue through the life of the facility. The results of the ISF Facility radiological environmental monitoring program will be reported to meet the 60day reporting requirement of 10 CFR 72.44 and the annual reporting requirement of 10 CFR 61 Subpart H (Refs. 7-6 and 7-20).

7.6.1.1 Gaseous Effluent Monitoring

An evaluation of the expected source term from the SNF to be received and processed at the ISF Facility indicates the presence of particulate and gaseous radionuclides. The primary particulate radionuclides are $^{137}Cs/^{137m}Ba$ and $^{90}Sr/^{90}Y$, and the primary gaseous radionuclides of concern are ^{129}I, and ^{3}H. Fuel packaging operations conducted in the FPA are the predominant activities that could liberate any gaseous isotopes. ISF Facility effluent monitoring consists of stack sampling for particulate radionuclides and stack sampling for ^{129}I and ^{3}H.

A particulate sample will be collected weekly depending on the work in process, ^{129}I samples will be collected biweekly and ^{3}H samples collected monthly. Iodine samples will be collected on a silver zeolite impregnated charcoal canister and ^{3}H will be collected using a three-stage bubbler collection system. Both of these sampling methods are proven methods for gas sampling. Action limits will be set at some fraction of the 10 CFR 20 Appendix B Table 2 limits. Typical action limits are 50 and 100 percent of the 10 CFR 20, Appendix B, Table 2 value.

7.6.1.2 Liquid Effluent Monitoring

Liquid effluents will not be monitored at the ISF Facility, which is designed as a zero-discharge facility relative to process generated liquids. The planned processes at the ISF Facility are inherently dry processes that do not generate liquids with the exception of the planned decontamination station and emergency eye wash stations and showers. Process liquids generated at the facility will be collected and transferred to the liquid waste storage tanks. Periodically a liquid treatment vendor will be contracted to dispose the collected liquid. Section 6.3, *Liquid Waste Treatment and Retention*, provides a discussion on management process generated liquid waste.

7.6.1.3 Solid Waste Monitoring

Solid waste generated during ISF Facility operations consists of spent PPE, metal, paper, rags, and other consumable items. Process-generated waste will be packaged for disposal at the INEEL site.

Process waste generated in the FPA will be transferred to the SWPA via the SWPA port. Before empty canisters being transferred to the SWPA for size reduction and packaging, the canisters will be assayed to determine the presence of any "hot particles" that would present an ALARA concern during the processing of the canisters.

Section 6.4, *Solid Waste*, discusses the procedure for managing site generated solid waste before waste being transferred to the SWPA it will be surveyed using portable or fixed survey equipment to verify that the administrative limit of 50 mrem/hr is not exceeded.

7.6.1.4 Environmental Monitoring

Pre-Operational Environmental Radiation Monitoring Program

The ISF Facility will be constructed on a clean (i.e., non-contaminated) site adjacent to other INEEL facilities. Construction activities will be performed by local craft without the need for radiological controls. The anticipated annual dose to workers during construction of the ISF Facility from nearby facilities is less than 0.32 mrem/yr (Ref. 7-21).

The ISF Facility is completely surrounded by the INEEL site. Many other radiological facilities are operated on the INEEL site, and some of them are near the ISF Facility. As described in Section 7.1, INEEL maintains a single environmental monitoring program for areas outside of these individual facilities to avoid excessive costs, if multiple organizations wanted to collect duplicate data; and to avoid technical conflicts should various methodologies be employed that might give conflicting results.

The ISF Facility pre-operational environmental radiation-monitoring program (PERMP) is based on the consideration that the site is radiologically clean to start and that the area of interest is limited to the ISF Facility itself. The program will provide background information to serve as a basis for later direct comparison with operational conditions. The program will address the following media:

- direct radiation exposure
- radionuclide concentrations in site air
- concentrations of radionuclides in site soil

Water will not be a medium of consideration because there will be no radioactive liquid discharges from the facility and because the possible contamination of surface or groundwater will be indirect, through either previous contamination of soil or previous emissions of contaminants in air.

Types of Samples/Measurements

Three types of samples or measurements will be taken/made as part of the PERMP, corresponding directly to the media considered.

Direct radiation will be measured at the ISF Facility boundary to develop knowledge of local background. Ten or more environmental TLDs will be placed on the ISF Facility boundary fence after its installation and exchanged quarterly. The locations will be chosen by the ES&H Manager. The doses received will be evaluated by a contractor approved for this kind of measurement by the NVLAP. Records of these doses received will be maintained in accordance with the project records system.

Ten TLDs adequately represent doses at the site boundary, and early installation provides for the longest period of time for data collection before operations. This gives the best available understanding of site conditions and any systematic variations that might occur.

Low volume particulate air samples will be collected at four locations on the construction site to develop a benchmark for operations. One location is in the predominant wind direction. Three others are located at directions approximately 90, 180, and 270 degrees from this predominant wind direction. Each air sampler typically will draw 1 to 2 cubic feet per minute samples on a continuous basis. The filter paper samples will be generally collected weekly and analyzed for gross alpha and beta radioactivity. The particulate air sampling portion of the program will be initialed after completion of both major soil movements and facility concrete work, and as effected by the installation of electric power.

Four sample stations are appropriate to measure airborne particulate radionuclides, because at least one is expected to be available to assess ground level releases, regardless of wind direction. Early installation and operation of the samplers are not justified because of the additional dust that is expected during earth moving and major construction activities. Weekly exchanges will give sufficient volumes of air for reasonable determinations of background concentrations.

Site soil will be collected and analyzed for radionuclides to establish background concentrations for reference. Five soil samples will be collected at the start of construction from randomly chosen locations within the ISF site and analyzed. The analyses performed will be gross alpha and beta analyses and gamma isotopic analyses. By using a one-tailed tolerance test, the choice of five randomly selected samples gives 75 percent assurance that 75 percent of the site soil will have concentrations less than the highest of the five concentrations measured, for each analyte. The following table summarizes the planned PERMP sampling frequency, and Figure 7.6-1 identifies each sample location.

Sample Media	No.	Frequency	Analysis
Air	4	Weekly	Gross alpha, beta, and gamma energy analysis
Soil	5	Start of/during construction	Gross alpha, beta, and gamma energy analysis
TLD	10	Quarterly	Thermoluminescence

Relationship to the Operational Environmental Radiation Monitoring Program (OERMP)

The PERMP will be carried over into the operational phase directly. During operations, soil samples will be collected quarterly, or after incidents involving a radioactive spill, or when routine surveys of the outdoor areas indicate unexpected anomalies. During eventual decontamination and decommissioning, additional soil samples will be taken to assess site conditions. At this time the number of samples will be chosen to give the desired degree of assurance that site conditions meet regulatory needs. The following table summarizes the planned OERMP sampling frequency; Figure 7.6-1 and Figure 7.6-2 identify each sample location.

Sample Media	No.	Frequency	Analysis
Air	5	Weekly	Gross alpha, beta, and gamma energy analysis
Soil	5	Quarterly	Gross alpha, beta, and gamma energy analysis
TLD	10	Monthly	Thermoluminescence

Input to the Operational Environmental Radiation Monitoring Program

Information gained during the pre-operational phase of the radiological monitoring program will be used to modify the plans for the operational phase in cases where unusual variation is seen. For example, if dose rates at two adjacent locations along the fence differ consistently, there may be a need to add an environmental TLD somewhere between them. However, because of the nature of the terrain, and the lack of planned liquid effluents, significant changes in the pre-operational program are not anticipated in the operational phase.

7.6.2 Analysis of Multiple Contribution

The total annual exposure to the maximally exposed individual (MEI) from multiple contribution of other nearby facilities and site background radiation will be less than both NRC and EPA limits. Nearby facilities that contribute to the background dose include the INTEC, Test Reactor Area (TRA), and Power Burst Facility/Waste Experimental Reduction Facility (PBF/WERF). These facilities are within 5 miles of the ISF Facility and contribute less than 0.32 mrem/year to the background radiation (Ref. 7-21).

The estimated dose from a loaded ISF canister in the storage vault through the vault wall at 1 foot is 15 mrem/hr. Likewise, the dose rate from fuel handling operations through the FPA walls is calculated to be less than 1 mrem/hr. Given the dose rates and the distance to the controlled boundary, the direct radiation contribution from ISF operations is negligible at the controlled area boundary.

During normal operations the FPA is estimated to add another 0.00003 mrem/yr additional dose due to direct and air-scattered radiation at the controlled area boundary.

To estimate the air-scattered radiation associated with leakage from 1) the ISF canister during the time it is in the CHM and 2) the loaded Storage Vault a standard Gaussian plume model was used. Both models used 1) the same source term input as the CAP88 model for the FPA estimate, 2) 1% element breakage, 3) a leakage rate of 10^{-4} cm^3/sec for the ISF canister and the storage tube, and the form and release fractions of ISG5 and ANSI/ANS 5.10. The ISF canister in the CHM model used a year of operations (80 transfers) and the Storage Vault model was based on 244 loaded storage tubes. Both models result in annual dose rates approximately two orders of magnitude less than the model for the FPA.

A total dose of less than 0.32 mrem/yr is estimated from all sources (i.e., nearby facilities and ISF operations). These exposures are well below the 10 CFR 72.104 limit of 25 mrem/yr and the 40 CFR 61.92 limit of 10 mrem/yr (Ref. 7-20).

7.6.3 Estimated Dose Equivalents

The source term for the MEI associated with ISF Facility operations consists of gases and particulate fractions of the material repackaged. The source term input into the CAP88 model is derived from the Peach Bottom Core 2 source term provided in Section 7.2. Estimated releases of gases and particulate are calculated using guidance from Interim Staff Guidance (ISG) 5 and ANSI/ANS5.10. Abatement emission factors were calculated using 40 CFR 61, Appendix D, Table 1 values. Table 7.6-1 provides the radionuclide ISG5 form and release fractions, ANSI/ANS5.10 airborne release fraction, and abatement factors for each radionuclide in the source term.

In the CAP88 model for the FPA airborne release the standard ISG5 release fractions (0.3 for gas, 0.0002 for volatiles, and 0.00003 for particulates) were used. These release fractions are based on PWR and BWR fuels. A parametric assessment was performed to evaluate the effect of using higher release fractions to account for the lack of specific similar data for the Shippingport, Peach Bottom and TRIAG fuels. The three ISG5 release fractions were all increased to 1.0 (100% release of the gas, volatiles, particulates from the fuel). This resulted in a change from 3.0×10^{-5} mrem/year to 7.9×10^{-5} mrem/year for 1% rod breakage. If the rod breakage were increased to 10% the result would change to 7.9×10^{-4} mrem/year. Therefore, it is concluded that the unavailability of specific release fractions for these fuels has no significant influence on being able to demonstrate the chronic release remains well within the regulatory limits.

The U.S. Environmental Protection Agency (EPA) CAP88 code is used to calculate the Effective Dose Equivalent (EDE) to the MEI, which includes the 50year Committed Effective Doses Equivalent (CEDE) to the MEI from internal exposures through the inhalation pathway. Meteorological data collected from the 10meter level of the Grid III meteorological tower is used for modeling releases from the ISF Facility stack. This meteorological data is representative of the conditions at the ISF facility due to the close location of the tower. Chapter 2, *Site Characteristics*, provides more detailed site characteristics.

The dose from ISF Facility operations, calculated by CAP88 code to the MEI, at Frenchman's Cabin (13.7 kilometers south southwest of the ISF Facility) is 0.00003 mrem/yr. This dose, when added to the nearby facilities dose of less than 0.32 mrem/yr, is well below the 10 mrem/yr limit to the public stipulated by 10 CFR 20.1101. Table 7.6-2 provides the annual whole-body doses estimated to be attributable to the ISF Facility stack effluent in each of the 16 compass sectors about the facility between each of the arcs having radii of 1.5, 3, 5, 6.5, and 8 kilometers. An 8kilometer radius centered on the ISF Facility is still within the INEEL controlled area. Therefore, the collective dose to any real individual outside of the restricted area, based on the 2000 census distribution presented in Chapter 2, is negligible.

7.6.3.1 Identification of Sources

Doses from stack exhaust will be determined by measuring the amounts of radionuclides emitted and meteorological modeling. The effluent concentrations will be determined using an isokinetic sampler in the ISF Facility stack and analyzing the activity of the various radionuclides collected. Modeling is based on information from the NOAA Air Resources Laboratory. The calculated offsite concentrations will be then converted to doses based on dose conversion factors developed in accordance with ICRP 30 and standard inhalation rates.

Modeling the estimated source term using CAP88 identified ^{129}I and ^{3}H as the dose contributing radionuclides released from the ISF stack. They contribute 33 and 61 percent of the dose to the maximally exposed individual respectively.

A description of the characteristics of each radionuclide listed above pertinent to its release and eventual biological impact follows.

Tritium (^{3}H) is produced both by ternary fission and by neutron reaction with light elements such as boron in control rods. Most of the fission product tritium produced is retained within the fuel and only a minor fraction is released to the environment (National Council on Radiation Protection Report No. 62). Tritium released to the atmosphere from ISF Facility operations will be from gases liberated due to rod breakage. The ISG5 rod breakage fraction of 1 percent for normal operations is used to estimate the activity of tritium available for release to the atmosphere.

Tritium release to the environment enters the hydraulic cycle. Transfer to humans is by inhalation, passage through skin, and ingestion in food and drinking water. The biological impact of tritium released from the ISF facility will be minimal (less than 20 percent of the 10 CFR 20, Appendix B, Table 2 effluent concentration).

Radionuclide ^{129}I is a fission product produced from ^{235}U in a thermal reactor. Because nearly all of the fission products generated in nuclear fuels are retained within the fuel cladding, the ^{129}I released to the atmosphere from ISF Facility operations will be from gases liberated due to rod breakage. The ISG5 rod breakage fraction of 1 percent for normal operations is used to estimate the activity available for release to the atmosphere.

Iodine released to the environment can enter the human food chain by depositing directly on vegetation. Ingestion of contaminated vegetation and of dairy products and meat from animals feeding on contaminated forage is the dominant pathway for ^{129}I exposure to humans. The biological impact of ^{129}I

released from the ISF Facility will be minimal (less than 2 percent of the 10 CFR 20, Appendix B, Table 2 effluent concentration).

7.6.3.2 Analysis of Effects and Consequences

EPA code CAP88 is used to calculate the EDE to the MEI from ISF Facility emissions. The EDE calculated by CAP88 incorporates local meteorological data in the form of joint frequency distribution of wind speed, wind direction, and atmospheric stability class. CAP88 evaluates various biological pathways and dose to critical organs.

CAP88 computes radionuclide concentrations in air, rates of deposition on ground surfaces, concentrations in food, and intake rates to people from inhalation of air and ingestion of food produced in the assessment area. The radionuclide concentrations in produce, leafy vegetables, milk, and meat consumed by humans are estimated by coupling the output of the atmospheric transport models with the Regulatory Guide 1.109 terrestrial food chain models. Dose and risk factors are provided for the pathways of ingestion and inhalation intake, ground level air immersion, and ground surface irradiation. Table 7.6-3 provides the risk factors for each radionuclide that contributes more than 10 percent of the total dose from ISF Facility emissions.

The Gaussian plume model used in CAP88 to estimate dispersion of radionuclides in air is one of the most commonly used models. Its results agree with experimental data, as well as with results of other models. The EPA Office of Radiation Programs has compared predictions of annual average ground-level concentration to actual environmental measurements and found good agreement.

Table 7.6-4 provides the radionuclide activity, release fractions, abatement fractions and abated potential to emit. The data is input into CAP88. Table 7.6-1 identifies the stack parameters (i.e., stack height, stack exit velocity), agriculture data (i.e., cattle density, land fraction crops), and code parameters (e.g., annual precipitation, annual temperature) input into CAP88.

7.6.4 Liquid Release

The ISF Facility is designed to collect process-generated liquids. No radioactive liquids are expected to be released from the facility.

7.6.4.1 Treated Process Effluent (from Waste Treatment Area)

The ISF Facility does not have any treated liquid process effluents. Section 6.3, *Liquid Waste Treatment and Retention*, discusses the management of process liquid waste generated.

7.6.4.2 Sewage

Liquids generated during the decontamination of personnel (e.g., eye wash stations and showers) will not be discharged to the facility sewer.

7.6.4.3 Drinking Water

A radioactive liquid release from the ISF Facility to drinking water is not a contamination pathway.

7.6.4.4 Rain Runoff

ISF Facility operations will be conducted inside the facility. No radioactive liquid releases from rain runoff are expected.

7.6.4.5 Laundry Waste

Laundry services will not be performed at the ISF Facility. Cleaning of nondisposable PPE will be contracted with an approved vendor.

7.6.4.6 Items Requiring Further Development

There are no items requiring further development. The facility is designed for zero liquid releases.

7.6.4.7 Changes Since Initial Submittal

No changes to the ISF Facility design are planned that will cause liquid release to be a contamination pathway.

7.7 REFERENCES

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7-2. U.S. Nuclear Regulatory Commission, Regulatory Guide 8.10, *Operating Philosophy for Maintaining Occupational Radiation Exposures As Low As Is Reasonably Achievable*, Revision 1R, September 1975.

7-3. Title 10, Code of Federal Regulations, Part 20, *Standards for Protection Against Radiation.*

7-4. ORIGEN2, Isotope Generation and Depletion Code System–Matrix Exponential Method, CCC217, Oak Ridge National Laboratory, RSIC Computer Code Collection.

7-5. ANSI/ANS5.101998, *Airborne Release Fractions at Non-Reactor Nuclear Facilities.*

7-6. Title 10, Code of Federal Regulations, Part 72, *Licensing Requirements for the Independent Storage of Spent Nuclear Fuel and High-Level Radioactive Waste.*

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7-9. Hall (2001), fax from Art Hall, Oregon State University, to Dave Hess, Morh and Associates, March 26, 2001.

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7-13. U.S. Nuclear Regulatory Commission, Regulatory Guide 8.11, *Applications of Bioassay for Uranium*, Revision 1, June 1974.

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7-18. U.S. Nuclear Regulatory Commission, NUREG1400, *Air Sampling in the Workplace*, Revision 1, June 1993.

7-19. U.S. Nuclear Regulatory Commission, Regulatory Guide 8.14, *Acceptable Programs for Respiratory Protection*, Revision 1, October 1999.

7-20. Title 40, Code of Federal Regulations, Part 61, *National Emission Standards for Hazardous Air Pollutants*.

7-21. DOEID (2000), *Idaho High-Level Waste and Facilities Disposition Draft Environmental Impact Statement,* DOE/EIS0287, December.

Table 7.1-1
Instruments Used in the Low-Background Counting Room

Instrument	No.	Media	Analysis
HPGe system	1	Solids, liquids, air filter	Gamma energy analysis
Low-background alpha-beta counter	1	Air filters, wipes	Gross alpha and beta
End-window G/M	2	Solids, filters, wipes	Gross activity (beta-gamma) analysis
Liquid scintillation	1	Gas sampling media, wipes, solids	Low-energy beta analysis

Table 7.1-2
Instruments Used to Measure Radiation Dose Rates, Levels of Contamination, and Concentrations of Airborne Radioactivity in the Field

Application/ Instrument	Individual Monitoring	Portable Instruments	Fixed Locale Instruments	Numbers of Instruments	Instrument Ranges
Dose Rate					
Area radiation monitors			X	15	0.1 to 100* or 0.1 to 1000*
Low dose rate instruments		X		6	0 to 5* or 0 to 50*
High dose rate instruments		X		3	0 to 5000*
Neutron dose rate instruments		X		3	0 to 1000*
TLDs	X		X	Personnel/area	0 to 1000**
Self-reading dosimeter	X			Personnel	0 to 200*
Alarming dosimeters	X			10% of Personnel	0 to 200*
Criticality dosimeters	X		X	Personnel/area	N/A
Contamination					
Alpha scintillation		X		3	0 to 50,000***
Alpha proportional		X		3	0 to 50,000***
GM frisker		X		9	0 to 50,000***
Scaler			X	2	0 to 100000 cts
Floor monitor		X		2	0 to 50,000***
Hand and foot monitors			X	2	0 to 5000***
Portal monitors			X	1	0 to 5000***
Airborne Radioactivity					
Hi-volume air sampler		X		3	10 to 30 cfm
Low-volume air sampler			X	12	1-5 cfm
Continuous air monitors			X	10	1 to 5 cfm
Lapel air samplers	X			4	1 to 2 Lpm
Stack monitor			X	1	1-5 cfm

Note:

*mrem/hr
**rem/hr
***counts per minute

Table 7.2-1
TRIGA Fuel Element Module Activities (curies)

Nuclide	Activity	% Activity
^{225}Ac	6.44×10^{-10}	0.00%
^{227}Ac	3.61×10^{-09}	0.00%
^{228}Ac	1.11×10^{-09}	0.00%
^{110}Ag	2.35×10^{-07}	0.00%
^{110m}Ag	1.77×10^{-05}	0.00%
^{241}Am	9.00×10^{-03}	0.01%
^{242}Am	9.29×10^{-06}	0.00%
^{242m}Am	9.34×10^{-06}	0.00%
^{243}Am	9.82×10^{-07}	0.00%
^{217}At	6.44×10^{-10}	0.00%
^{137m}Ba	$1.68\times10^{+01}$	17.93%
^{10}Be	1.34×10^{-07}	0.00%
^{211}Bi	3.61×10^{-09}	0.00%
^{212}Bi	3.06×10^{-07}	0.00%
^{213}Bi	6.44×10^{-10}	0.00%
^{14}C	8.56×10^{-04}	0.00%
^{113m}Cd	2.00×10^{-03}	0.00%
^{115m}Cd	5.95×10^{-18}	0.00%
^{141}Ce	2.27×10^{-20}	0.00%
^{144}Ce	6.05×10^{-01}	0.65%
^{36}Cl	1.87×10^{-05}	0.00%
^{242}Cm	9.31×10^{-06}	0.00%
^{243}Cm	1.15×10^{-06}	0.00%
^{244}Cm	1.06×10^{-05}	0.00%
^{245}Cm	1.10×10^{-10}	0.00%
^{60}Co	$7.01\times10^{+00}$	7.50%
^{51}Cr	7.44×10^{-25}	0.00%
^{134}Cs	3.63×10^{-01}	0.39%
^{135}Cs	2.14×10^{-04}	0.00%
^{137}Cs	$1.77\times10^{+01}$	18.96%
^{152}Eu	9.05×10^{-03}	0.01%
^{154}Eu	9.05×10^{-02}	0.10%
^{155}Eu	1.58×10^{-01}	0.17%
^{55}Fe	$3.44\times10^{+00}$	3.68%
^{59}Fe	1.12×10^{-16}	0.00%
^{221}Fr	6.44×10^{-10}	0.00%
^{223}Fr	4.98×10^{-11}	0.00%
^{153}Gd	7.10×10^{-06}	0.00%
^{3}H	6.80×10^{-02}	0.07%
^{129}I	4.88×10^{-06}	0.00%

Nuclide	Activity	% Activity
^{129}I	2.35×10^{-08}	0.00%
^{114}In	5.60×10^{-18}	0.00%
^{114m}In	5.85×10^{-18}	0.00%
^{115m}In	4.18×10^{-22}	0.00%
^{85}Kr	$1.52\times10^{+00}$	1.63%
^{54}Mn	1.43×10^{-02}	0.02%
^{93}Mo	6.22×10^{-05}	0.00%
^{93m}Nb	1.76×10^{-04}	0.00%
^{94}Nb	5.41×10^{-05}	0.00%
^{95}Nb	3.86×10^{-09}	0.00%
^{95m}Nb	1.29×10^{-11}	0.00%
^{59}Ni	3.61×10^{-03}	0.00%
^{63}Ni	4.23×10^{-01}	0.45%
^{237}Np	8.25×10^{-06}	0.00%
^{238}Np	4.67×10^{-08}	0.00%
^{239}Np	9.82×10^{-07}	0.00%
^{231}Pa	2.07×10^{-08}	0.00%
^{233}Pa	8.25×10^{-06}	0.00%
^{234}Pa	6.79×10^{-08}	0.00%
^{234m}Pa	5.22×10^{-05}	0.00%
^{209}Pb	6.44×10^{-10}	0.00%
^{211}Pb	3.61×10^{-09}	0.00%
^{212}Pb	3.06×10^{-07}	0.00%
^{107}Pd	4.17×10^{-06}	0.00%
^{145}Pm	7.75×10^{-06}	0.00%
^{147}Pm	$9.40\times10^{+00}$	10.07%
^{148}Pm	4.90×10^{-19}	0.00%
^{148m}Pm	8.70×10^{-18}	0.00%
^{211}Po	1.01×10^{-11}	0.00%
^{212}Po	1.96×10^{-07}	0.00%
^{213}Po	6.31×10^{-10}	0.00%
^{215}Po	3.61×10^{-09}	0.00%
^{216}Po	3.06×10^{-07}	0.00%
^{144}Pr	6.05×10^{-01}	0.65%
^{144m}Pr	7.26×10^{-03}	0.01%
^{236}Pu	6.23×10^{-08}	0.00%
^{237}Pu	2.27×10^{-22}	0.00%
^{238}Pu	6.78×10^{-03}	0.01%
^{239}Pu	3.67×10^{-02}	0.04%
^{240}Pu	1.42×10^{-02}	0.02%

Nuclide	Activity	% Activity
^{241}Pu	6.31×10^{-01}	0.68%
^{242}Pu	1.54×10^{-06}	0.00%
^{223}Ra	3.61×10^{-09}	0.00%
^{224}Ra	3.06×10^{-07}	0.00%
^{225}Ra	6.44×10^{-10}	0.00%
^{228}Ra	1.11×10^{-09}	0.00%
^{87}Rb	5.92×10^{-09}	0.00%
^{106}Rh	2.17×10^{-01}	0.23%
^{219}Rn	3.61×10^{-09}	0.00%
^{220}Rn	3.06×10^{-07}	0.00%
^{103}Ru	7.91×10^{-17}	0.00%
^{106}Ru	2.17×10^{-01}	0.23%
^{124}Sb	6.78×10^{-14}	0.00%
^{125}Sb	3.96×10^{-01}	0.42%
^{126}Sb	1.13×10^{-05}	0.00%
^{126m}Sb	8.09×10^{-05}	0.00%
^{79}Se	8.66×10^{-05}	0.00%
^{145}Sm	5.10×10^{-07}	0.00%
^{147}Sm	1.91×10^{-09}	0.00%
^{151}Sm	1.49×10^{-01}	0.16%
^{119m}Sn	3.44×10^{-03}	0.00%
^{121m}Sn	3.78×10^{-04}	0.00%
^{123}Sn	1.07×10^{-06}	0.00%
^{126}Sn	8.09×10^{-05}	0.00%
^{89}Sr	1.23×10^{-12}	0.00%
^{90}Sr	$1.67\times10^{+01}$	17.90%
^{160}Tb	2.29×10^{-12}	0.00%
^{99}Tc	2.94×10^{-03}	0.00%
^{123}mTe	1.39×10^{-10}	0.00%
^{125}mTe	9.67×10^{-02}	0.10%
^{127}Te	2.46×10^{-07}	0.00%
^{127m}Te	2.52×10^{-07}	0.00%
^{129}Te	1.39×10^{-21}	0.00%
^{129m}Te	2.13×10^{-21}	0.00%
^{227}Th	3.56×10^{-09}	0.00%
^{228}Th	3.05×10^{-07}	0.00%
^{229}Th	6.44×10^{-10}	0.00%
^{230}Th	8.81×10^{-11}	0.00%
^{231}Th	6.69×10^{-05}	0.00%
^{232}Th	1.68×10^{-09}	0.00%

Table 7.2-1 (continued)
TRIGA Fuel Element Module Activities (curies)

Nuclide	Activity	% Activity
^{234}Th	5.22×10^{-05}	0.00%
^{207}Tl	3.60×10^{-09}	0.00%
^{208}Tl	1.10×10^{-07}	0.00%
^{209}Tl	1.39×10^{-11}	0.00%
^{232}U	3.20×10^{-07}	0.00%
^{233}U	8.12×10^{-07}	0.00%
^{234}U	9.91×10^{-07}	0.00%

Nuclide	Activity	% Activity
^{235}U	6.69×10^{-05}	0.00%
^{236}U	8.44×10^{-05}	0.00%
^{237}U	1.55×10^{-05}	0.00%
^{238}U	5.22×10^{-05}	0.00%
^{90}Y	$1.67 \times 10^{+01}$	17.90%
^{91}Y	1.31×10^{-10}	0.00%
^{65}Zn	4.07×10^{-04}	0.00%

Nuclide	Activity	% Activity
^{93}Zr	5.29×10^{-04}	0.00%
^{95}Zr	1.74×10^{-09}	0.00%
Total	**$9.34 \times 10^{+01}$**	**100%**

Table 7.2-2
Peach Bottom Core 1 Radionuclide Activities (curies)

Nuclide	Activity	% Activity
^{225}Ac	5.97×10^{-04}	0.01%
^{227}Ac	2.22×10^{-04}	0.01%
^{228}Ac	1.66×10^{-04}	0.00%
^{241}Am	1.98×10^{-01}	4.25%
^{242}Am	8.82×10^{-05}	0.00%
^{242m}Am	8.86×10^{-05}	0.00%
^{243}Am	8.90×10^{-05}	0.00%
^{217}At	5.97×10^{-04}	0.01%
^{210}Bi	3.14×10^{-08}	0.00%
^{211}Bi	2.22×10^{-04}	0.01%
^{212}Bi	1.56×10^{-04}	0.00%
^{213}Bi	5.97×10^{-04}	0.01%
^{214}Bi	9.75×10^{-08}	0.00%
^{14}C	2.53×10^{-03}	0.05%
^{36}Cl	7.17×10^{-05}	0.00%
^{242}Cm	7.31×10^{-05}	0.00%
^{244}Cm	1.72×10^{-03}	0.04%
^{245}Cm	2.70×10^{-07}	0.00%
^{246}Cm	8.76×10^{-09}	0.00%
^{135}Cs	1.78×10^{-03}	0.04%
^{221}Fr	5.97×10^{-04}	0.01%
^{223}Fr	3.06×10^{-06}	0.00%
^{129}I	4.95×10^{-05}	0.00%
^{93m}Nb	3.31×10^{-03}	0.07%
^{94}Nb	3.32×10^{-05}	0.00%
^{59}Ni	9.87×10^{-05}	0.00%
^{63}Ni	9.87×10^{-03}	0.21%
^{237}Np	4.99×10^{-04}	0.01%
^{238}Np	4.43×10^{-07}	0.00%
^{239}Np	8.90×10^{-05}	0.00%
^{231}Pa	3.25×10^{-04}	0.01%
^{233}Pa	4.99×10^{-04}	0.01%
^{234}Pa	6.21×10^{-09}	0.00%
^{234m}Pa	4.78×10^{-06}	0.00%
^{209}Pb	5.97×10^{-04}	0.01%
^{210}Pb	3.13×10^{-08}	0.00%
^{211}Pb	2.22×10^{-04}	0.01%
^{212}Pb	1.56×10^{-04}	0.00%
^{214}Pb	9.75×10^{-08}	0.00%
^{107}Pd	3.21×10^{-05}	0.00%
^{210}Po	3.14×10^{-08}	0.00%
^{211}Po	6.22×10^{-07}	0.00%
^{212}Po	9.97×10^{-05}	0.00%
^{213}Po	5.84×10^{-04}	0.01%
^{214}Po	9.74×10^{-08}	0.00%
^{215}Po	2.22×10^{-04}	0.01%
^{216}Po	1.56×10^{-04}	0.00%
^{218}Po	9.75×10^{-08}	0.00%
^{238}Pu	$1.17\times10^{+00}$	24.91%
^{239}Pu	2.78×10^{-02}	0.59%
^{240}Pu	2.17×10^{-02}	0.46%
^{241}Pu	$1.38\times10^{+00}$	29.49%
^{242}Pu	2.86×10^{-05}	0.00%
^{223}Ra	2.22×10^{-04}	0.01%
^{224}Ra	1.56×10^{-04}	0.00%
^{225}Ra	5.97×10^{-04}	0.01%
^{226}Ra	9.75×10^{-08}	0.00%
^{228}Ra	1.66×10^{-04}	0.00%
^{219}Rn	2.22×10^{-04}	0.01%
^{220}Rn	1.56×10^{-04}	0.00%
^{222}Rn	9.75×10^{-08}	0.00%
^{126}Sb	1.18×10^{-04}	0.00%
^{126m}Sb	8.41×10^{-04}	0.02%
^{79}Se	9.10×10^{-04}	0.02%
^{151}Sm	$1.61\times10^{+00}$	34.51%
^{126}Sn	8.41×10^{-04}	0.02%
^{99}Tc	2.75×10^{-02}	0.59%
^{227}Th	2.19×10^{-04}	0.01%
^{228}Th	1.56×10^{-04}	0.00%
^{229}Th	5.97×10^{-04}	0.01%
^{230}Th	1.10×10^{-05}	0.00%
^{231}Th	4.95×10^{-04}	0.01%
^{232}Th	1.69×10^{-04}	0.00%
^{234}Th	4.78×10^{-06}	0.00%
^{207}Tl	2.21×10^{-04}	0.01%
^{208}Tl	5.59×10^{-05}	0.00%
^{209}Tl	1.29×10^{-05}	0.00%
^{233}U	1.77×10^{-01}	3.77%
^{234}U	2.94×10^{-02}	0.63%
^{235}U	4.95×10^{-04}	0.01%
^{236}U	5.14×10^{-09}	0.00%
^{237}U	3.38×10^{-05}	0.00%
^{238}U	4.78×10^{-06}	0.00%
^{93}Zr	4.33×10^{-03}	0.09%
Total	**$4.68\times10^{+00}$**	**100%**

Table 7.2-3
Peach Bottom Core 2 Radionuclide Activities (curies)

Nuclide	Activity	% Activity
^{225}Ac	9.32×10^{-04}	0.00%
^{227}Ac	3.75×10^{-04}	0.00%
^{228}Ac	1.40×10^{-04}	0.00%
^{241}Am	4.96×10^{-01}	0.05%
^{242}Am	3.65×10^{-04}	0.00%
^{242m}Am	3.67×10^{-04}	0.00%
^{243}Am	3.32×10^{-03}	0.00%
^{217}At	9.32×10^{-04}	0.00%
^{137m}Ba	$2.19 \times 10^{+02}$	23.57%
^{10}Be	1.21×10^{-04}	0.00%
^{210}Bi	5.13×10^{-08}	0.00%
^{211}Bi	3.76×10^{-04}	0.00%
^{212}Bi	1.28×10^{-01}	0.01%
^{213}Bi	9.32×10^{-04}	0.00%
^{214}Bi	1.26×10^{-07}	0.00%
^{14}C	8.34×10^{-03}	0.00%
^{113m}Cd	1.48×10^{-02}	0.00%
^{144}Ce	1.36×10^{-08}	0.00%
^{36}Cl	2.12×10^{-04}	0.00%
^{242}Cm	3.02×10^{-04}	0.00%
^{243}Cm	2.82×10^{-03}	0.00%
^{244}Cm	3.09×10^{-01}	0.03%
^{245}Cm	1.16×10^{-04}	0.00%
^{246}Cm	1.70×10^{-05}	0.00%
^{60}Co	2.69×10^{-01}	0.03%
^{134}Cs	3.62×10^{-02}	0.00%
^{135}Cs	3.15×10^{-03}	0.00%
^{137}Cs	$2.32 \times 10^{+02}$	24.91%
^{152}Eu	2.17×10^{-02}	0.00%
^{154}Eu	$2.99 \times 10^{+00}$	0.32%
^{155}Eu	2.97×10^{-01}	0.03%
^{55}Fe	5.01×10^{-04}	0.00%
^{221}Fr	9.32×10^{-04}	0.00%
^{223}Fr	5.17×10^{-06}	0.00%
^{3}H	$1.21 \times 10^{+00}$	0.13%
^{129}I	1.25×10^{-04}	0.00%
^{85}Kr	$8.51 \times 10^{+00}$	0.92%
^{93}Mo	3.29×10^{-06}	0.00%
^{93m}Nb	7.50×10^{-03}	0.00%
^{94}Nb	9.85×10^{-05}	0.00%

Nuclide	Activity	% Activity
^{59}Ni	2.80×10^{-04}	0.00%
^{63}Ni	2.96×10^{-02}	0.00%
^{237}Np	2.09×10^{-03}	0.00%
^{238}Np	1.83×10^{-06}	0.00%
^{239}Np	3.32×10^{-03}	0.00%
^{231}Pa	5.93×10^{-04}	0.00%
^{233}Pa	2.09×10^{-03}	0.00%
^{234}Pa	4.38×10^{-09}	0.00%
^{234m}Pa	3.37×10^{-06}	0.00%
^{209}Pb	9.32×10^{-04}	0.00%
^{210}Pb	5.13×10^{-08}	0.00%
^{211}Pb	3.76×10^{-04}	0.00%
^{212}Pb	1.28×10^{-01}	0.01%
^{214}Pb	1.26×10^{-07}	0.00%
^{107}Pd	7.63×10^{-05}	0.00%
^{145}Pm	3.57×10^{-05}	0.00%
^{147}Pm	1.93×10^{-01}	0.02%
^{210}Po	4.82×10^{-08}	0.00%
^{211}Po	1.05×10^{-06}	0.00%
^{212}Po	8.18×10^{-02}	0.01%
^{213}Po	9.12×10^{-04}	0.00%
^{214}Po	1.26×10^{-07}	0.00%
^{215}Po	3.76×10^{-04}	0.00%
^{216}Po	1.28×10^{-01}	0.01%
^{218}Po	1.26×10^{-07}	0.00%
^{144}Pr	1.36×10^{-08}	0.00%
^{236}Pu	1.20×10^{-07}	0.00%
^{238}Pu	$1.73 \times 10^{+01}$	1.86%
^{239}Pu	2.43×10^{-02}	0.00%
^{240}Pu	3.06×10^{-02}	0.00%
^{241}Pu	$4.66 \times 10^{+00}$	0.50%
^{242}Pu	3.69×10^{-04}	0.00%
^{223}Ra	3.76×10^{-04}	0.00%
^{224}Ra	1.28×10^{-01}	0.01%
^{225}Ra	9.32×10^{-04}	0.00%
^{226}Ra	1.26×10^{-07}	0.00%
^{228}Ra	1.40×10^{-04}	0.00%
^{87}Rb	1.40×10^{-07}	0.00%
^{106}Rh	4.64×10^{-07}	0.00%
^{219}Rn	3.76×10^{-04}	0.00%

Nuclide	Activity	% Activity
^{220}Rn	1.28×10^{-01}	0.01%
^{222}Rn	1.26×10^{-07}	0.00%
^{106}Ru	4.64×10^{-07}	0.00%
^{125}Sb	1.81×10^{-02}	0.00%
^{126}Sb	3.36×10^{-04}	0.00%
^{126m}Sb	2.40×10^{-03}	0.00%
^{79}Se	2.44×10^{-03}	0.00%
^{147}Sm	2.06×10^{-08}	0.00%
^{151}Sm	$1.75 \times 10^{+00}$	0.19%
^{121m}Sn	4.50×10^{-04}	0.00%
^{126}Sn	2.40×10^{-03}	0.00%
^{90}Sr	$2.20 \times 10^{+02}$	23.62%
^{99}Tc	5.56×10^{-02}	0.01%
^{125m}Te	4.41×10^{-03}	0.00%
^{227}Th	3.71×10^{-04}	0.00%
^{228}Th	1.27×10^{-01}	0.01%
^{229}Th	9.32×10^{-04}	0.00%
^{230}Th	1.47×10^{-05}	0.00%
^{231}Th	1.63×10^{-04}	0.00%
^{232}Th	1.44×10^{-04}	0.00%
^{234}Th	3.37×10^{-06}	0.00%
^{207}Tl	3.75×10^{-04}	0.00%
^{208}Tl	4.59×10^{-02}	0.01%
^{209}Tl	2.01×10^{-05}	0.00%
^{232}U	1.20×10^{-01}	0.01%
^{233}U	3.14×10^{-01}	0.03%
^{234}U	3.89×10^{-02}	0.00%
^{235}U	1.63×10^{-04}	0.00%
^{236}U	1.89×10^{-03}	0.00%
^{237}U	1.14×10^{-04}	0.00%
^{238}U	3.37×10^{-06}	0.00%
^{90}Y	$2.20 \times 10^{+02}$	23.63%
^{93}Zr	9.83×10^{-03}	0.00%
Total	**$9.31 \times 10^{+02}$**	**100%**

Table 7.2-4
Shippingport Type IV Reflector Module Activities (curies)

Nuclide	Activity	% Activity
^{225}Ac	7.24×10^{-02}	0.00%
^{227}Ac	1.65×10^{-01}	0.01%
^{228}Ac	1.48×10^{-01}	0.01%
^{241}Am	3.98×10^{-04}	0.00%
^{242}Am	3.45×10^{-07}	0.00%
^{242m}Am	3.46×10^{-07}	0.00%
^{243}Am	2.02×10^{-08}	0.00%
^{217}At	7.24×10^{-02}	0.00%
^{137m}Ba	$5.57\times10^{+02}$	22.02%
^{10}Be	5.23×10^{-08}	0.00%
^{210}Bi	3.75×10^{-06}	0.00%
^{211}Bi	1.65×10^{-01}	0.01%
^{212}Bi	$7.39\times10^{+00}$	0.29%
^{213}Bi	7.24×10^{-02}	0.00%
^{214}Bi	1.24×10^{-05}	0.00%
^{14}C	3.60×10^{-02}	0.00%
^{113m}Cd	3.93×10^{-02}	0.00%
^{144}Ce	4.16×10^{-05}	0.00%
^{36}Cl	5.44×10^{-07}	0.00%
^{242}Cm	2.86×10^{-07}	0.00%
^{243}Cm	1.68×10^{-08}	0.00%
^{244}Cm	1.34×10^{-07}	0.00%
^{60}Co	$2.63\times10^{+00}$	0.10%
^{134}Cs	7.03×10^{-02}	0.00%
^{135}Cs	1.02×10^{-02}	0.00%
^{137}Cs	$5.89\times10^{+02}$	23.28%
^{152}Eu	9.36×10^{-02}	0.00%
^{154}Eu	7.17×10^{-01}	0.03%
^{155}Eu	6.22×10^{-01}	0.03%
^{55}Fe	6.65×10^{-02}	0.00%
^{221}Fr	7.24×10^{-02}	0.00%
^{223}Fr	2.28×10^{-03}	0.00%
^{3}H	$1.41\times10^{+00}$	0.06%
^{129}I	4.45×10^{-04}	0.00%
^{85}Kr	$4.75\times10^{+01}$	1.88%
^{54}Mn	6.73×10^{-08}	0.00%
^{93}Mo	7.49×10^{-05}	0.00%
^{93}mNb	2.28×10^{-02}	0.00%
^{94}Nb	3.04×10^{-03}	0.00%
^{59}Ni	9.81×10^{-04}	0.00%
^{63}Ni	1.03×10^{-01}	0.00%
^{237}Np	2.64×10^{-08}	0.00%
^{239}Np	2.02×10^{-08}	0.00%
^{231}Pa	3.08×10^{-01}	0.01%
^{233}Pa	2.64×10^{-08}	0.00%
^{234m}Pa	3.49×10^{-07}	0.00%
^{209}Pb	7.24×10^{-02}	0.00%
^{210}Pb	3.75×10^{-06}	0.00%
^{211}Pb	1.65×10^{-01}	0.01%
^{212}Pb	$7.39\times10^{+00}$	0.29%
^{214}Pb	1.24×10^{-05}	0.00%
^{107}Pd	9.47×10^{-05}	0.00%
^{145}Pm	6.14×10^{-04}	0.00%
^{147}Pm	$6.81\times10^{+00}$	0.27%
^{210}Po	3.48×10^{-06}	0.00%
^{211}Po	4.63×10^{-04}	0.00%
^{212}Po	$4.73\times10^{+00}$	0.19%
^{213}Po	7.08×10^{-02}	0.00%
^{214}Po	1.24×10^{-05}	0.00%
^{215}Po	1.65×10^{-01}	0.01%
^{216}Po	$7.39\times10^{+00}$	0.29%
^{218}Po	1.24×10^{-05}	0.00%
^{144}Pr	4.16×10^{-05}	0.00%
^{144m}Pr	5.00×10^{-07}	0.00%
^{238}Pu	1.88×10^{-05}	0.00%
^{239}Pu	9.14×10^{-04}	0.00%
^{240}Pu	2.17×10^{-04}	0.00%
^{241}Pu	6.09×10^{-03}	0.00%
^{242}Pu	2.00×10^{-08}	0.00%
^{223}Ra	1.65×10^{-01}	0.01%
^{224}Ra	$7.39\times10^{+00}$	0.29%
^{225}Ra	7.24×10^{-02}	0.00%
^{226}Ra	1.24×10^{-05}	0.00%
^{228}Ra	1.48×10^{-01}	0.01%
^{87}Rb	4.22×10^{-07}	0.00%
^{106}Rh	1.81×10^{-04}	0.00%
^{219}Rn	1.65×10^{-01}	0.01%
^{220}Rn	$7.39\times10^{+00}$	0.29%
^{222}Rn	1.24×10^{-05}	0.00%
^{106}Ru	1.81×10^{-04}	0.00%
^{125}Sb	$1.19\times10^{+00}$	0.05%
^{126}Sb	1.49×10^{-03}	0.00%
^{126m}Sb	1.06×10^{-02}	0.00%
^{79}Se	9.79×10^{-03}	0.00%
^{147}Sm	8.64×10^{-08}	0.00%
^{151}Sm	$6.94\times10^{+00}$	0.27%
^{119m}Sn	6.51×10^{-08}	0.00%
^{121}mSn	4.57×10^{-02}	0.00%
^{126}Sn	1.06×10^{-02}	0.00%
^{90}Sr	$6.11\times10^{+02}$	24.14%
^{99}Tc	9.94×10^{-02}	0.00%
^{125m}Te	2.89×10^{-01}	0.01%
^{227}Th	1.63×10^{-01}	0.01%
^{228}Th	$7.36\times10^{+00}$	0.29%
^{229}Th	7.24×10^{-02}	0.00%
^{230}Th	1.22×10^{-03}	0.00%
^{231}Th	1.65×10^{-06}	0.00%
^{232}Th	1.58×10^{-01}	0.01%
^{234}Th	3.49×10^{-07}	0.00%
^{207}Tl	1.65×10^{-01}	0.01%
^{208}Tl	$2.66\times10^{+00}$	0.11%
^{209}Tl	1.56×10^{-03}	0.00%
^{232}U	$7.03\times10^{+00}$	0.28%
^{233}U	$3.18\times10^{+01}$	1.26%
^{234}U	2.28×10^{-01}	0.01%
^{235}U	1.65×10^{-06}	0.00%
^{236}U	4.02×10^{-07}	0.00%
^{237}U	1.49×10^{-07}	0.00%
^{238}U	3.49×10^{-07}	0.00%
^{90}Y	$6.11\times10^{+02}$	24.15%
^{93}Zr	3.42×10^{-02}	0.00%
Total	**$2.53\times10^{+03}$**	**100%**

Table 7.2-5
Shippingport Type V Reflector Module Activities (curies)

Nuclide	Activity	% Activity
^{225}Ac	5.27×10^{-02}	0.00%
^{227}Ac	1.20×10^{-01}	0.01%
^{228}Ac	1.08×10^{-01}	0.01%
^{241}Am	2.90×10^{-04}	0.00%
^{242}Am	2.51×10^{-07}	0.00%
^{242m}Am	2.52×10^{-07}	0.00%
^{243}Am	1.47×10^{-08}	0.00%
^{217}At	5.27×10^{-02}	0.00%
^{137m}Ba	$4.05 \times 10^{+02}$	22.02%
^{10}Be	3.81×10^{-08}	0.00%
^{210}Bi	2.73×10^{-06}	0.00%
^{211}Bi	1.20×10^{-01}	0.01%
^{212}Bi	$5.38 \times 10^{+00}$	0.29%
^{213}Bi	5.27×10^{-02}	0.00%
^{214}Bi	9.04×10^{-06}	0.00%
^{14}C	2.62×10^{-02}	0.00%
^{113m}Cd	2.86×10^{-02}	0.00%
^{144}Ce	3.03×10^{-05}	0.00%
^{36}Cl	3.96×10^{-07}	0.00%
^{242}Cm	2.08×10^{-07}	0.00%
^{243}Cm	1.22×10^{-08}	0.00%
^{244}Cm	9.76×10^{-08}	0.00%
^{60}Co	$1.92 \times 10^{+00}$	0.10%
^{134}Cs	5.11×10^{-02}	0.00%
^{135}Cs	7.40×10^{-03}	0.00%
^{137}Cs	$4.29 \times 10^{+02}$	23.28%
^{152}Eu	6.81×10^{-02}	0.00%
^{154}Eu	5.22×10^{-01}	0.03%
^{155}Eu	4.53×10^{-01}	0.03%
^{55}Fe	4.84×10^{-02}	0.00%
^{221}Fr	5.27×10^{-02}	0.00%
^{223}Fr	1.66×10^{-03}	0.00%
^{3}H	$1.03 \times 10^{+00}$	0.06%
^{129}I	3.24×10^{-04}	0.00%
^{85}Kr	$3.46 \times 10^{+01}$	1.88%
^{54}Mn	4.90×10^{-08}	0.00%
^{93}Mo	5.45×10^{-05}	0.00%
^{93m}Nb	1.66×10^{-02}	0.00%
^{94}Nb	2.21×10^{-03}	0.00%

Nuclide	Activity	% Activity
^{59}Ni	7.14×10^{-04}	0.00%
^{63}Ni	7.51×10^{-02}	0.00%
^{237}Np	1.92×10^{-08}	0.00%
^{239}Np	1.47×10^{-08}	0.00%
^{231}Pa	2.24×10^{-01}	0.01%
^{233}Pa	1.92×10^{-08}	0.00%
^{234m}Pa	2.25×10^{-07}	0.00%
^{209}Pb	5.27×10^{-02}	0.00%
^{210}Pb	2.73×10^{-06}	0.00%
^{211}Pb	1.20×10^{-01}	0.01%
^{212}Pb	$5.38 \times 10^{+00}$	0.29%
^{214}Pb	9.04×10^{-06}	0.00%
^{107}Pd	6.90×10^{-05}	0.00%
^{145}Pm	4.47×10^{-04}	0.00%
^{147}Pm	$4.95 \times 10^{+00}$	0.27%
^{210}Po	2.53×10^{-06}	0.00%
^{211}Po	3.37×10^{-04}	0.00%
^{212}Po	$3.45 \times 10^{+00}$	0.19%
^{213}Po	5.16×10^{-02}	0.00%
^{214}Po	9.04×10^{-06}	0.00%
^{215}Po	1.20×10^{-01}	0.01%
^{216}Po	$5.38 \times 10^{+00}$	0.29%
^{218}Po	9.04×10^{-06}	0.00%
^{144}Pr	3.03×10^{-05}	0.00%
^{144m}Pr	3.64×10^{-07}	0.00%
^{238}Pu	1.37×10^{-05}	0.00%
^{239}Pu	6.65×10^{-04}	0.00%
^{240}Pu	1.58×10^{-04}	0.00%
^{241}Pu	3.84×10^{-03}	0.00%
^{242}Pu	1.45×10^{-08}	0.00%
^{223}Ra	1.20×10^{-01}	0.01%
^{224}Ra	$5.38 \times 10^{+00}$	0.29%
^{225}Ra	5.27×10^{-02}	0.00%
^{226}Ra	9.04×10^{-06}	0.00%
^{228}Ra	1.08×10^{-01}	0.01%
^{87}Rb	3.07×10^{-07}	0.00%
^{106}Rh	1.32×10^{-04}	0.00%
^{219}Rn	1.20×10^{-01}	0.01%
^{220}Rn	$5.38 \times 10^{+00}$	0.29%

Nuclide	Activity	% Activity
^{222}Rn	9.04×10^{-06}	0.00%
^{106}Ru	1.32×10^{-04}	0.00%
^{125}Sb	8.63×10^{-01}	0.05%
^{126}Sb	1.08×10^{-03}	0.00%
^{126m}Sb	7.73×10^{-03}	0.00%
^{79}Se	7.13×10^{-03}	0.00%
^{147}Sm	6.29×10^{-08}	0.00%
^{151}Sm	$5.05 \times 10^{+00}$	0.27%
^{119m}Sn	4.74×10^{-08}	0.00%
^{121m}Sn	3.33×10^{-02}	0.00%
^{126}Sn	7.73×10^{-03}	0.00%
^{90}Sr	$4.45 \times 10^{+02}$	24.14%
^{99}Tc	7.23×10^{-02}	0.00%
^{125m}Te	2.11×10^{-01}	0.01%
^{227}Th	1.19×10^{-01}	0.01%
^{228}Th	$5.36 \times 10^{+00}$	0.29%
^{229}Th	5.27×10^{-02}	0.00%
^{230}Th	8.85×10^{-04}	0.00%
^{231}Th	1.20×10^{-06}	0.00%
^{232}Th	1.15×10^{-01}	0.01%
^{234}Th	2.54×10^{-07}	0.00%
^{207}Tl	1.20×10^{-01}	0.01%
^{208}Tl	$1.93 \times 10^{+00}$	0.11%
^{209}Tl	1.14×10^{-03}	0.00%
^{232}U	$5.12 \times 10^{+00}$	0.28%
^{233}U	$2.32 \times 10^{+01}$	1.26%
^{234}U	1.66×10^{-01}	0.01%
^{235}U	1.20×10^{-06}	0.00%
^{236}U	2.92×10^{-07}	0.00%
^{237}U	1.09×10^{-07}	0.00%
^{238}U	2.54×10^{-07}	0.00%
^{90}Y	$4.45 \times 10^{+02}$	24.15%
^{93}Zr	2.49×10^{-02}	0.00%
Total	**$1.84 \times 10^{+03}$**	**100%**

Table 7.2-6
Photon Production Rate (photons/sec) for a Single Fuel Element or Fuel Module

Group	Mean Energy (MeV)	PBC1	PBC2	SH Type IV Reflector	SH Type IV Reflector	TRIGA
1	0.010	$9.78\times10^{+09}$	$6.49\times10^{+12}$	$1.80\times10^{+13}$	$1.31\times10^{+13}$	$5.33\times10^{+11}$
2	0.025	$2.65\times10^{+08}$	$1.32\times10^{+12}$	$3.70\times10^{+12}$	$2.69\times10^{+12}$	$1.18\times10^{+11}$
3	0.038	$6.76\times10^{+07}$	$1.41\times10^{+12}$	$3.73\times10^{+12}$	$2.72\times10^{+12}$	$1.17\times10^{+11}$
4	0.058	$2.72\times10^{+09}$	$1.23\times10^{+12}$	$3.40\times10^{+12}$	$2.48\times10^{+12}$	$1.00\times10^{+11}$
5	0.085	$6.64\times10^{+07}$	$7.32\times10^{+11}$	$2.14\times10^{+12}$	$2.56\times10^{+12}$	$6.10\times10^{+10}$
6	0.125	$1.68\times10^{+07}$	$5.14\times10^{+11}$	$1.32\times10^{+12}$	$9.62\times10^{+11}$	$4.22\times10^{+10}$
7	0.225	$3.22\times10^{+07}$	$6.25\times10^{+11}$	$1.87\times10^{+12}$	$1.36\times10^{+12}$	$5.04\times10^{+10}$
8	0.375	$5.35\times10^{+07}$	$2.65\times10^{+11}$	$7.64\times10^{+11}$	$5.56\times10^{+11}$	$2.63\times10^{+10}$
9	0.575	$7.47\times10^{+07}$	$8.40\times10^{+12}$	$2.15\times10^{+13}$	$1.56\times10^{+13}$	$6.63\times10^{+11}$
10	0.850	$1.13\times10^{+07}$	$9.79\times10^{+10}$	$1.87\times10^{+11}$	$1.36\times10^{+11}$	$1.64\times10^{+10}$
11	1.250	$1.29\times10^{+06}$	$9.30\times10^{+10}$	$2.51\times10^{+11}$	$1.83\times10^{+11}$	$5.02\times10^{+11}$
12	1.750	$1.09\times10^{+06}$	$3.01\times10^{+09}$	$1.30\times10^{+10}$	$9.49\times10^{+09}$	$1.85\times10^{+08}$
13	2.250	$4.45\times10^{+02}$	$2.34\times10^{+05}$	$1.42\times10^{+06}$	$1.03\times10^{+06}$	$1.48\times10^{+08}$
14	2.750	$1.79\times10^{+06}$	$1.59\times10^{+09}$	$9.21\times10^{+10}$	$6.71\times10^{+10}$	$1.57\times10^{+06}$
15	3.500	$7.06\times10^{+01}$	$4.83\times10^{+03}$	$8.13\times10^{+02}$	$5.92\times10^{+02}$	$1.88\times10^{+05}$
16	5.000	$2.85\times10^{+01}$	$2.05\times10^{+03}$	$1.49\times10^{+02}$	$1.09\times10^{+02}$	$3.56\times10^{+00}$
17	7.000	$3.12\times10^{+00}$	$2.35\times10^{+02}$	$9.71\times10^{+00}$	$7.07\times10^{+00}$	4.03×10^{-01}
18	9.000	3.48×10^{-01}	$2.69\times10^{+01}$	6.15×10^{-01}	4.48×10^{-01}	4.59×10^{-02}
	Total	$1.31\times10^{+10}$	$2.12\times10^{+13}$	$5.70\times10^{+13}$	$4.24\times10^{+13}$	$2.23\times10^{+12}$

Table 7.2-7
Total Neutron Spectrum (neutrons/sec-cm^3) for Peach Bottom and TRIGA Fuels[1]

Group	Energy (MeV)	Peach Bottom	TRIGA
1	9.5 - 10.0	$5.54X10^{-03}$	$2.06X10^{-02}$
2	9.0 - 9.5	$8.25X10^{-03}$	$3.23X10^{-02}$
3	8.5 - 9.0	$1.22X10^{-02}$	$5.05X10^{-02}$
4	8.0 - 9.5	$2.02X10^{-02}$	$7.84X10^{-02}$
5	7.5 - 8.0	$5.17X10^{-02}$	$1.21X10^{-01}$
6	7.0 - 7.5	$5.32X10^{-01}$	$1.86X10^{-01}$
7	6.5 -7.0	$1.25X10^{+00}$	$2.85X10^{-01}$
8	6.0 - 6.5	$1.53X10^{+00}$	$4.31X10^{-01}$
9	5.5 - 6.0	$1.72X10^{+00}$	$6.48X10^{-01}$
10	5.0 -5.5	$2.33X10^{+00}$	$9.64X10^{-01}$
11	4.5 - 5.0	$2.57X10^{+00}$	$1.42X10^{+00}$
12	4.0 - 4.5	$2.28X10^{+00}$	$2.06X10^{+00}$
13	3.5 - 4.0	$2.13X10^{+00}$	$2.94X10^{+00}$
14	3.0 - 3.5	$1.39X10^{+00}$	$4.13X10^{+00}$
15	2.5 - 3.0	$8.25X10^{-01}$	$5.67X10^{+00}$
16	2.0 - 2.5	$9.99X10^{-01}$	$7.54X10^{+00}$
17	1.5 - 2.0	$1.23X10^{+00}$	$9.61X10^{+00}$
18	1.0 - 1.5	$1.45X10^{+00}$	$1.15X10^{+01}$
19	0.5 - 1.0	$1.63X10^{+00}$	$1.22X10^{+01}$
20	0.0 - 0.5	$2.06X10^{+00}$	$8.72X10^{+00}$
	Total	**$2.40X10^{+01}$**	**$6.86X10^{+01}$**

1 Note: the Shippingport Reflectors (Type IV and V) do not produce a measurable neutron spectrum.

Table 7.2-8
Postulated Airborne Radioactive Material Sources
(uCi/ml)

Nuclide	Normal	Occurrence
^{241}Am	1.10×10^{-17}	1.10×10^{-16}
^{137}Cs	2.27×10^{-13}	2.27×10^{-12}
^{129}I	6.92×10^{-13}	6.92×10^{-12}
^{238}Pu	3.82×10^{-16}	3.82×10^{-15}
^{239}Pu	5.38×10^{-19}	5.38×10^{-18}
^{240}Pu	6.77×10^{-19}	6.77×10^{-18}
^{241}Pu	1.03×10^{-16}	1.03×10^{-15}
^{151}Sm	3.87×10^{-17}	3.87×10^{-16}
^{99}Tc	1.23×10^{-18}	1.23×10^{-17}
^{233}U	6.95×10^{-18}	6.95×10^{-17}
^{234}U	8.60×10^{-19}	8.60×10^{-18}

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Table 7.3-1
Photon Intensity Spectra by Canister

Group Number	Mean Energy (MeV)	Peach Bottom 1 Total Photon Spectrum	Peach Bottom 2 Total Photon Spectrum	Shippingport Reflector (IV) Total Photon Spectrum	Shippingport Reflector (V) Total Photon Spectrum	TRIGA Total Photon Spectrum
1	0.010	$9.776x10^{+09}$	$6.494x10^{+12}$	$1.798x10^{+13}$	$1.309x10^{+13}$	$5.331x10^{+11}$
2	0.025	$2.650x10^{+08}$	$1.320x10^{+12}$	$3.696x10^{+12}$	$2.691x10^{+12}$	$1.181x10^{+11}$
3	0.038	$6.762x10^{+07}$	$1.407x10^{+12}$	$3.733x10^{+12}$	$2.718x10^{+12}$	$1.165x10^{+11}$
4	0.058	$2.718x10^{+09}$	$1.233x10^{+12}$	$3.400x10^{+12}$	$2.476x10^{+12}$	$1.003x10^{+11}$
5	0.085	$6.641x10^{+07}$	$7.320x10^{+11}$	$2.140x10^{+12}$	$2.558x10^{+12}$	$6.100x10^{+10}$
6	0.125	$1.680x10^{+07}$	$5.144x10^{+11}$	$1.321x10^{+12}$	$9.619x10^{+11}$	$4.215x10^{+10}$
7	0.225	$3.218x10^{+07}$	$6.249x10^{+11}$	$1.873x10^{+12}$	$1.364x10^{+12}$	$5.040x10^{+10}$
8	0.375	$5.350x10^{+07}$	$2.650x10^{+11}$	$7.640x10^{+11}$	$5.561x10^{+11}$	$2.632x10^{+10}$
9	0.575	$7.472x10^{+07}$	$8.403x10^{+12}$	$2.149x10^{+13}$	$1.565x10^{+13}$	$6.634x10^{+11}$
10	0.850	$1.132x10^{+07}$	$9.787x10^{+10}$	$1.872x10^{+11}$	$1.362x10^{+11}$	$1.642x10^{+10}$
11	1.250	$1.294x10^{+06}$	$9.299x10^{+10}$	$2.507x10^{+11}$	$1.826x10^{+11}$	$5.016x10^{+11}$
12	1.750	$1.088x10^{+06}$	$3.011x10^{+09}$	$1.304x10^{+10}$	$9.494x10^{+09}$	$1.846x10^{+08}$
13	2.250	$4.454x10^{+02}$	$2.340x10^{+05}$	$1.418x10^{+06}$	$1.032x10^{+06}$	$1.476x10^{+08}$
14	2.750	$1.788x10^{+06}$	$1.594x10^{+09}$	$9.213x10^{+10}$	$6.707x10^{+10}$	$1.572x10^{+06}$
15	3.500	$7.055x10^{+01}$	$4.835x10^{+03}$	$8.135x10^{+02}$	$5.922x10^{+02}$	$1.880x10^{+05}$
16	5.000	$2.847x10^{+01}$	$2.053x10^{+03}$	$1.494x10^{+02}$	$1.087x10^{+02}$	$3.556x10^{+00}$
17	7.000	$3.121x10^{+00}$	$2.349x10^{+02}$	$9.706x10^{+00}$	$7.065x10^{+00}$	$4.029x10^{-01}$
18	9.000	$3.483x10^{-01}$	$2.686x10^{+01}$	$6.150x10^{-01}$	$4.477x10^{-01}$	$4.592x10^{-02}$
Total Photons/Assembly		$1.309x10^{+10}$	$2.119x10^{+13}$	$5.694x10^{+13}$	$4.246x10^{+13}$	$2.230x10^{+12}$
Peaking Factor		1	1	1	1	2.62
Assemblies/Canister		10	10	1	1	108
Total Photons/Canister		**$1.31x10^{+11}$**	**$2.12x10^{+14}$**	**$5.69x10^{+13}$**	**$4.25x10^{+13}$**	**$6.31x10^{+14}$**

Table 7.3-2
Compositions of the Materials Modeled

316 SS[2]	Density (g/cc)=	$7.92000 \times 10^{+00}$
		Atom density
material	Atom fraction	Atoms/bcm
carbon	1.59003×10^{-03}	1.39609×10^{-04}
^{55}Mn	2.31750×10^{-02}	2.03483×10^{-03}
Silicon	2.26663×10^{-02}	1.99017×10^{-03}
^{50}Cr	9.06217×10^{-03}	7.95685×10^{-04}
^{52}Cr	1.74590×10^{-01}	1.53295×10^{-02}
^{53}Cr	1.97927×10^{-02}	1.73786×10^{-03}
^{54}Cr	4.92150×10^{-03}	4.32122×10^{-04}
^{54}Fe	4.32897×10^{-02}	3.80096×10^{-03}
^{56}Fe	6.84499×10^{-01}	6.01010×10^{-02}
^{57}Fe	1.64137×10^{-02}	1.44117×10^{-03}
	Atom dens. =	8.78029×10^{-02}
Concrete[3]	Density (g/cc)=	$2.40000 \times 10^{+00}$
		Atom density
Material	Atom fraction	Atoms/bcm
^{1}H	1.34844×10^{-01}	1.03479×10^{-02}
^{2}H	2.02305×10^{-05}	1.55249×10^{-06}
^{10}B	2.08755×10^{-05}	1.60199×10^{-06}
^{16}O	5.66232×10^{-01}	4.34526×10^{-02}
^{17}O	2.26581×10^{-04}	1.73878×10^{-05}
^{27}Al	2.03673×10^{-02}	1.56299×10^{-03}
Silicon	1.84648×10^{-01}	1.41699×10^{-02}
Calcium	8.37105×10^{-02}	6.42394×10^{-03}
^{54}Fe	5.85922×10^{-04}	4.49637×10^{-05}
^{56}Fe	9.10860×10^{-03}	6.98994×10^{-04}
^{57}Fe	2.08547×10^{-04}	1.60039×10^{-05}
^{58}Fe	2.78066×10^{-05}	2.13388×10^{-06}
	Atom dens. =	7.67400×10^{-02}

JABROC[4]	Density (g/cc)=	$1.30000 \times 10^{+00}$
		Atom density
Material	Atom fraction	Atoms/bcm
Hydrogen	4.62119×10^{-01}	4.62965×10^{-02}
Carbon	3.21879×10^{-01}	3.22468×10^{-02}
N14	8.31659×10^{-04}	8.33181×10^{-05}
N15	3.00860×10^{-06}	3.01411×10^{-07}
O16	2.11930×10^{-01}	2.12318×10^{-02}
Na23	2.8480×10^{-03}	2.85321×10^{-04}
Calcium	3.88989×10^{-04}	3.89701×10^{-05}
	Atom dens. =	1.00183×10^{-01}
AIR	Density (g/cc)=	0.00029
	Atom dens. =	1
LEAD	Density (g/cc)=	11.4
	Atom dens. =	1

1 Handbook of Chemistry and Physics, 54th Edition, CRC Press

2 American Society for Metals, Metals Handbook, Ninth Edition, 1978 Volume 1 properties and selections: Irons and Steels

3 ANSI/ANS6.41997, Nuclear Analysis and Design of Concrete Radiation Shielding for Nuclear Power Plants

4 Permuli Gloucester Limited publication 8.200/1, Typical Properties of Jabroc 'N', Permal: Gloucester Limited

Table 7.4-1
Estimated Annual Occupancy Times

Area	Annual Occupancy (hours)	Maximum Hours for an Individual (hours)	Total Person-Hours	Design Dose Rate mrem/hr
Cask Receipt Area	1600	574	2295	0.9
Cask return	1306	527	1962	0.5
FPA	1820	1820	9100	0.5
CCA	1765	466	2231	1.5
SWPA	1593	797	2390	5
Storage Area	675	364	1036	0.5

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Table 7.4-2
Summary of Occupational Dose for Fuel Handling Operations

Task	Craft (number)	Frequency	Duration (minutes)	Effective Average Person Dose Rate (mrem/hour)	Total Individual Dose (mrem/year)	Total Annual Dose (man-mrem)
Cask Receipt						
Receive cask at security gate	Operator (1)	1.25 per week	60	0.32	21	21
	RPT (1)		60 with10 for cask survey	0.512	33	33
	QA/QC (1)		60 with10 for cask inspection	0.512	33	33
	Security (1)		60	0.16	11	11
Prepare cask for transfer to trolley	Operator (2)		141 (20% at side, 50% at ends)	0.552	84	168
	RPT (1)		141 with 15 for survey	0.331	25	25
	QA/QC (1)		141 with 15 for inspection	0.166	13	13
Position and tie down cask on trolley	Operator (3)		902	0.8	78	234
	RPT (1)		90	0.8	78	78
Prepare cask for FPA unloading	Operator (3)		40	4.9	212	637
	RPT (1)		40	4.9	212	212
					Total	**1465**
Cask Return						
Survey and decontamination	Operator (3)	1.25 per week		3.82	435	1304
	RPT (1)		105	3.82	435	435
Prepare cask for shipment	Operator (2)		84	3.82	346	695
	RPT (1)			3.82	348	348
	QA/QC (1)			3.82	348	348
Cask dispatch	Operator (1)		115	0.01	1.3	1.3
	QA/QC (1)		115	0.01	1.3	1.3
	Security (1)		115	0.005	0.63	0.63
	RPT (1)		115	0.005	0.63	0.63
					Total	**3133.86**
CCA Operations						
Survey, decontamination, and weld preparation	Operator (1)	1 per week	35	3.12	97	97
	Weld Tech (2)		35	0.24	8	15
	RPT (1)		35	0.62	19	19
	QA/QC (1)		35	1.1	34	34
Close cask	Operator (1)		203	2.08	366	366
	Weld Tech (2)		203	2.08	366	732
	RPT (1)		203	1.04	183	183
	QA/QC (1)		203	1.04	183	183
Trolley Preparation	Operator (2)		18	6	94	187
					Total	**1816**
FPA Operating Gallery						
Fuel transfer operations	Operator (4)	Weekly	2100	0.045	82	328
	RPT (1)		2100	0.023	41	41
					Total	**369**

Table 7.4-2
Summary of Occupational Dose for Fuel Handling Operations

Task	Craft (number)	Frequency	Duration (minutes)	Effective Average Person Dose Rate (mrem/hour)	Total Individual Dose (mrem/year)	Total Annual Dose (man-mrem)
Storage Area Operations						
Retrieve canister from trolley	Operator (2)	1 per week	96	0.4	34	67
	RPT (1)		96	0.34	28	28
Manual opening of storage tube	Operators (2)		17	1.6	24	47
	RPT (1)		17	0.8	12	12
CHM operation	Operator (2)		58	0.4	20	40
Manual closing of storage tube	Operator (2)		29	1.6	40	80
	RPT (1)		29	0.8	20	20
	QA/QC (1)		29	1.6	40	40
					Total	**334**
SWPA Operations						
Size reduction large component	Operator (2)	5 per month	312	20 / (2)	6240 / (624)	12480 / (1248)
	RPT (1)		312	10 / (1)	3120 / (312)	3120 / (312)
Size reduction small component	Operator (2)		404	20 / (2)	8080 / (808)	16160 / (1616)
	RPT (1)		404	10 / (1)	4040 / (404)	4040 / (404)
Operate compactor	Operator (2)	2 per month	200	3	240	480
	RPT (1)		200	1.5	120	120
Waste tank operations	Operator (1)	1 per quarter	20	3	4	4
	RPT (1)			1.5	2	2
					Total	**36406 / (4186)**
Maintenance & Repair						
Routine operations	Millwright (2)	Weekly	2100	1	910	1820
	Machinist (1)			1	910	910
	Electronic Tech (2)			1	455	910
					Total	**3640**
Radiological Control						
Routine operations	RPT (4)	Weekly	2100	0.25	228	910
					Total	**910**

1 The values in this column represent estimated average dose rates in the area where personnel will be working to perform the associated task. For operations where it is considered that temporary/shadow shielding will be effective in keeping dose rates ALARA, two values are provided. The first value is the projected dose rate assuming no credit for temporary/shadow shielding. The second value is the projected dose rate when credit is taken for radiation attenuation by the use of temporary/shadow shielding.

Table 7.5-1
Minimum Routine Survey Requirements of the ISF Facility

	Routine Survey							
	Radiation				Contamination			
Location	Daily	Weekly	Monthly	Each Shift	Daily	Weekly	Monthly	Each Shift
Occupied decontamination facilities		X						X
Areas within the radiologically controlled area not normally used for processing			X				X	
Occupied radioactive sample preparation and/or counting rooms	X				X			
Accessible areas adjacent to sample preparation, counting rooms, unoccupied sample preparation, and/or unoccupied rooms.			X			X		
High radiation area boundaries		X						
Radiation area boundaries		X						
Radioactive material storage areas		2				2		
Selected areas outside radiologically controlled areas (on a rotating basis)			X				X	
Occupied contamination areas				X				X
Exits from occupied contamination areas								X
Occupied areas immediately surrounding contamination areas						X		
Vehicles in restricted area					1		X	
Owner-controlled area fence line	X							

1 Vehicles are surveyed before leaving or entering the restricted area. Before leaving a radiologically controlled area, the material shall be surveyed for release.

2 Weekly or when nonexempt material is moved in or out.

Table 7.5-2
Minimum Air Activity Survey Requirements of the ISF Facility

	Air Activity Surveys		
Requirements	**Sample Frequency**	**Sample Type**	**Comments**
Occupied contamination areas when continuous air monitor is not operating.	Once a Day	Particulate	Tritium and/or charcoal samples will be considered if the potential for tritium gas or radioactive iodine exists.
Air samples in the immediate vicinity of Radiological work that has the potential to cause airborne radioactivity.	Immediately after work starts and at a frequency determined by the ES&H Manager	Particulate	Tritium and/or charcoal samples will be considered if the potential for tritium gas or radioactive iodine exists.
Air samples upon initially entering spaces where a significant potential exists for airborne radioactivity.	Upon Entry	Particulate	Tritium and/or charcoal samples will be considered if the potential for tritium gas or radioactive iodine exists.
Operating ventilation exhaust stack	Continuously	Particulate	Tritium and/or charcoal samples will be considered if the potential for tritium gas or radioactive iodine exists.

Table 7.6-1
Parameters Postulated for Chronic Release

Data and Assumption used to Estimate Radioactive Source	
Form (physical, chemical)	Particulate and gas in oxide form
Particle size	1 micron
Physical and chemical data related to transport or removal	ISG5 release fractions: 0.3 for gas 0.0002 for volatile 0.00003 for particulate ANSI/ANS5.10 airborne release fractions: 1 for gas 0.00004 for volatile 0.00004 for particulate
Data and Assumptions Used to Estimate	
Leakage fraction	ISG5 1% rod breakage
Absorption and filtration efficiencies	Fabric filter removal factor: 0.1 HEPA filter removal factor: 0.01
Release flow rates and pathways	Stack exit velocity 13.5 m/s
Dispersion Data	
Stack or building leakage source	81foot tall, 38inch diameter stack
Building wake (ground source)	N/A
Boundary distances	13.7 kilometers
Dose (annual average by sector)	**Sector** **mrem/yr** **Sector** **mrem/yr** N $8.5x10^{-6}$ S $9.9x10^{-6}$ NNW $4.9x10^{-6}$ SSE $6x10^{-6}$ NW $3.3x10^{-6}$ SE $4x10^{-6}$ WNW $2.9x10^{-6}$ ESE $3.9x10^{-6}$ W $3.3x10^{-6}$ E $5.3x10^{-6}$ WSW $6.9x10^{-6}$ ENE $1.1x10^{-5}$ SW $1.6x10^{-5}$ NE $2.4x10^{-5}$ SSW $1.5x10^{-5}$ NNE $1.5x10^{-5}$
Deposits, decay, and washout coefficients	CAP88 calculated
Dose Data	
Dose model (code)	CAP88
Liquid and gaseous source terms	No liquid source term. Gaseous source term from fission gases liberated during fuel handling operations.
Biological pathways	Inhalation, ingestion, air immersion and ground surface irradiation
Dose model (code) parameters and input	Distance; 1.5, 3, 5, 6.5, 8, 13.7 km Wind File: GRID3L Precipitation:. 22 cm/yr Ambient Temp: 6 degrees Celsius Height of lid: 50 m Stack height: 24.7 m Diameter: 0.965 m Plume rise: 13.54 m/s Cattle density(beef) 0.072 #/km2 Cattle density (milk) 0.0084 #/km2 Land fraction crops 0.072

Table 7.6-2
Annual Whole-Body Collective Doses (Person-Rem) Estimated to be Attributable to the ISF Facility Stack Effluent

	1500 m mrem/yr	3000 m mrem/yr	5000 m mrem/yr	6500 m mrem/yr	8000 m mrem/yr
N	0.00022	0.00010	0.00004	0.00003	0.00002
NNW	0.00013	0.00006	0.00002	0.00002	0.00001
NW	0.00008	0.00004	0.00002	0.00001	0.00001
WNW	0.00007	0.00003	0.00001	0.00001	0.00001
W	0.00008	0.00004	0.00002	0.00001	0.00001
WSW	0.00018	0.00008	0.00003	0.00002	0.00002
SW	0.00046	0.00020	0.00009	0.00006	0.00004
SSW	0.00042	0.00019	0.00008	0.00005	0.00004
S	0.00024	0.00012	0.00005	0.00003	0.00002
SSE	0.00012	0.00007	0.00003	0.00002	0.00001
SE	0.00007	0.00004	0.00002	0.00001	0.00001
ESE	0.00007	0.00004	0.00002	0.00001	0.00001
E	0.00011	0.00006	0.00003	0.00002	0.00001
ENE	0.00028	0.00013	0.00006	0.00004	0.00003
NE	0.00057	0.00026	0.00012	0.00008	0.00006
NNE	0.00039	0.00018	0.00008	0.00006	0.00004

Table 7.6-3
Risk Factors for Each Radionuclide that Contributes more than 10% of the Total Dose from ISF Facility Emissions

Cancer	Ingestion[1]	Inhalation[1]	Air Immersion[2]	Ground Surface[2]
Risk Factors for ^{3}H				
Leukemia	2.62×10^{-8}	3.93×10^{-8}	0	0
Bone	1.16×10^{-9}	1.74×10^{-9}	0	0
Thyroid	3.77×10^{-9}	5.64×10^{-9}	0	0
Breast	3.25×10^{-8}	4.90×10^{-8}	0	0
Lung	4.15×10^{-8}	6.20×10^{-8}	0	0
Stomach	3.51×10^{-8}	4.06×10^{-8}	0	0
Bowel	1.93×10^{-8}	2.18×10^{-8}	0	0
Liver	2.90×10^{-8}	4.35×10^{-8}	0	0
Pancreas	1.97×10^{-8}	2.96×10^{-8}	0	0
Urinary	1.07×10^{-8}	1.62×10^{-8}	0	0
Other	2.41×10^{-8}	3.62×10^{-8}	0	0
Risk Factors for ^{129}I				
Leukemia	2.49×10^{-7}	1.66×10^{-7}	2.47	1.28×10^{-3}
Bone	1.11×10^{-8}	7.43×10^{-9}	6.41×10^{1}	3.31×10^{-4}
Thyroid	1.31×10^{-4}	8.70×10^{-5}	6.28	3.29×10^{-3}
Breast	4.65×10^{-7}	3.10×10^{-7}	$3.48 \times 10^{+1}$	1.86×10^{-2}
Lung	2.87×10^{-7}	7.46×10^{-7}	$1.02 \times 10^{+1}$	5.28×10^{-3}
Stomach	1.88×10^{-7}	7.51×10^{-8}	4.91	2.54×10^{-3}
Bowel	1.07×10^{-7}	4.13×10^{-8}	2.34	1.21×10^{-3}
Liver	1.68×10^{-7}	1.13×10^{-7}	5.77	2.96×10^{-3}
Pancreas	1.21×10^{-7}	8.03×10^{-8}	2.39	1.24×10^{-3}
Urinary	6.24×10^{-8}	4.14×10^{-8}	4.25	2.22×10^{-3}
Other	1.48×10^{-7}	9.82×10^{-8}	2.92	1.52×10^{-3}

1 Lifetime risk/100,000 pico curies

2 Lifetime risk – cubic centimeter/100,000 pico curies

Table 7.6-4
Radionuclide Activity Source Term for CAP88 Model

Nuclide	RF	ARF	AF	Ci/yr	Nuclide	RF	ARF	AF	Ci/yr
^{225}Ac	3.00×10^{-5}	4.00×10^{-5}	1.00×10^{-5}	6.44×10^{-17}	^{210}Po	3.00×10^{-5}	4.00×10^{-5}	1.00×10^{-5}	3.33×10^{-21}
^{227}Ac	3.00×10^{-5}	4.00×10^{-5}	1.00×10^{-5}	2.59×10^{-17}	^{211}Po	3.00×10^{-5}	4.00×10^{-5}	1.00×10^{-5}	7.27×10^{-20}
^{228}Ac	3.00×10^{-5}	4.00×10^{-5}	1.00×10^{-5}	9.66×10^{-18}	^{212}Po	3.00×10^{-5}	4.00×10^{-5}	1.00×10^{-5}	5.65×10^{-15}
^{241}Am	3.00×10^{-5}	4.00×10^{-5}	1.00×10^{-5}	3.43×10^{-14}	^{213}Po	3.00×10^{-5}	4.00×10^{-5}	1.00×10^{-5}	6.30×10^{-17}
^{242}Am	3.00×10^{-5}	4.00×10^{-5}	1.00×10^{-5}	2.52×10^{-17}	^{214}Po	3.00×10^{-5}	4.00×10^{-5}	1.00×10^{-5}	8.74×10^{-21}
^{241m}Am	3.00×10^{-5}	4.00×10^{-5}	1.00×10^{-5}	2.54×10^{-17}	^{215}Po	3.00×10^{-5}	4.00×10^{-5}	1.00×10^{-5}	2.60×10^{-17}
^{243}Am	3.00×10^{-5}	4.00×10^{-5}	1.00×10^{-5}	2.29×10^{-16}	^{216}Po	3.00×10^{-5}	4.00×10^{-5}	1.00×10^{-5}	8.83×10^{-15}
^{217}At	3.00×10^{-5}	4.00×10^{-5}	1.00×10^{-5}	6.44×10^{-17}	^{218}Po	3.00×10^{-5}	4.00×10^{-5}	1.00×10^{-5}	8.74×10^{-21}
^{137m}Ba	2.00×10^{-4}	4.00×10^{-5}	1.00×10^{-5}	1.01×10^{-10}	^{144}Pr	3.00×10^{-5}	4.00×10^{-5}	1.00×10^{-5}	9.41×10^{-22}
^{10}Be	3.00×10^{-5}	4.00×10^{-5}	1.00×10^{-5}	8.37×10^{-18}	^{236}Pu	3.00×10^{-5}	4.00×10^{-5}	1.00×10^{-5}	8.27×10^{-21}
^{210}Bi	3.00×10^{-5}	4.00×10^{-5}	1.00×10^{-5}	3.55×10^{-21}	^{238}Pu	3.00×10^{-5}	4.00×10^{-5}	1.00×10^{-5}	1.19×10^{-12}
^{211}Bi	3.00×10^{-5}	4.00×10^{-5}	1.00×10^{-5}	2.60×10^{-17}	^{239}Pu	3.00×10^{-5}	4.00×10^{-5}	1.00×10^{-5}	1.68×10^{-15}
^{212}Bi	3.00×10^{-5}	4.00×10^{-5}	1.00×10^{-5}	8.83×10^{-15}	^{240}Pu	3.00×10^{-5}	4.00×10^{-5}	1.00×10^{-5}	2.12×10^{-15}
^{213}Bi	3.00×10^{-5}	4.00×10^{-5}	1.00×10^{-5}	6.44×10^{-17}	^{241}Pu	3.00×10^{-5}	4.00×10^{-5}	1.00×10^{-5}	3.22×10^{-13}
^{214}Bi	3.00×10^{-5}	4.00×10^{-5}	1.00×10^{-5}	8.74×10^{-21}	^{242}Pu	3.00×10^{-5}	4.00×10^{-5}	1.00×10^{-5}	2.55×10^{-17}
^{14}C	3.00×10^{-5}	4.00×10^{-5}	1.00×10^{-5}	5.77×10^{-16}	^{223}Ra	3.00×10^{-5}	4.00×10^{-5}	1.00×10^{-5}	2.60×10^{-17}
^{113m}Cd	3.00×10^{-5}	4.00×10^{-5}	1.00×10^{-5}	1.03×10^{-15}	^{224}Ra	3.00×10^{-5}	4.00×10^{-5}	1.00×10^{-5}	8.83×10^{-15}
^{144}Ce	3.00×10^{-5}	4.00×10^{-5}	1.00×10^{-5}	9.41×10^{-22}	^{225}Ra	3.00×10^{-5}	4.00×10^{-5}	1.00×10^{-5}	6.44×10^{-17}
^{36}Cl	3.00×10^{-5}	4.00×10^{-5}	1.00×10^{-5}	1.47×10^{-17}	^{226}Ra	3.00×10^{-5}	4.00×10^{-5}	1.00×10^{-5}	8.74×10^{-21}
^{242}Cm	3.00×10^{-5}	4.00×10^{-5}	1.00×10^{-5}	2.09×10^{-17}	^{228}Ra	3.00×10^{-5}	4.00×10^{-5}	1.00×10^{-5}	9.66×10^{-18}
^{243}Cm	3.00×10^{-5}	4.00×10^{-5}	1.00×10^{-5}	1.95×10^{-16}	^{87}Rb	3.00×10^{-5}	4.00×10^{-5}	1.00×10^{-5}	9.68×10^{-21}
^{244}Cm	3.00×10^{-5}	4.00×10^{-5}	1.00×10^{-5}	2.13×10^{-14}	^{106}Rh	3.00×10^{-5}	4.00×10^{-5}	1.00×10^{-5}	3.21×10^{-20}
^{245}Cm	3.00×10^{-5}	4.00×10^{-5}	1.00×10^{-5}	8.02×10^{-18}	^{219}Rn	3.00×10^{-1}	0	0	6.49×10^{-4}
^{246}Cm	3.00×10^{-5}	4.00×10^{-5}	1.00×10^{-5}	1.17×10^{-18}	^{220}Rn	3.00×10^{-1}	0	0	2.21×10^{-1}
^{60}Co	3.00×10^{-5}	4.00×10^{-5}	1.00×10^{-5}	1.86×10^{-14}	^{222}Rn	3.00×10^{-1}	0	0	2.18×10^{-7}
^{134}Cs	2.00×10^{-4}	4.00×10^{-5}	1.00×10^{-5}	1.67×10^{-14}	^{106}Ru	2.00×10^{-4}	4.00×10^{-5}	1.00×10^{-5}	2.14×10^{-19}
^{135}Cs	2.00×10^{-4}	4.00×10^{-5}	1.00×10^{-5}	1.45×10^{15}	^{125}Sb	3.00×10^{-5}	4.00×10^{-5}	1.00×10^{-5}	1.25×10^{-15}
^{137}Cs	2.00×10^{-4}	4.00×10^{-5}	1.00×10^{-5}	1.07×10^{-10}	^{126}Sb	3.00×10^{-5}	4.00×10^{-5}	1.00×10^{-5}	2.32×10^{-17}
^{152}x10u	3.00×10^{-5}	4.00×10^{-5}	1.00×10^{-5}	1.50×10^{-15}	^{126m}Sb	3.00×10^{-5}	4.00×10^{-5}	1.00×10^{-5}	1.66×10^{-16}
^{154}x10u	3.00×10^{-5}	4.00×10^{-5}	1.00×10^{-5}	2.07×10^{-13}	^{79}Sei	3.00×10^{-5}	4.00×10^{-5}	1.00×10^{-5}	1.68×10^{-16}
^{155}x10u	3.00×10^{-5}	4.00×10^{-5}	1.00×10^{-5}	2.05×10^{-14}	^{147}Sm	3.00×10^{-5}	4.00×10^{-5}	1.00×10^{-5}	1.43×10^{-21}
^{55}Fe	3.00×10^{-5}	4.00×10^{-5}	1.00×10^{-5}	3.46×10^{-17}	^{151}Sm	3.00×10^{-5}	4.00×10^{-5}	1.00×10^{-5}	1.21×10^{-13}
^{221}Fr	3.00×10^{-5}	4.00×10^{-5}	1.00×10^{-5}	6.44×10^{-17}	^{121m}Sn	3.00×10^{-5}	4.00×10^{-5}	1.00×10^{-5}	3.11×10^{-17}
^{223}Fr	3.00×10^{-5}	4.00×10^{-5}	1.00×10^{-5}	3.58×10^{-19}	^{126}Sn	3.00×10^{-5}	4.00×10^{-5}	1.00×10^{-5}	1.66×10^{-16}
^{3}H	3.00×10^{-1}	0	0	2.10×10^{0}	^{90}Sr	2.00×10^{-4}	4.00×10^{-5}	1.00×10^{-5}	1.01×10^{-10}

Table 7.6-4 (continued)
Radionuclide Activity Source Term for CAP88 Model

Nuclide	RF	ARF	AF	Ci/yr	Nuclide	RF	ARF	AF	Ci/yr
^{129}I	$3.00x10^{-1}$	0	0	$2.16x10^{-4}$	^{99}Tc	$3.00x10^{-5}$	$4.00x10^{-5}$	$1.00x10^{-5}$	$3.84x10^{-15}$
^{85}Kr	$3.00x10^{-1}$	0	0	$1.47x10^{1}$	^{125m}Te	$3.00x10^{-5}$	$4.00x10^{-5}$	$1.00x10^{-5}$	$3.05x10^{-16}$
^{93}Mo	$3.00x10^{-5}$	$4.00x10^{-5}$	$1.00x10^{-5}$	$2.27x10^{-19}$	^{227}Th	$3.00x10^{-5}$	$4.00x10^{-5}$	$1.00x10^{-5}$	$2.56x10^{-17}$
^{93m}Nb	$3.00x10^{-5}$	$4.00x10^{-5}$	$1.00x10^{-5}$	$5.18x10^{-16}$	^{228}Th	$3.00x10^{-5}$	$4.00x10^{-5}$	$1.00x10^{-5}$	$8.79x10^{-15}$
^{94}Nb	$3.00x10^{-5}$	$4.00x10^{-5}$	$1.00x10^{-5}$	$6.81x10^{-18}$	^{229}Th	$3.00x10^{-5}$	$4.00x10^{-5}$	$1.00x10^{-5}$	$6.44x10^{-17}$
^{59}Ni	$3.00x10^{-5}$	$4.00x10^{-5}$	$1.00x10^{-5}$	$1.93x10^{-17}$	^{230}Th	$3.00x10^{-5}$	$4.00x10^{-5}$	$1.00x10^{-5}$	$1.02x10^{-18}$
^{63}Ni	$3.00x10^{-5}$	$4.00x10^{-5}$	$1.00x10^{-5}$	$2.04x10^{-15}$	^{231}Th	$3.00x10^{-5}$	$4.00x10^{-5}$	$1.00x10^{-5}$	$1.13x10^{-17}$
^{237}Np	$3.00x10^{-5}$	$4.00x10^{-5}$	$1.00x10^{-5}$	$1.44x10^{-16}$	^{232}Th	$3.00x10^{-5}$	$4.00x10^{-5}$	$1.00x10^{-5}$	$9.96x10^{-18}$
^{238}Np	$3.00x10^{-5}$	$4.00x10^{-5}$	$1.00x10^{-5}$	$1.27x10^{-19}$	^{234}Th	$3.00x10^{-5}$	$4.00x10^{-5}$	$1.00x10^{-5}$	$2.33x10^{-19}$
^{239}Np	$3.00x10^{-5}$	$4.00x10^{-5}$	$1.00x10^{-5}$	$2.29x10^{-16}$	^{207}Tl	$3.00x10^{-5}$	$4.00x10^{-5}$	$1.00x10^{-5}$	$2.59x10^{-17}$
^{231}Pa	$3.00x10^{-5}$	$4.00x10^{-5}$	$1.00x10^{-5}$	$4.10x10^{-17}$	^{208}Tl	$3.00x10^{-5}$	$4.00x10^{-5}$	$1.00x10^{-5}$	$3.17x10^{-15}$
^{233}Pa	$3.00x10^{-5}$	$4.00x10^{-5}$	$1.00x10^{-5}$	$1.44x10^{-16}$	^{209}Tl	$3.00x10^{-5}$	$4.00x10^{-5}$	$1.00x10^{-5}$	$1.39x10^{-18}$
^{234}Pa	$3.00x10^{-5}$	$4.00x10^{-5}$	$1.00x10^{-5}$	$3.03x10^{-22}$	^{232}U	$3.00x10^{-5}$	$4.00x10^{-5}$	$1.00x10^{-5}$	$8.31x10^{-15}$
^{234m}Pa	$3.00x10^{-5}$	$4.00x10^{-5}$	$1.00x10^{-5}$	$2.33x10^{-19}$	^{233}U	$3.00x10^{-5}$	$4.00x10^{-5}$	$1.00x10^{-5}$	$2.17x10^{-14}$
^{209}Pb	$3.00x10^{-5}$	$4.00x10^{-5}$	$1.00x10^{-5}$	$6.44x10^{-17}$	^{234}U	$3.00x10^{-5}$	$4.00x10^{-5}$	$1.00x10^{-5}$	$2.69x10^{-15}$
^{210}Pb	$3.00x10^{-5}$	$4.00x10^{-5}$	$1.00x10^{-5}$	$3.54x10^{-21}$	^{235}U	$3.00x10^{-5}$	$4.00x10^{-5}$	$1.00x10^{-5}$	$1.13x10^{-17}$
^{211}Pb	$3.00x10^{-5}$	$4.00x10^{-5}$	$1.00x10^{-5}$	$2.60x10^{-17}$	^{236}U	$3.00x10^{-5}$	$4.00x10^{-5}$	$1.00x10^{-5}$	$1.31x10^{-16}$
^{212}Pb	$3.00x10^{-5}$	$4.00x10^{-5}$	$1.00x10^{-5}$	$8.83x10^{-15}$	^{237}U	$3.00x10^{-5}$	$4.00x10^{-5}$	$1.00x10^{-5}$	$7.91x10^{-18}$
^{214}Pb	$3.00x10^{-5}$	$4.00x10^{-5}$	$1.00x10^{-5}$	$8.74x10^{-21}$	^{238}U	$3.00x10^{-5}$	$4.00x10^{-5}$	$1.00x10^{-5}$	$2.33x10^{-19}$
^{107}Pd	$3.00x10^{-5}$	$4.00x10^{-5}$	$1.00x10^{-5}$	$5.27x10^{-18}$	^{90}Y	$2.00x10^{-4}$	$4.00x10^{-5}$	$1.00x10^{-5}$	$1.01x10^{-10}$
^{145}Pm	$3.00x10^{-5}$	$4.00x10^{-5}$	$1.00x10^{-5}$	$2.47x10^{-18}$	^{93}Zr	$3.00x10^{-5}$	$4.00x10^{-5}$	$1.00x10^{-5}$	$6.80x10^{-16}$
^{147}Pm	$3.00x10^{-5}$	$4.00x10^{-5}$	$1.00x10^{-5}$	$1.34x10^{-14}$					

RF ISG 5 release fraction

ARF ANSI/ANS 5.10 airborne release fraction

AF 40 CFR 61 Appendix D abatement factor (1 – prefilter and 2 – HEPA filters)

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Figure 7.3-1
Facility Radiation Area Boundaries First Floor

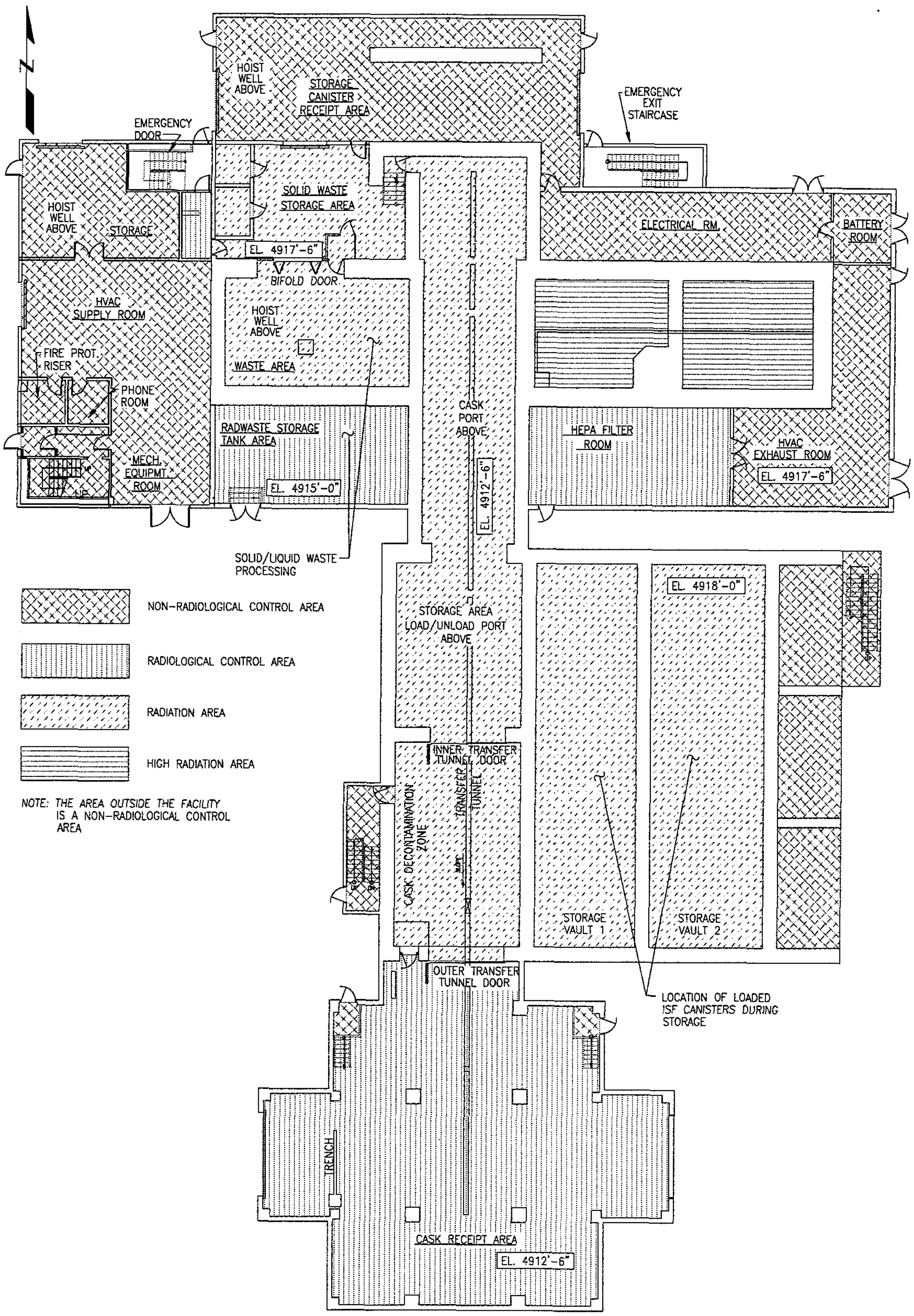

Figure 7.3-2
Facility Radiation Area Boundaries Second Floor

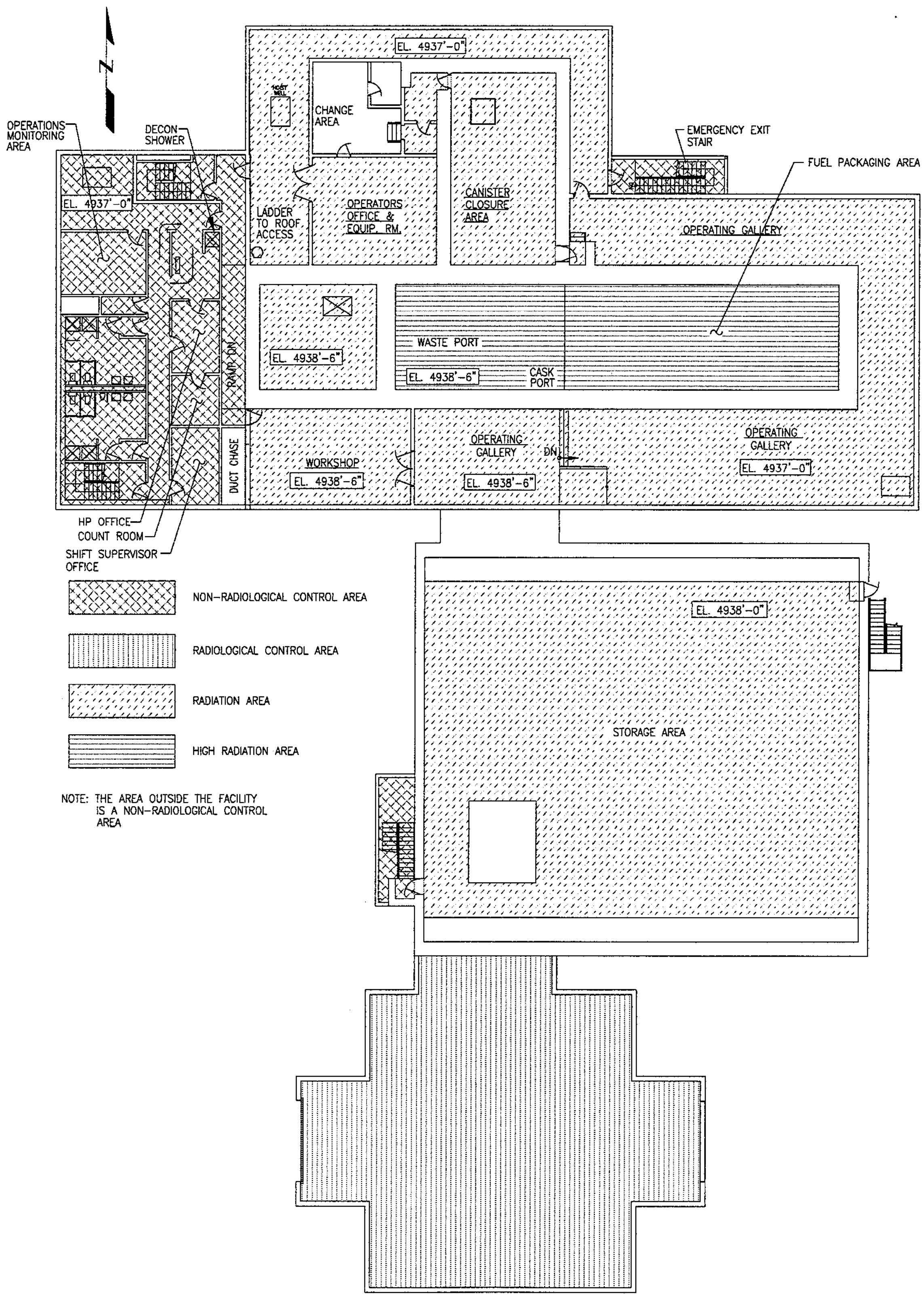

Figure 7.3-3
Monitoring Equipment Locations – Cask Receipt Area, First Floor

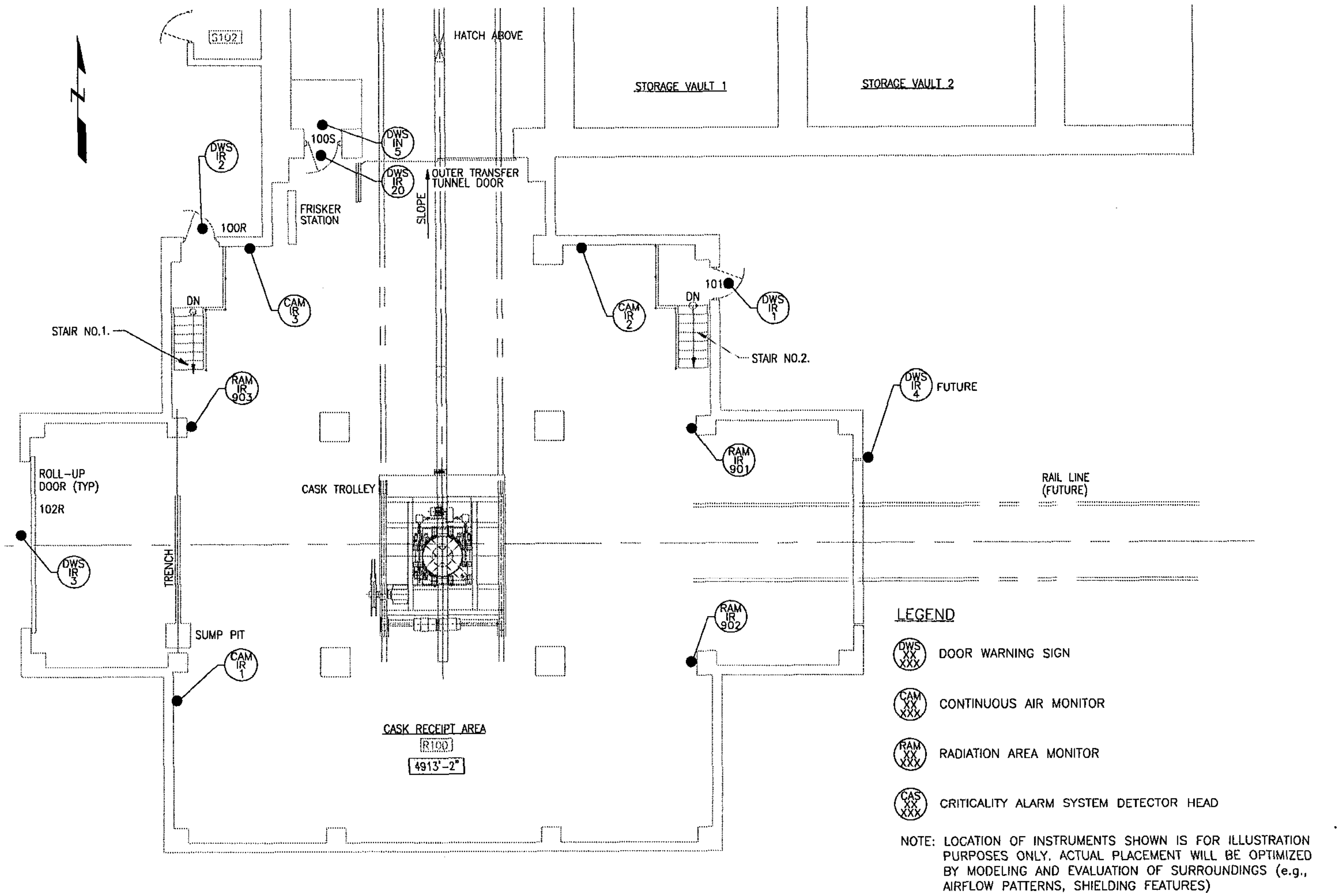

Figure 7.3-4
Monitoring Equipment Locations – Storage Area, First Floor

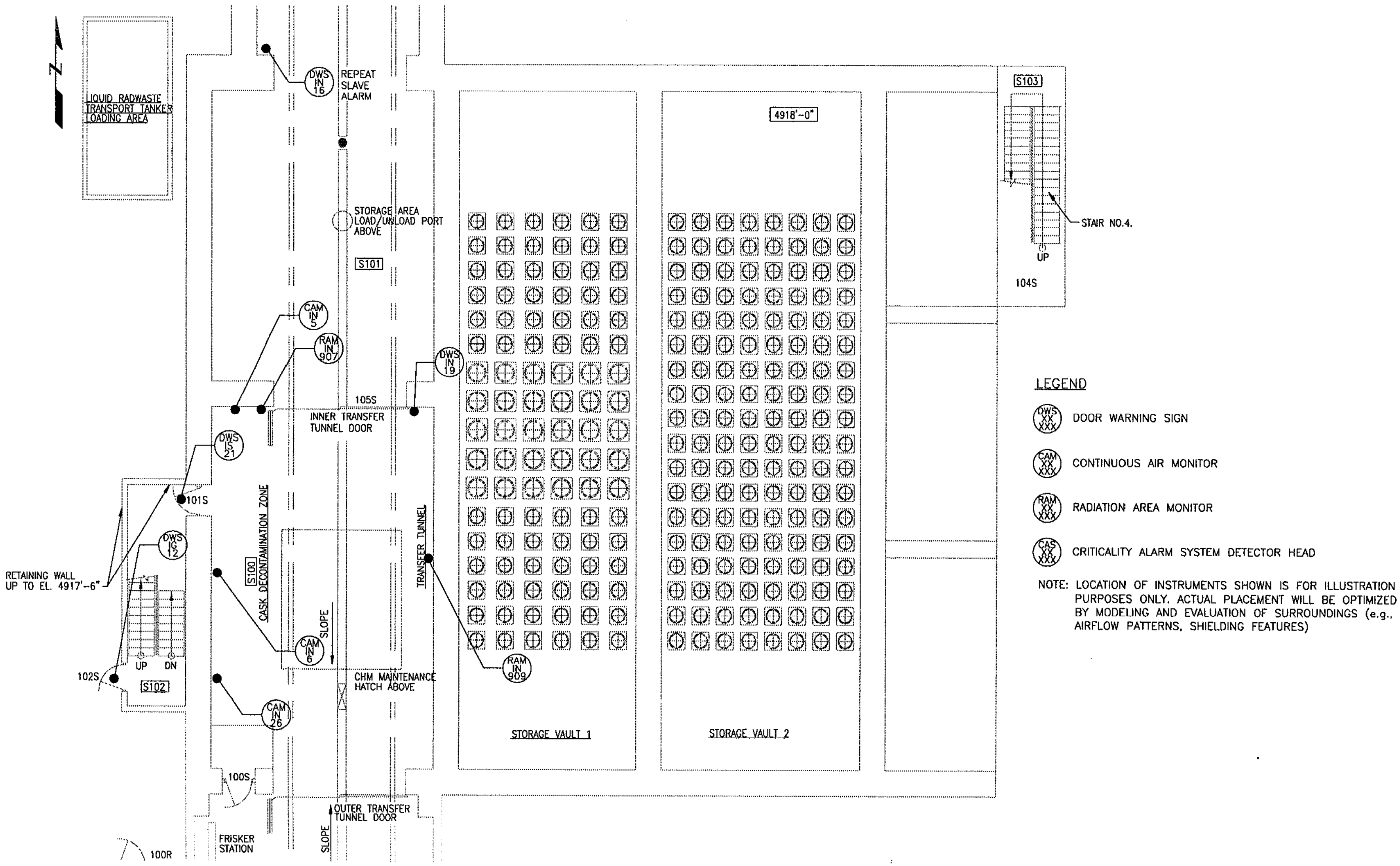

Figure 7.3-5
Monitoring Equipment Locations – Transfer Area, First Floor

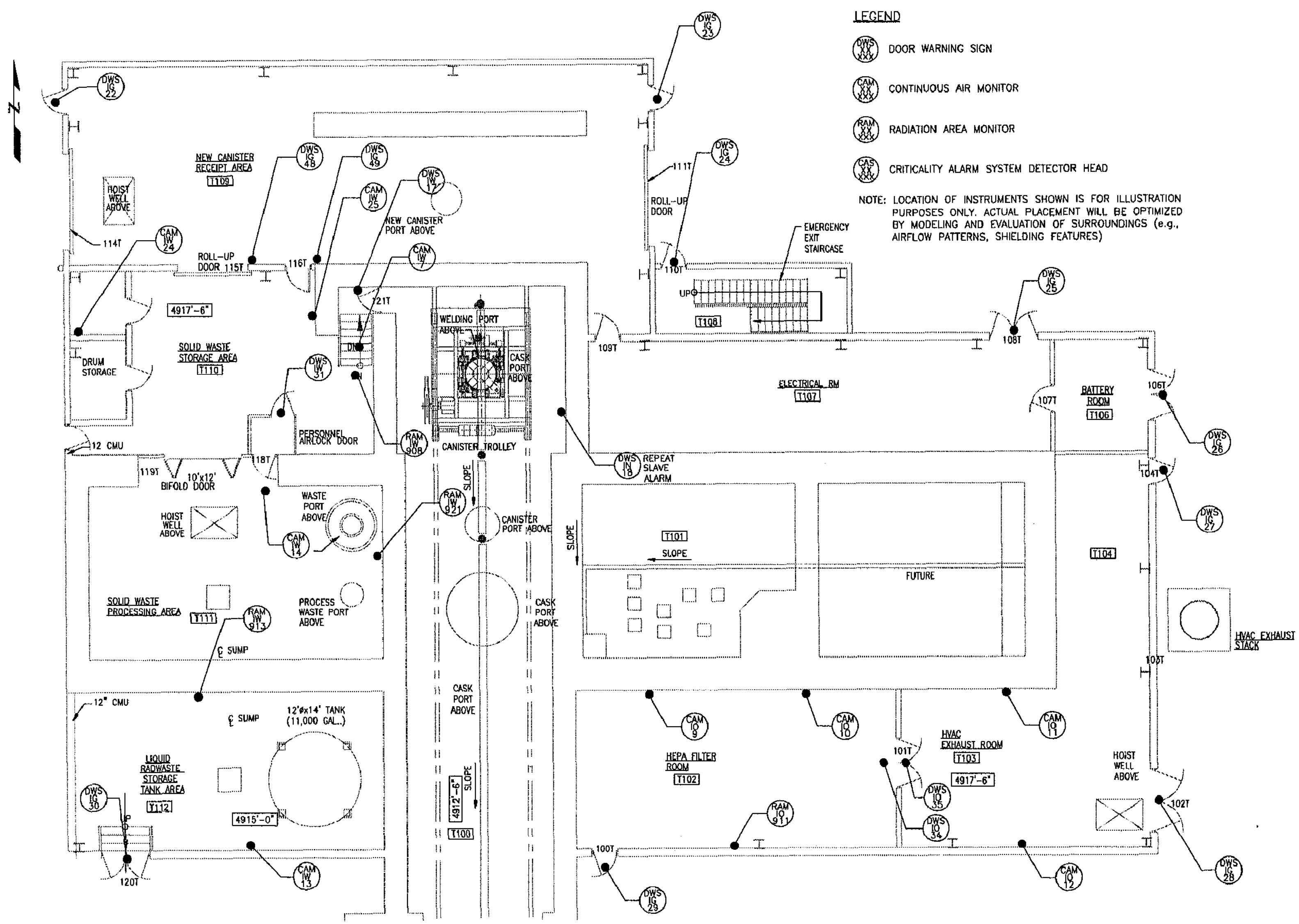

Figure 7.3-6
Monitoring Equipment Locations – Storage Area, Second Floor

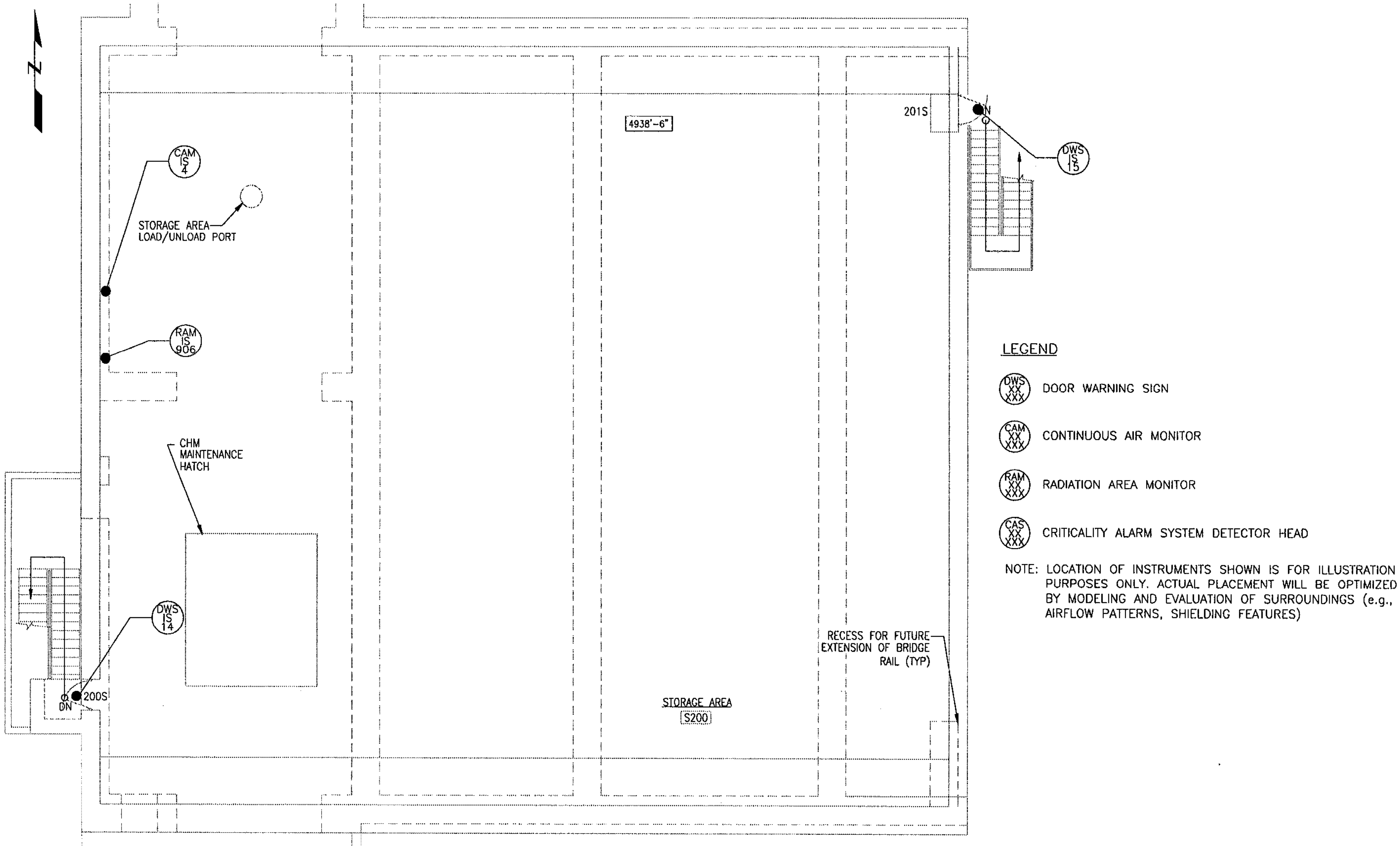

Figure 7.3-7
Monitoring Equipment Locations – Transfer Area, Second Floor

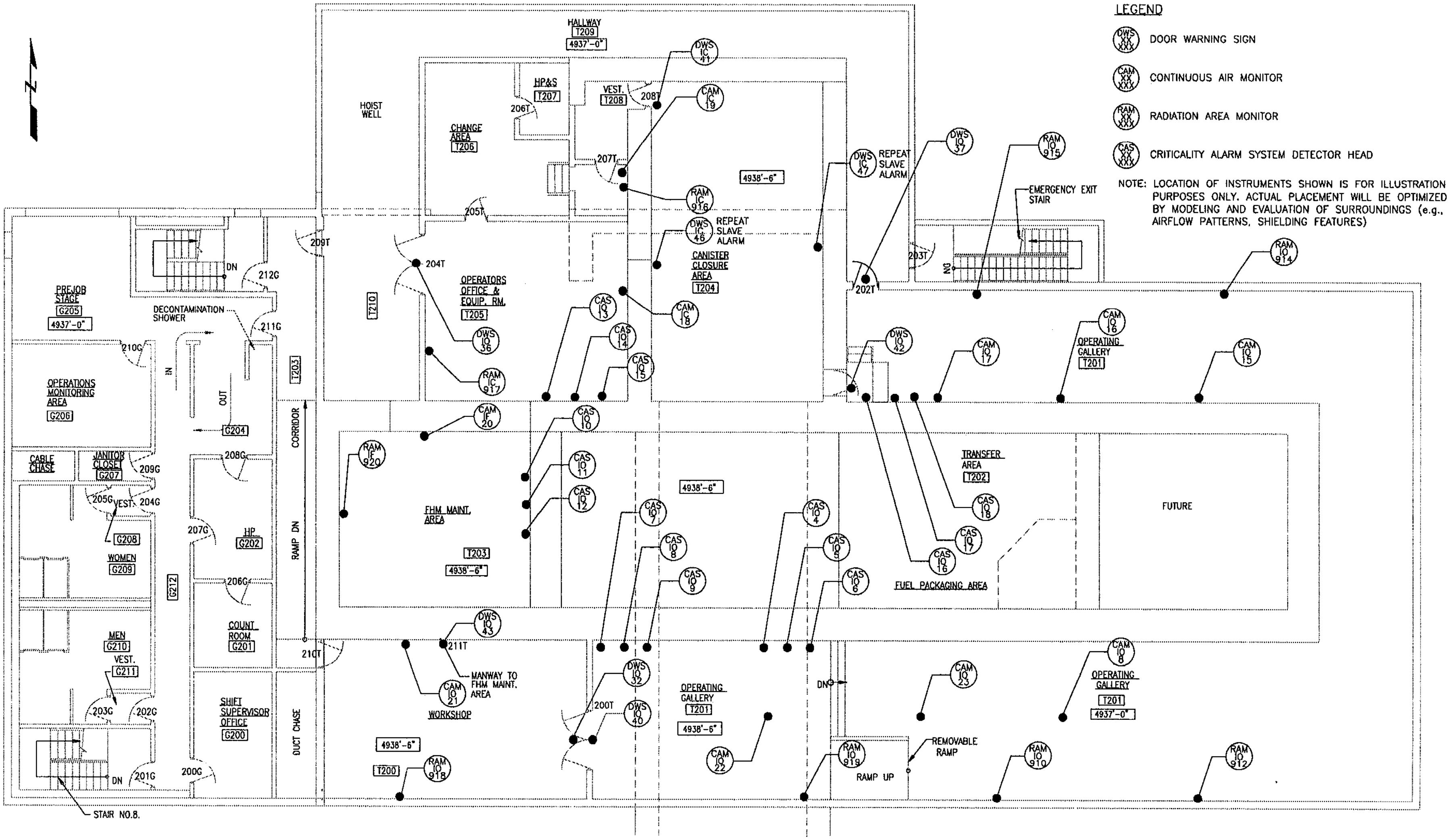

Figure 7.3-8
Monitoring Equipment Locations – Operations Area, First and Second Floor

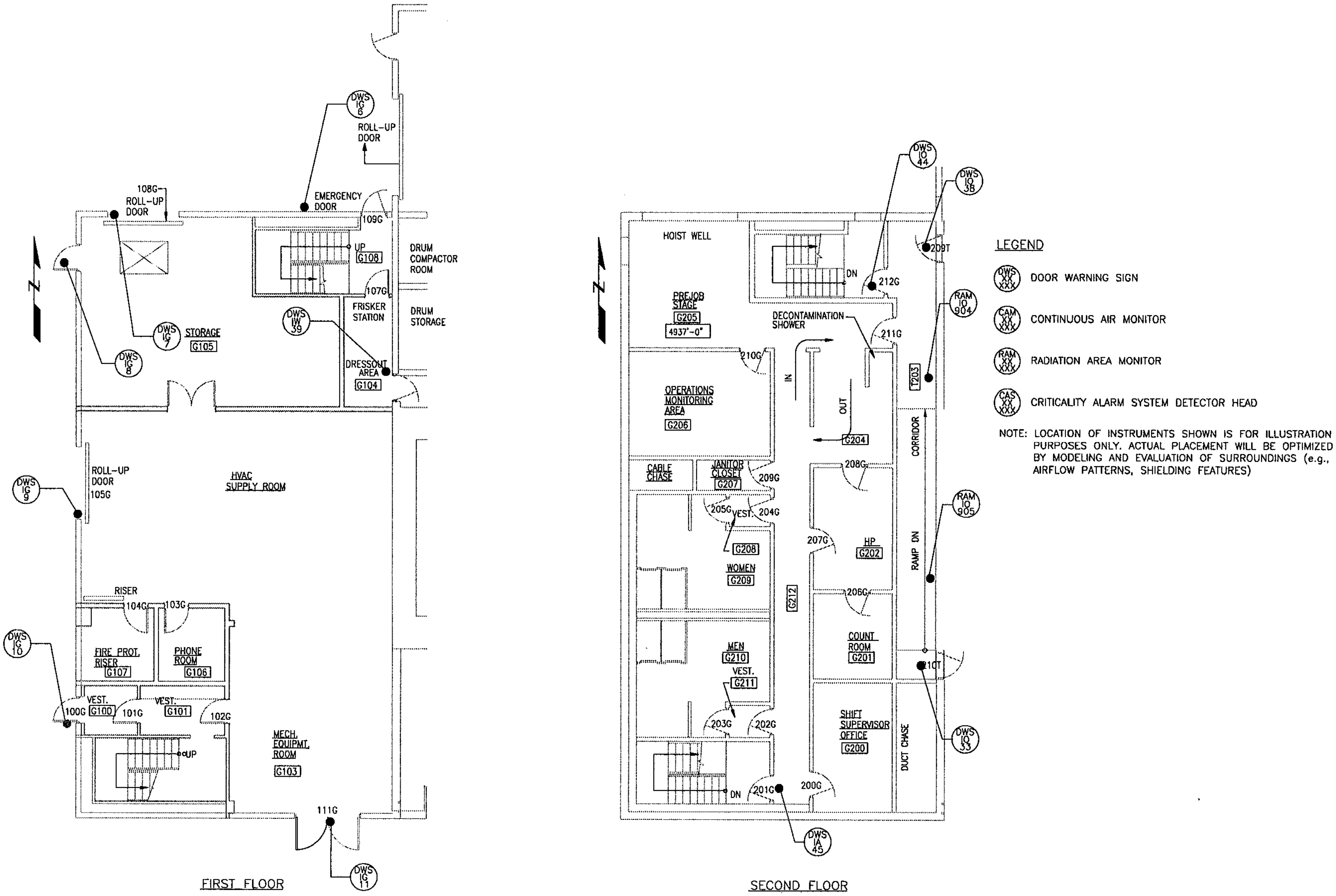

Figure 7.3-9
Storage Vault Shield Wall Dimensions

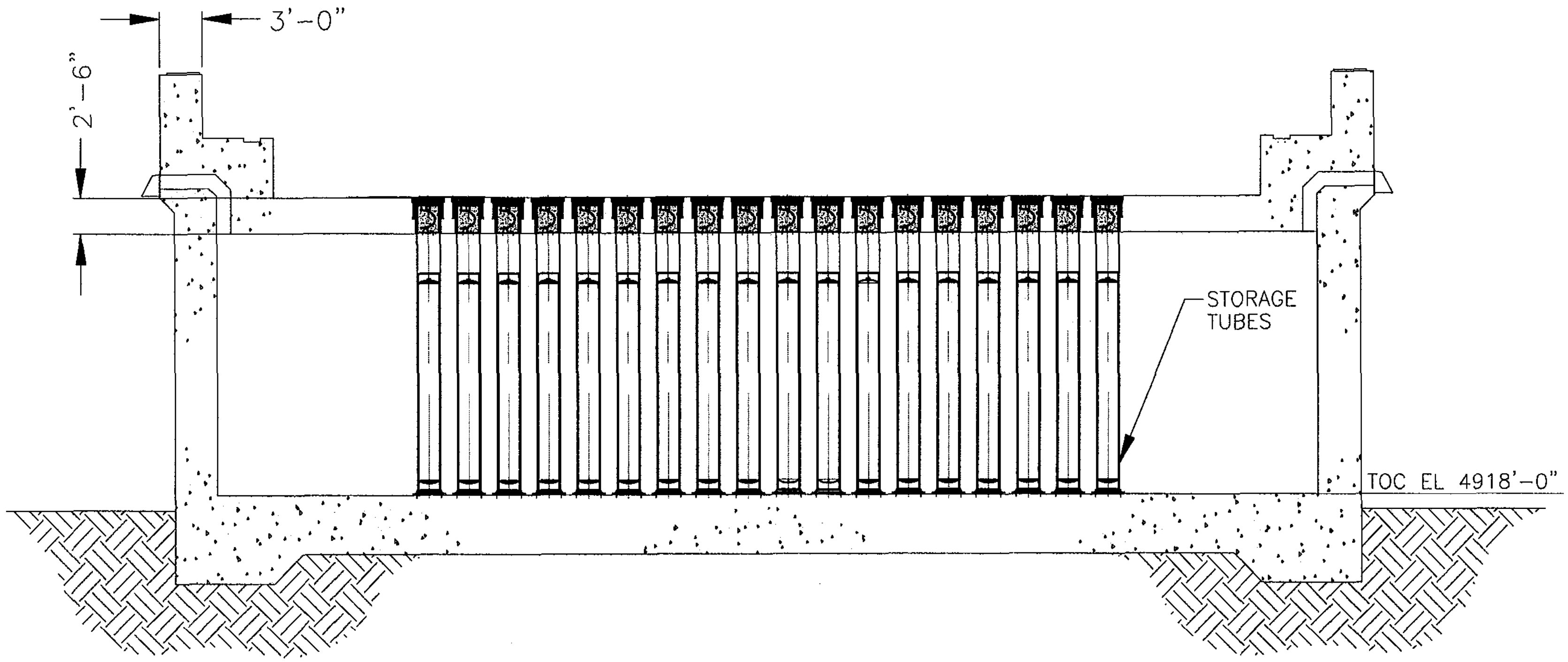

Figure 7.3-10
Transfer Area Shield Wall Dimensions

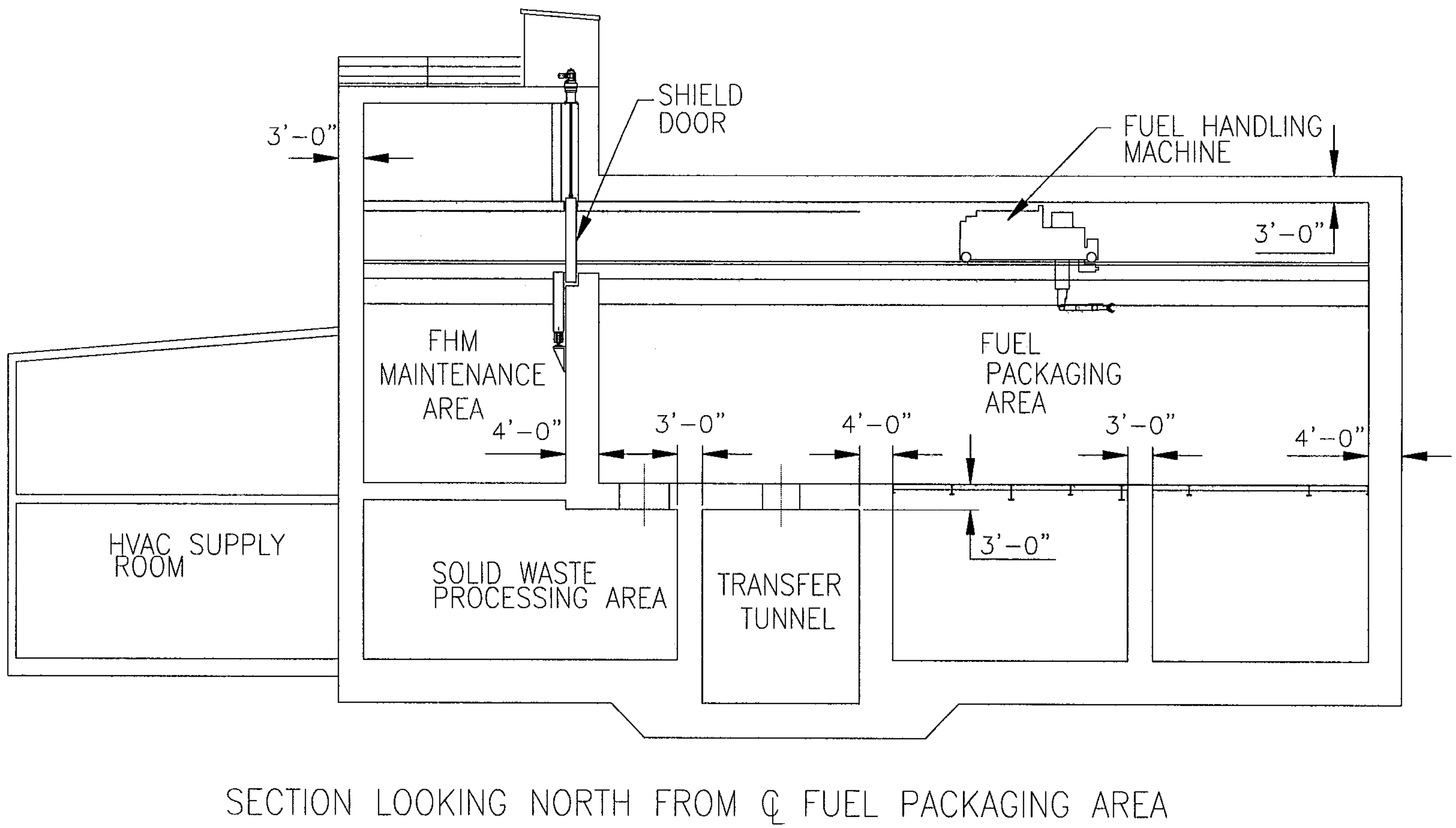

Figure 7.3-11
Radial Model of TRIGA Fuel in a Peach Bottom Cask

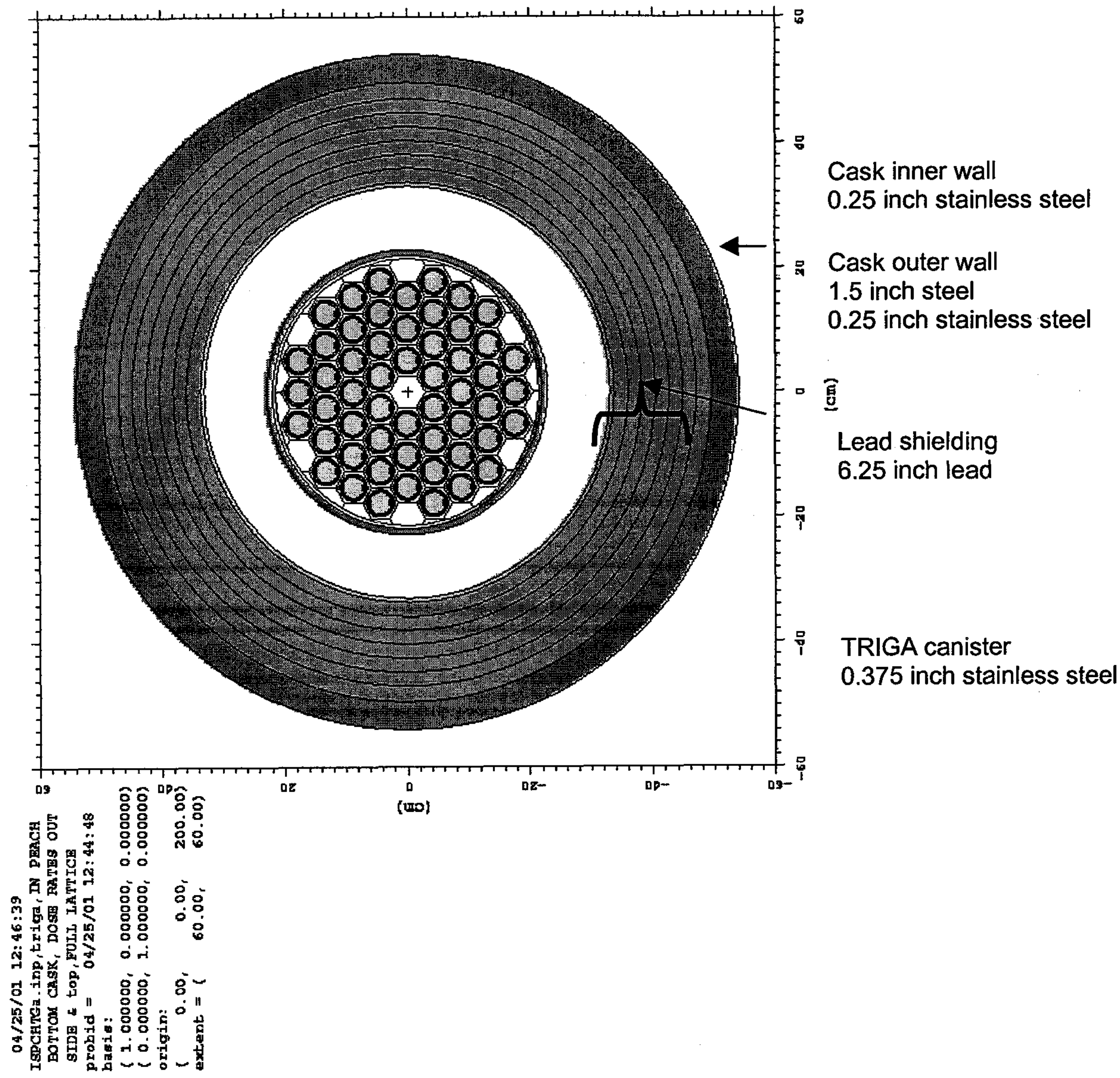

Figure 7.3-12
Axial Model of TRIGA Fuel in a Peach Bottom Cask

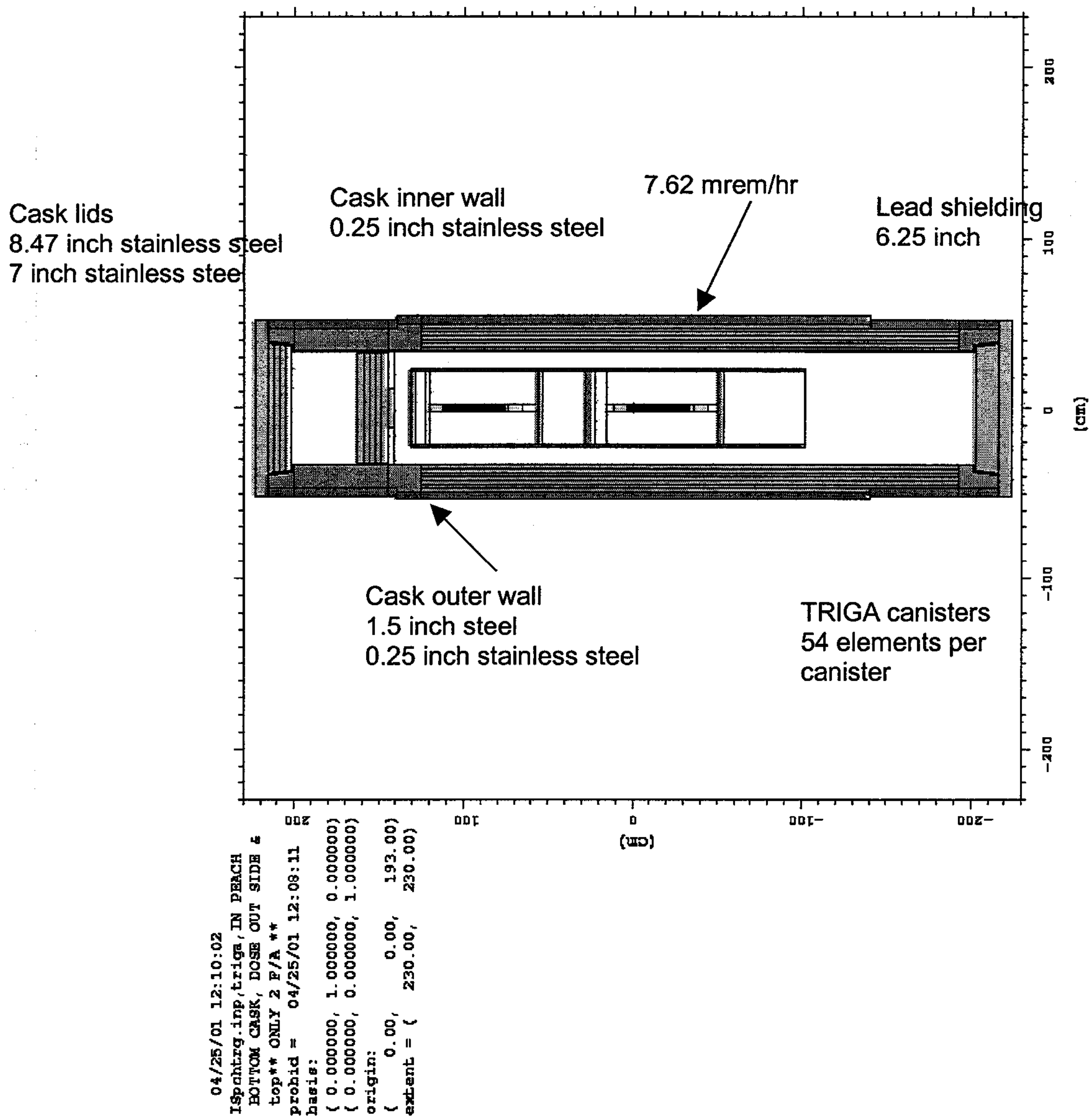

Figure 7.3-13
Radial Model of TRIGA Fuel Canister in FPA

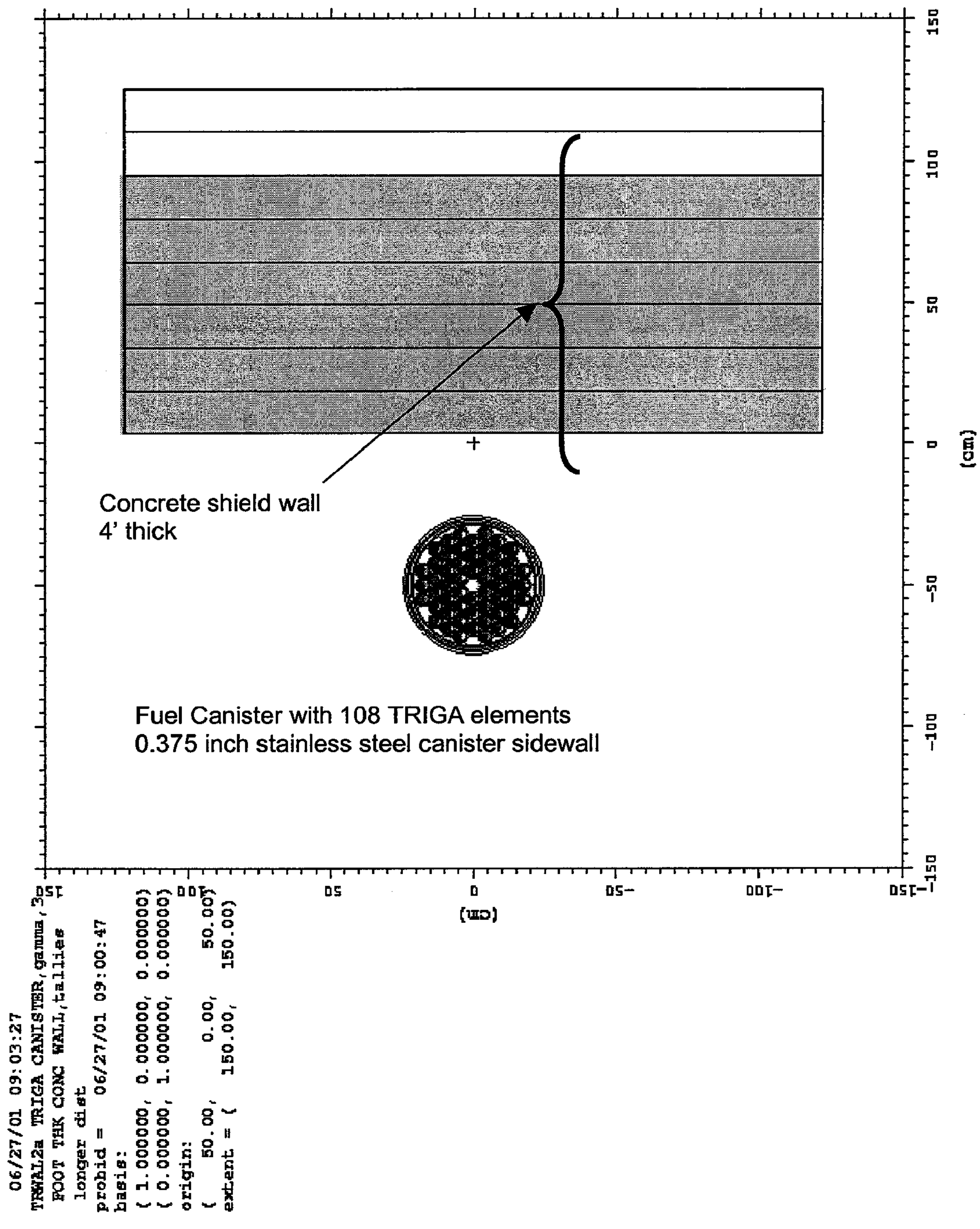

Figure 7.3-14
Axial Model of TRIGA Fuel in FPA

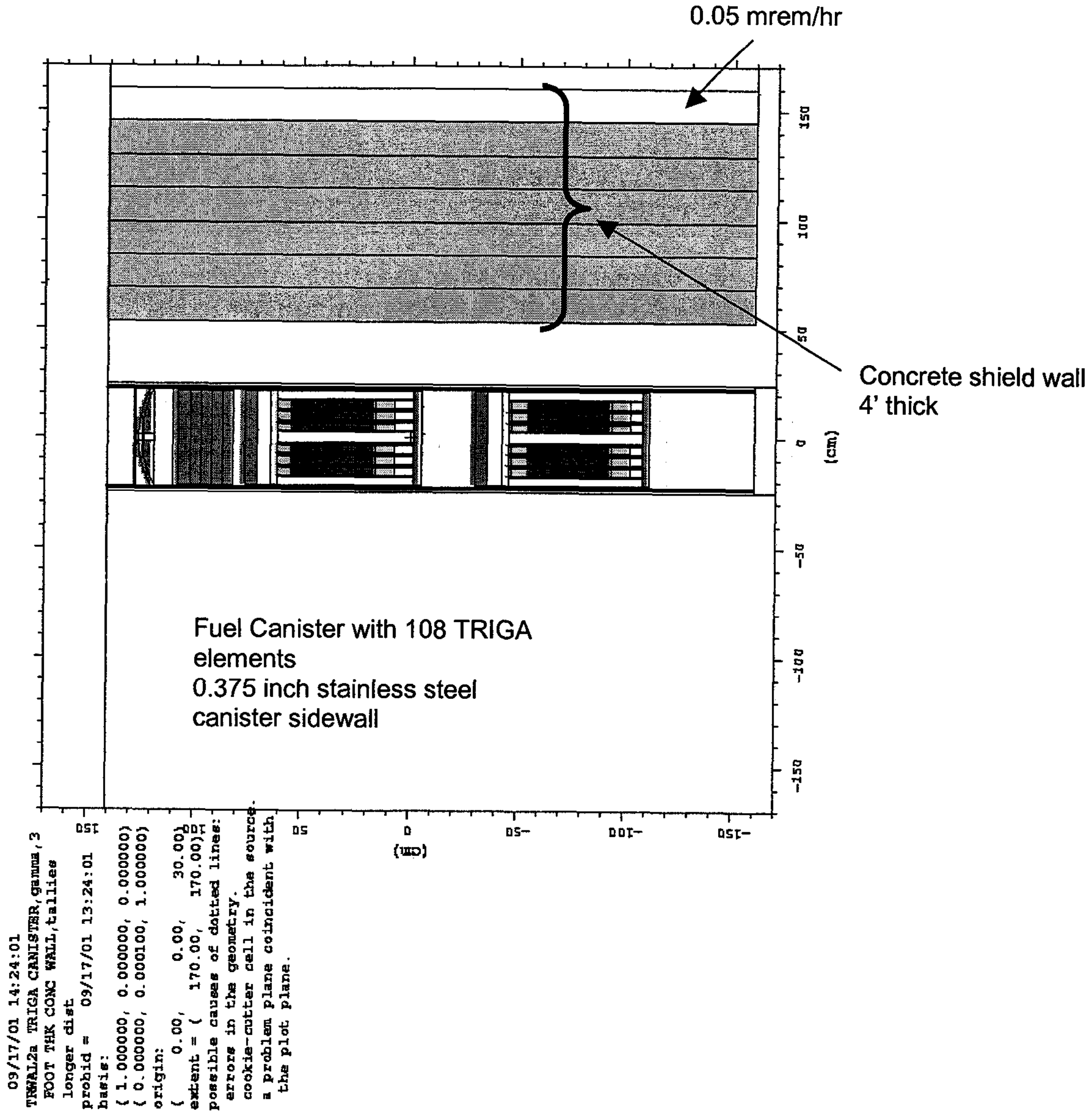

Figure 7.3-15
Canister Closure Area, Canister Funnel Layout

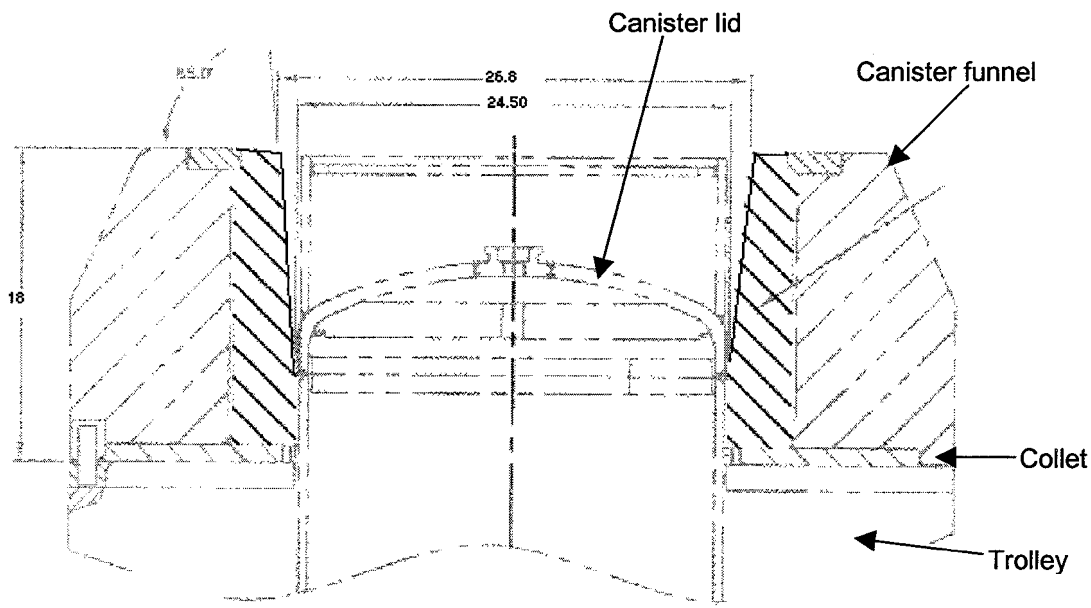

Figure 7.3-16
Model of Top Portion of the Storage Canister

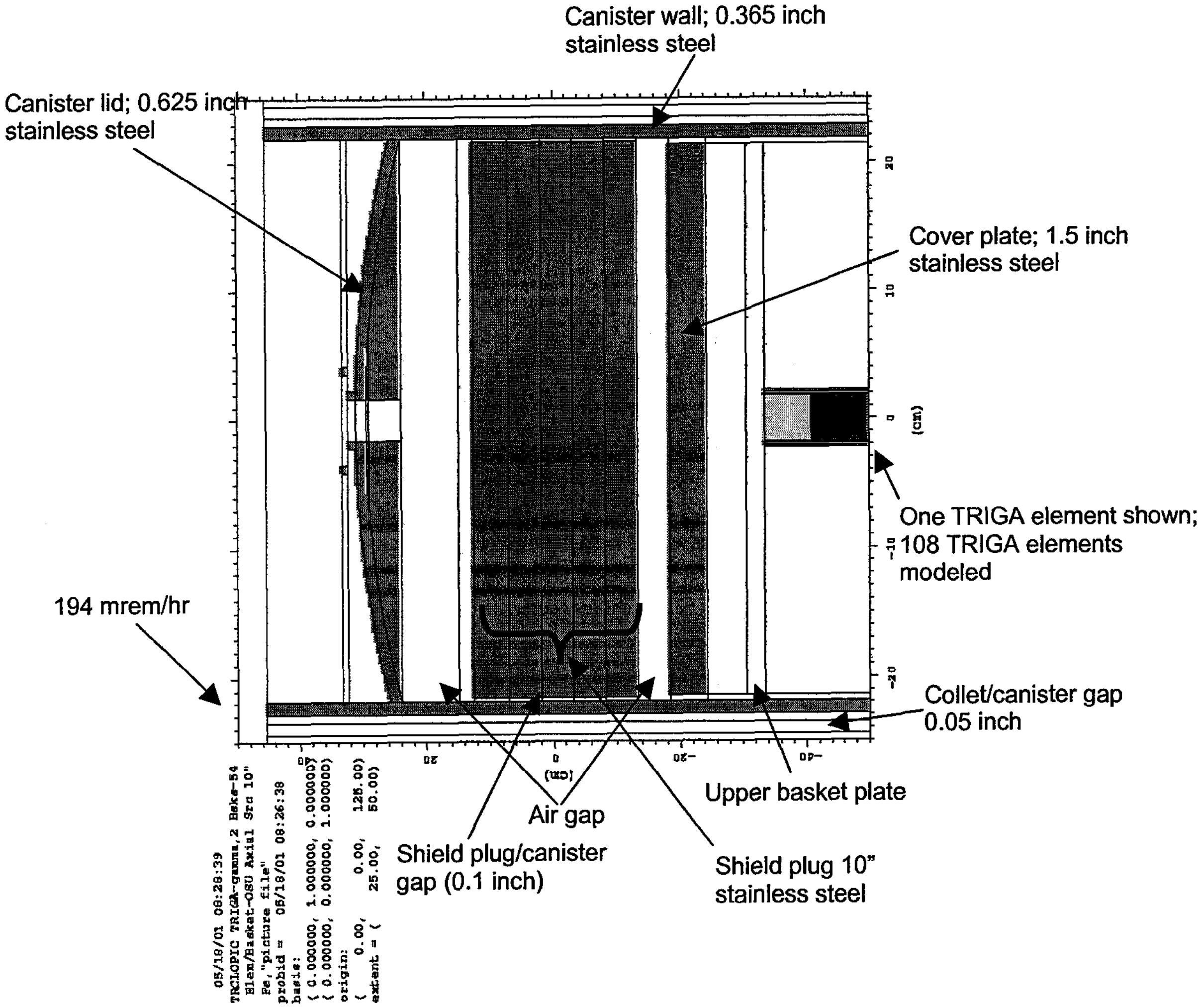

Figure 7.3-17
Model of Storage Canister Containing Two TRIGA Canisters

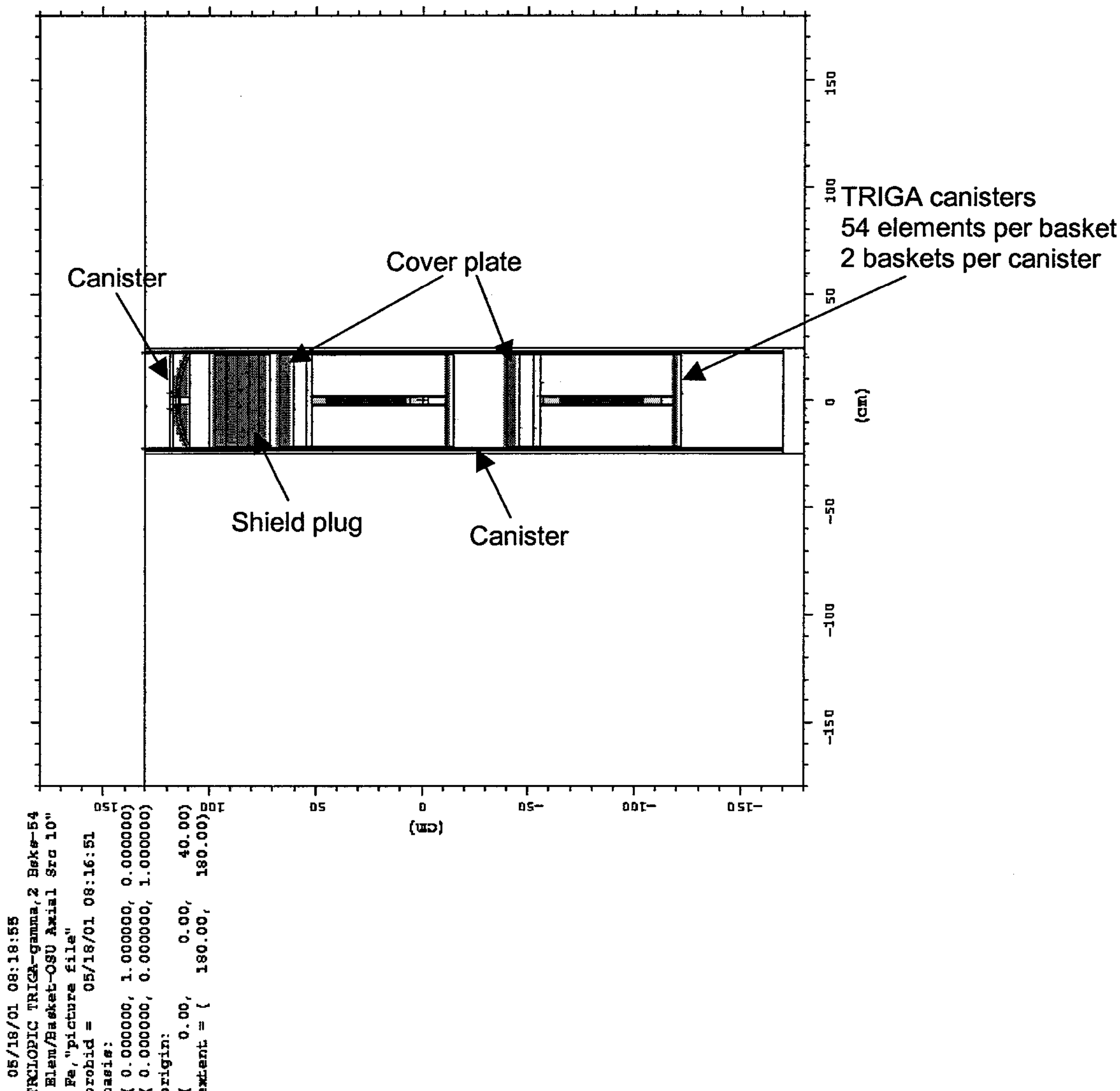

Figure 7.3-18
Radial Model of TRIGA Fuel in Shielded Canister Trolley

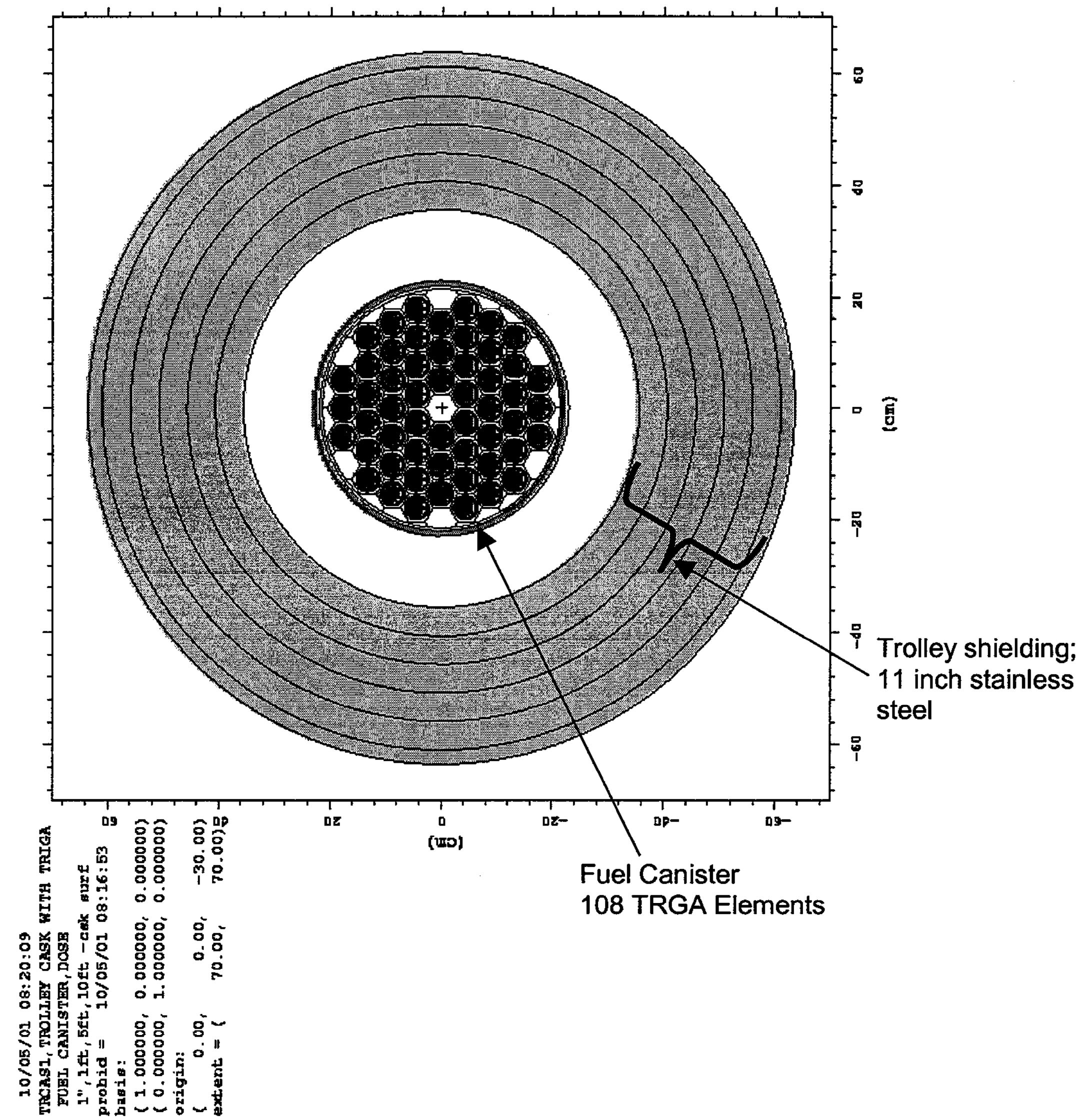

Figure 7.3-19
Axial Model of TRIGA Fuel in Shielded Canister Trolley

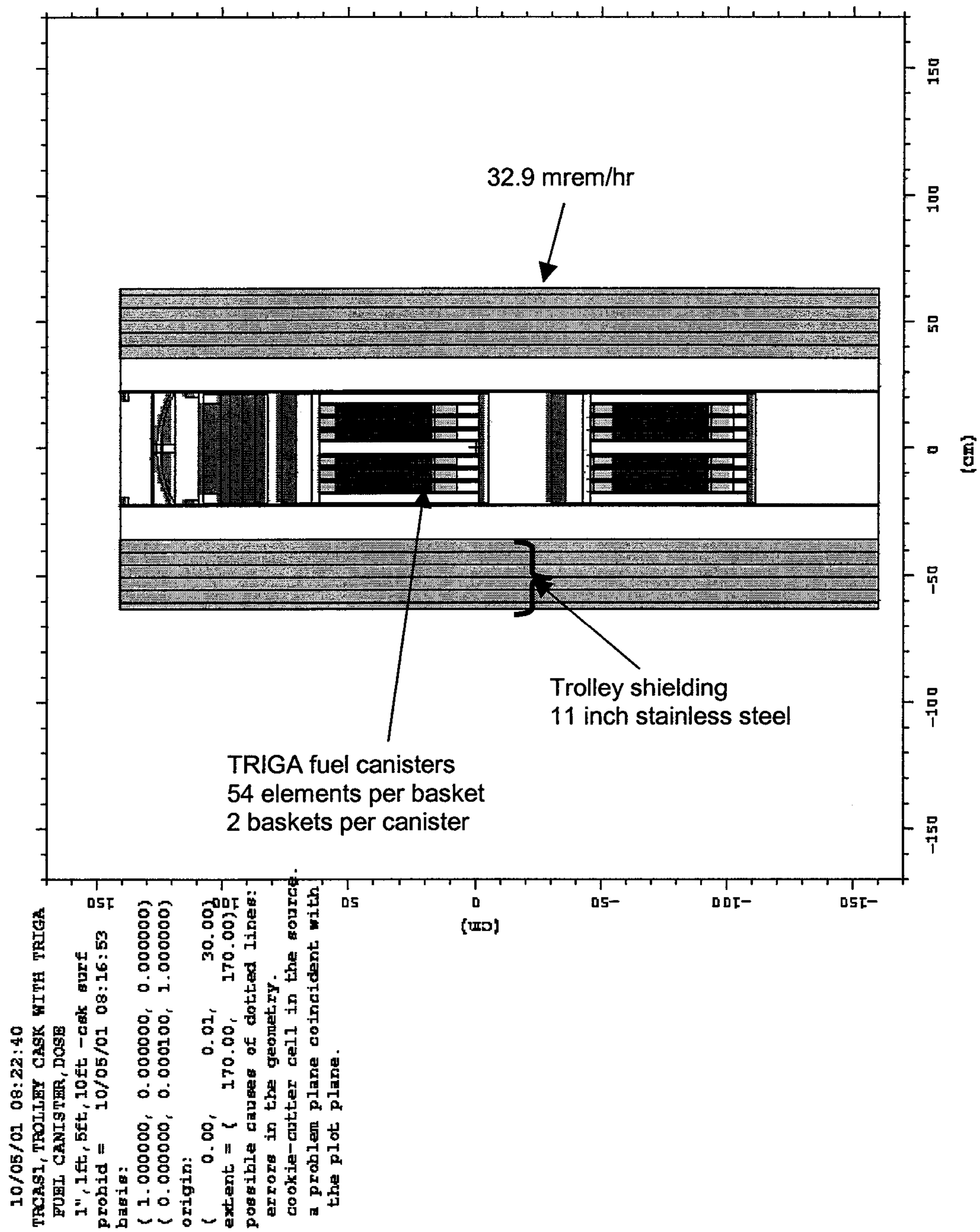

Figure 7.3-20
Storage Area with 5 x 5 Array of Tubes

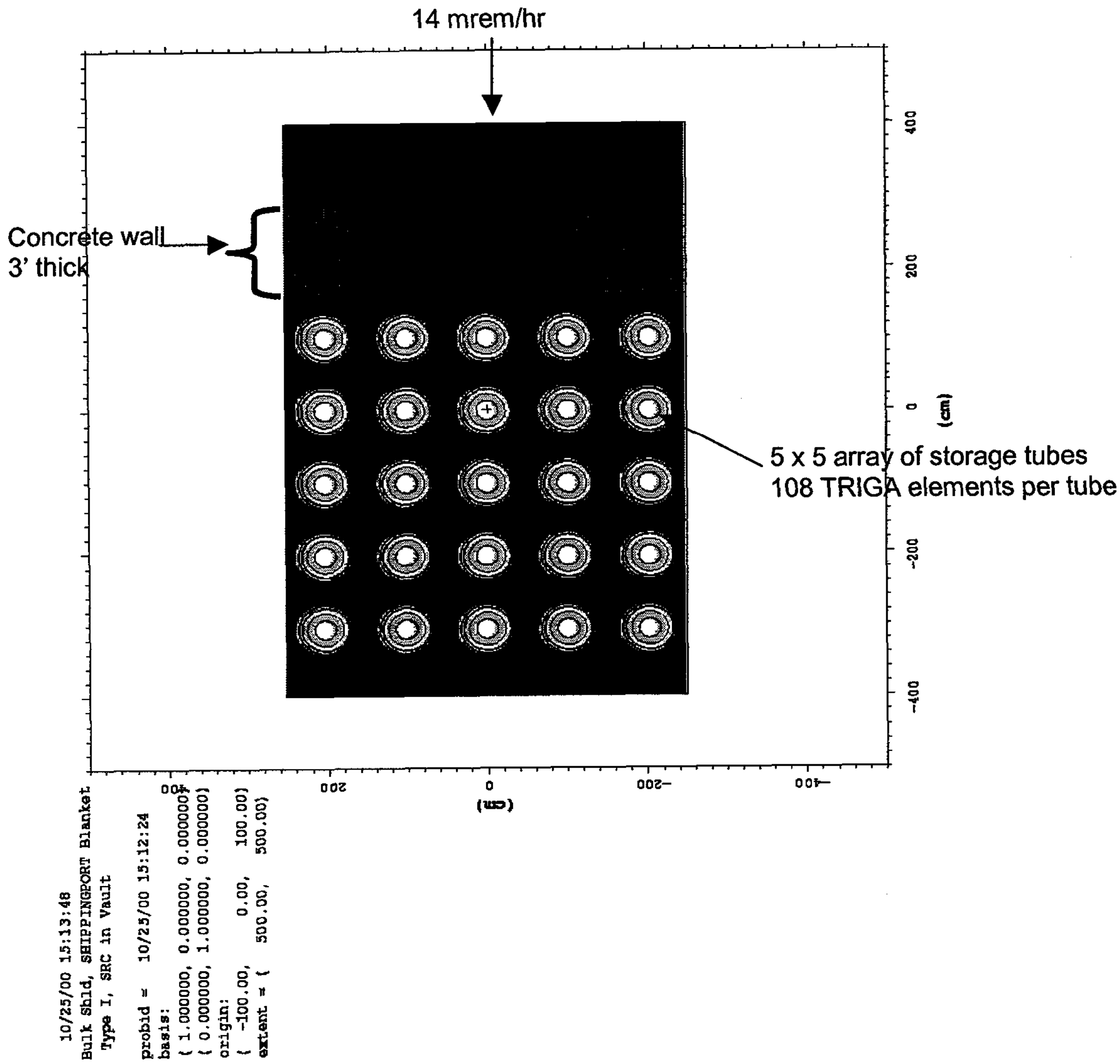

Figure 7.3-21
Canister Handling Machine Axial View

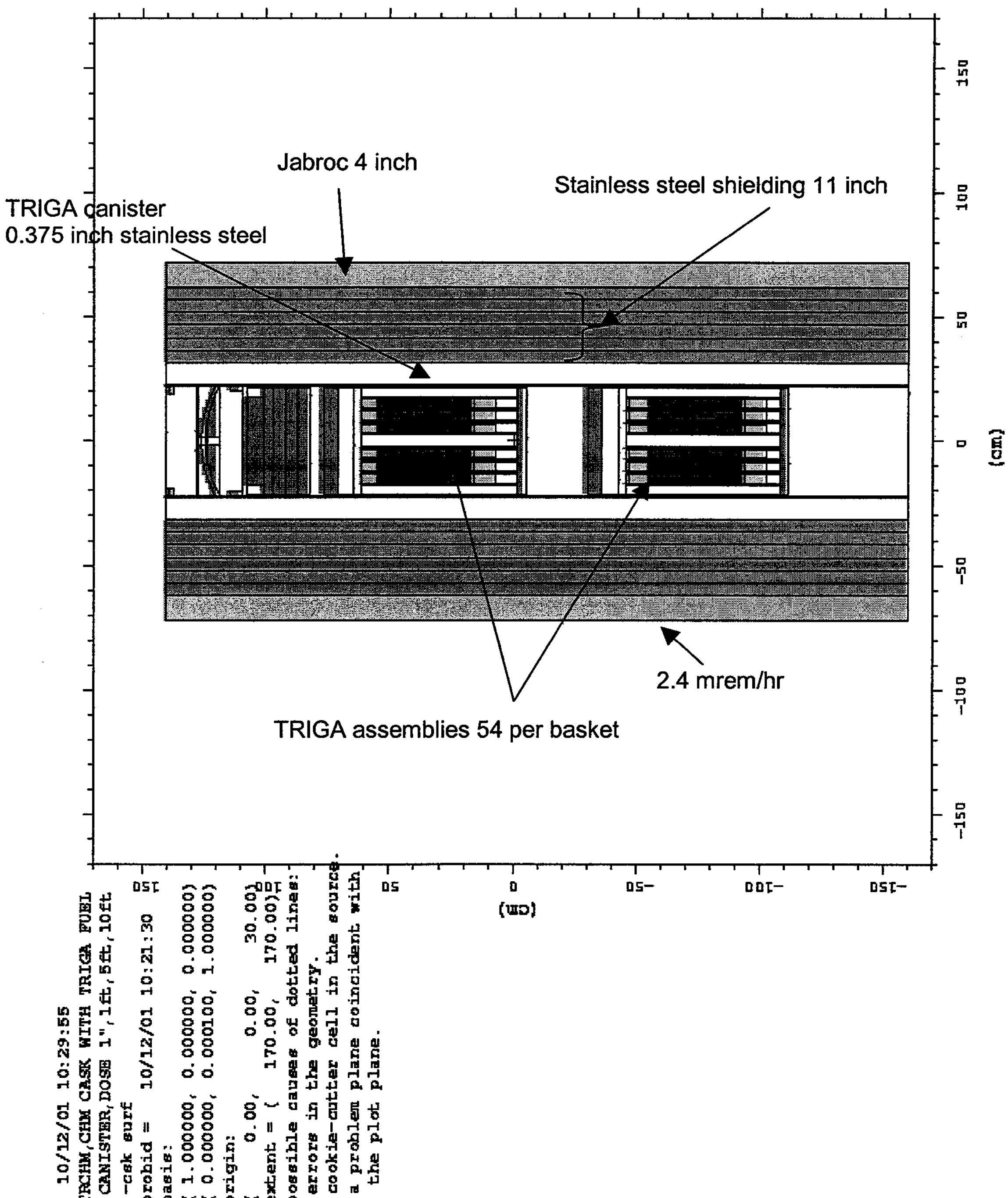

Figure 7.3-22
Canister Handling Machine Radial View

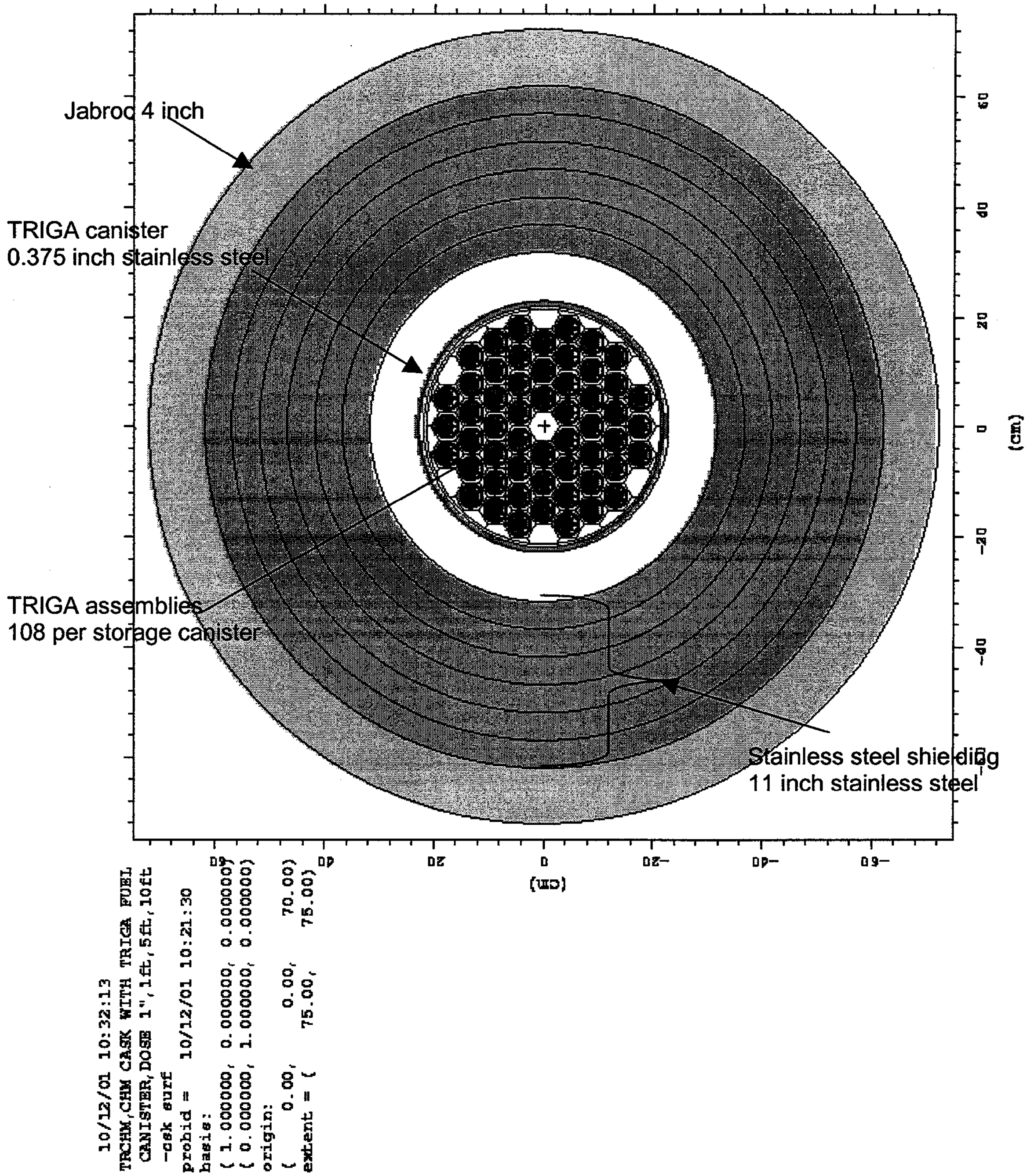

Figure 7.6-1
Pre-Operational Environmental Radiation Monitoring Locations

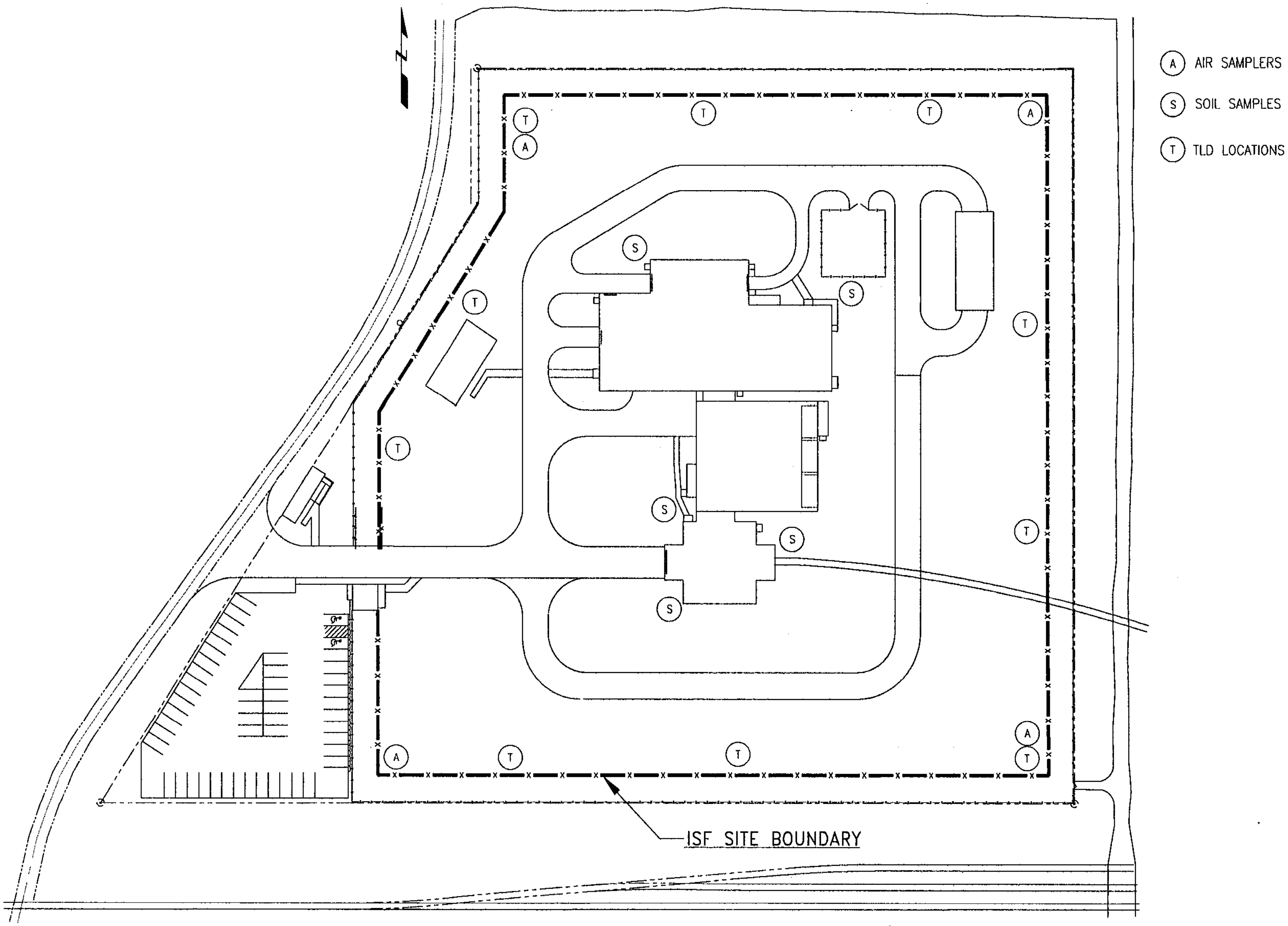

Figure 7.6-2
Operational Environmental Radiation Monitoring Locations