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# 7. RADIATION PROTECTION

This chapter presents the radiation protection features of the NUHOMS®-12T system using the NRC certified MP-187 for transportation from TAN to INTEC. Appendix E of this SAR presents the radiation protection features of the NUHOMS®-12T system using the NRC 10 CFR 72 approved OS-197 Transfer Cask for transportation from TAN to INTEC.

## 7.1 Ensuring That Occupational Radiation Exposures Are As-Low-As-Reasonably-Achievable

### 7.1.1 Policy Considerations

It is the policy of the DOE Idaho Operations Office (DOE-ID) to take every precaution to control radiation and the spread of radioactive contamination in the performance of work, to be in full compliance with the requirements established by the NRC, to prevent unnecessary radiation exposure to employees and the public, and to prevent harmful effects to the environment. Radiological operations at the ISFSI will be conducted in a manner consistent with those at the Idaho National Laboratory (INL). Radiation exposures to workers and the public and releases of radioactivity to the environment are maintained below regulatory limits, and deliberate efforts are taken to further reduce exposures and releases to As-Low-As-Reasonably-Achievable (ALARA) levels.

To comply with this policy, all levels of line management are accountable for radiological performance. The responsibility for compliance with the radiological protection requirements and for minimizing personnel radiation exposure begins at the worker level and broadens as it progresses upward through the line organization. Line managers are responsible for taking all necessary actions to ensure that requirements are implemented and that performance is monitored and corrected as necessary. Radiological Control Technicians (RCTs) assist line management by routinely evaluating and monitoring all radiological conditions. Also, RCTs oversee activities to ensure that all reasonable precautions are taken by personnel.

The requirements for the ALARA policy and program are provided by 10 CFR Part 20, "Standards for Protection Against Radiation" [7.1]. DOE-ID is committed to reducing safety and health risks associated with hazardous substances (including ionizing radiation) by promoting ALARA policy awareness, and reducing and keeping radiation exposures to ALARA levels. The following methods are used to achieve ALARA objectives.

A. Establishing employee and organizational level ALARA goals, tracking employee exposure, and maintaining associated records.

B. Allocating the appropriate technical, administrative, and supervisory resources.

C. Appointing an ALARA committee to oversee and evaluate efforts, and to provide technical assistance for identifying needed improvements. The TMI-2 ISFSI Facility Safety Officer acts as the ISFSI representative on the INTEC ALARA Committee. Radiation safety issues that arise are brought to the ALARA Committee by the Facility Safety Officer.

D. Controlling access to radiation and radioactive contamination areas.

E. Minimizing the working time required in high radiation areas and high surface contamination areas, as appropriate.

F. Using engineered controls (e.g., ventilation, remote handling, and shielding) and monitoring equipment (e.g., continuous air monitors and remote area monitors).

G. Requiring an ALARA review of procedures and work packages for the TMI-2 ISFSI that involve radiological work resulting in individual and/or collective radiation exposure exceeding thresholds established by the ALARA Committee or requiring entry into 1 rem/h radiation fields.

### 7.1.2 Design Considerations

The NUHOMS®-12T design installed at INL incorporates design features and improvements from previously constructed NUHOMS® installations at H.B. Robinson, Oconee, Calvert Cliffs, and Davis-Besse. The NUHOMS® 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. The ALARA design criteria for the NUHOMS®-12T system results in generally lower dose rates and exposures than those of the previously licensed Standardized NUHOMS® system [7.3]. Specific features of the NUHOMS® system that are directed toward ensuring ALARA include:

A. Thick concrete walls and roof on the HSM to minimize the on-site and off-site dose contribution from the ISFSI.

B. A thick shield plug on each end of the DSC to reduce the dose to plant workers while performing venting and sealing operations, and during transfer and storage of the DSC in the HSM.

C. Use of a heavy shielded cask for DSC handling and transfer operations to ensure that the dose to employees and the general public is minimized.

D. Fuel loading procedures which follow accepted practice and build on existing experience.

E. A recess in the HSM access opening to dock and secure the cask during DSC transfer so as to reduce direct and scattered radiation exposure.

F. Use of a heavy shielded door on the HSM to minimize direct and scattered radiation exposure.

G. Use of a passive system design for long-term storage that requires minimal maintenance.

H. Use of proven procedures and experience to control contamination during canister handling and transfer operations.

I. Use of temporary shielding during DSC closure operations as necessary to further reduce the direct and scattered dose.

J. A labyrinth design for the DSC vent and purge ports to minimize exposures during welding and filter installation and change-out.

Further ALARA measures may be implemented, as necessary, by the DOE-ID.

### 7.1.3 Operational Considerations

Consistent with the DOE-ID's overall commitment to keep occupational radiation exposures ALARA, specific plans and procedures will be followed by ISFSI operations personnel to ensure that ALARA goals are achieved consistent with the intent of Regulatory Guides 8.8 [7.4] and 8.10 [7.5] and the requirements of 10 CFR Part 20. Since the ISFSI is a passive system, minimal maintenance is expected on a normal basis. Maintenance activities that could involve significant radiation exposure of personnel will be carefully planned utilizing previous operating experience. Maintenance activities will be performed using well-trained and certified personnel and proper equipment. Where applicable, formal ALARA reviews will be prepared which specify radiation exposure reduction techniques, such as those set out in Regulatory Guide 8.8.

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## 7.2 Radiation Sources

### 7.2.1 Characterization of Sources

Each DSC stored in the NUHOMS®-12T ISFSI at INL contains up to 12 TMI-2 core debris canisters. At the time of the accident (March 28, 1979), the TMI-2 core was in its initial fuel cycle at 97% of full power with an average core burnup of 3,175 megawatt days per metric ton of uranium (MWd/MTU). The core material includes fuel assemblies, including all fuel rods, control rods, axial power shaping rods, guide tubes, instrument tubes, spacer sleeves, spacer grids, and end fittings. Core material also includes primary startup sources, control rod spiders, coupling mechanisms and other miscellaneous material as noted in Reference 3.3.

The core had 177 Babcock and Wilcox 15x15 fuel assemblies, identical in mechanical construction. Each assembly had 208 fuel rods, 16 control rod guide tubes, one instrument tube, seven spacer sleeves, eight spacer grids, and two end fittings. The guide tubes, spacer grids, and end fittings formed the structural cage which arranged the rods and tubes in a 15x15 array. Three initial enrichments were used in the core, ranging from 1.98 weight percent (w/o) $^{235}U$ to 2.98 w/o $^{235}U$.

The TMI-2 core debris is contained in three types of canisters: *fuel*, *knockout*, and *filter*. The internal structure of each canister type is the differentiating feature and is dictated by the canister function. *Fuel* canisters contain a mixture of large pieces of core debris up to partial-length, full cross-section fuel assemblies. *Knockout* canisters contain loose core rubble of a size small enough to be vacuumed up from the rubble bed. *Filter* canisters contain the filter elements and the small fuel fines removed by the many filters which cleaned the water circulated through the vacuum defueling system and the defueling water cleanup system.

Design basis neutron and gamma-ray sources for the TMI-2 debris canisters were calculated using the ORIGEN2.1 computer code [7.6]. An intact B&W 15x15 fuel assembly was modeled with a burnup of 3,175 MWd/MTU, an initial enrichment of 1.98 w/o $^{235}U$ (use of the minimum enrichment results in conservative sources), and a specific power of 27.137 MW/MTU. These parameters correspond to a cycle length of 117 days prior to the accident. The specific power used in the ORIGEN model was increased by a factor of 1.879 to convert from a core average burnup to a peak assembly burnup. The resulting sources were decayed for 19 years (to March 1998) and scaled from the 1515 lb fuel assembly weight to the 1908 lb weight of the peak canister payload. This design basis source was then conservatively used in the shielding calculations for all of the TMI-2 canisters.

The neutron source, spectrum, and flux-to-dose rate conversion factors [7.8] are provided in Table 7.2-1. The energy structure is that of the CASK-81 cross-section library [7.9] and the energy spectrum corresponds to the neutron spectrum from the spontaneous fission of $^{240}Pu$, which is the primary neutron source. Similarly, the gamma-ray source, spectrum, and flux-to-dose rate factors are provided in Table 7.2-2. The gamma source spectrum was mapped from the 18 ORIGEN2.1 energy groups to the CASK-81 energy groups by assuming that the photon energies are logarithmically distributed within each energy group.

### 7.2.2 Airborne Radioactive Material Sources

The potential for airborne radioactive material sources existed during fuel handling at the TAN facility, evacuating and sealing of the DSC, and DSC transfer and storage. Potential airborne releases from handling of the TMI-2 debris canisters at the TAN facility were handled in accordance with existing DOE-ID practices. DSC evacuating and sealing operations were performed using procedures which prevent airborne leakage. During these operations, all vent lines were routed to the existing radwaste systems at the TAN facility.

During storage, the DSC cavity is vented to the atmosphere through HEPA grade filters. No significant releases are expected from the DSC for the following reasons: (1) Much of the volatile fission product inventory was released during the accident and the remainder is entrapped within the fuel matrix as determined by extensive examinations performed on the core materials following the accident; (2) No differential pressure exists between the DSC and the atmosphere to provide a driving force for a significant release; (3) The DSC HEPA filters will prevent the release of solids and particulates as specified in Section 4.3.1.1. Although no significant releases are expected, both normal and accident (confinement failure) releases have been postulated with release fractions calculated using the methodology of Reference [7.7] to demonstrate the safety of the system. The postulated releases are non-mechanistic in that there is no conceivable event which would result in a significant release of the TMI-2 core debris material.

During normal operation, all 29 DSCs and the spare HSM are assumed to vent to the atmosphere. Each DSC is assumed to contain up to 12 design basis (as defined in Section 7.2.1) TMI-2 canisters. Although excessive temperatures would be required to release the volatile fission products from the fuel matrix, 10% of the noble gas inventory is assumed to be released directly to the environment each year. Additionally, per Reference [7.7], 0.1% and 0.0001% per year of particulates and solids, respectively, are assumed to be released from the canisters to the DSC. Of these releases, 1% of the solids and particulates are released from the DSC to the HEPA filter system which removes all but 0.03% of particulates. The 62 *filter* canisters are assumed to contain particulate matter and the remaining 282 canisters, which contain the larger debris material, are assumed to contain primarily all solids.

Table 7.2-3 provides the total quantity of potential airborne sources in each canister, the postulated annual release during normal operation, and the postulated accident release. Source activities have been taken from the design basis ORIGEN2.1 model discussed in Section 7.2.1. The postulated accident condition is a breach of one DSC or a failure of the HEPA filters of one DSC. Release fractions for the accident are identical to those of normal operation with the exception that no credit is taken for the HEPA filters. This condition is assumed to occur for one DSC containing 12 *filter* canisters (100% particulate). The periodic surveillances and radiological control surveys will limit the duration of undetected releases resulting from the postulated accident.

**Table 7.2-1**
**Neutron Energy Spectrum and Flux-to-Dose Conversion Factors for PWR Spent Fuel[1]**

| Cask Group | $E_{upper}$ (MeV) | Spectrum | Source (n/sec/can) | Flux-to-Dose Factor (mrem/hr per n/cm$^2$/sec) |
|---|---|---|---|---|
| 1 | 1.492E+01 | 7.005E-05 | 4.830E+01 | 1.945E-01 |
| 2 | 1.220E+01 | 5.129E-04 | 3.536E+02 | 1.597E-01 |
| 3 | 1.000E+01 | 2.456E-03 | 1.693E+03 | 1.471E-01 |
| 4 | 8.180E+00 | 1.170E-02 | 8.067E+03 | 1.477E-01 |
| 5 | 6.360E+00 | 3.197E-02 | 2.204E+04 | 1.534E-01 |
| 6 | 4.960E+00 | 4.775E-02 | 3.292E+04 | 1.506E-01 |
| 7 | 4.060E+00 | 1.101E-01 | 7.590E+04 | 1.389E-01 |
| 8 | 3.010E+00 | 9.376E-02 | 6.465E+04 | 1.284E-01 |
| 9 | 2.460E+00 | 2.250E-02 | 1.551E+04 | 1.253E-01 |
| 10 | 2.350E+00 | 1.250E-01 | 8.615E+04 | 1.263E-01 |
| 11 | 1.830E+00 | 2.224E-01 | 1.534E+05 | 1.289E-01 |
| 12 | 1.110E+00 | 1.938E-01 | 1.336E+05 | 1.169E-01 |
| 13 | 5.500E-01 | 1.238E-01 | 8.538E+04 | 6.521E-02 |
| 14 | 1.110E-01 | 1.413E-02 | 9.741E+03 | 9.188E-03 |
| 15 | 3.350E-03 | 7.106E-05 | 4.900E+01 | 3.713E-03 |
| 16 | 5.830E-04 | 5.166E-06 | 3.562E+00 | 4.009E-03 |
| 17 | 1.010E-04 | 3.398E-07 | 2.343E-01 | 4.295E-03 |
| 18 | 2.900E-05 | 4.909E-08 | 3.385E-02 | 4.476E-03 |
| 19 | 1.010E-05 | 1.058E-08 | 7.294E-03 | 4.567E-03 |
| 20 | 3.060E-06 | 1.649E-09 | 1.137E-03 | 4.536E-03 |
| 21 | 1.120E-06 | 3.634E-10 | 2.505E-04 | 4.370E-03 |
| 22 | 4.140E-07 | 1.050E-10 | 7.237E-05 | 3.714E-03 |
| **Total** | | **1.000E+00** | **6.895E+05** | **-** |

(1) An additional source term for neutrons of 4.5 MeV average energy for each of two canisters, one stored in DSC 1/HSM 4 and another stored in DSC 5/HSM 22, is 7.3E6 n/sec/canister due to the presence of AmBeCm startup source material. [7.20]

**Table 7.2-2**
**Gamma Energy Spectrum and Flux-to-Dose Conversion Factors for PWR Fuel**

| Cask Group | $E_{upper}$ (MeV) | Source ($\gamma$/sec/can) | Flux-to-Dose Factor (mrem/hr per $\gamma/cm^2$/sec) |
|---|---|---|---|
| 23 | 10.00 | 2.401E+02 | 8.772E-03 |
| 24 | 8.00 | 1.523E+03 | 7.479E-03 |
| 25 | 6.50 | 8.988E+03 | 6.375E-03 |
| 26 | 5.00 | 1.027E+04 | 5.414E-03 |
| 27 | 4.00 | 1.924E+05 | 4.622E-03 |
| 28 | 3.00 | 7.196E+06 | 3.960E-03 |
| 29 | 2.50 | 7.066E+07 | 3.469E-03 |
| 30 | 2.00 | 3.533E+10 | 3.019E-03 |
| 31 | 1.66 | 3.504E+12 | 2.628E-03 |
| 32 | 1.33 | 8.250E+12 | 2.205E-03 |
| 33 | 1.00 | 1.364E+12 | 1.833E-03 |
| 34 | 0.80 | 9.471E+13 | 1.523E-03 |
| 35 | 0.60 | 1.773E+14 | 1.173E-03 |
| 36 | 0.40 | 5.260E+12 | 8.759E-04 |
| 37 | 0.30 | 9.887E+12 | 6.306E-04 |
| 38 | 0.20 | 2.068E+13 | 3.834E-04 |
| 39 | 0.10 | 4.836E+13 | 2.669E-04 |
| 40 | 0.05 | 2.679E+14 | 9.348E-04 |
| **Total** | | **6.372E+14** | **-** |

**Table 7.2-3**
**Postulated Airborne Radioactive Material Sources**

| Nuclide | Activity (Ci/can) | Activity Percent | Normal Operation Release[1] (Ci/yr) | Accident Release[2] (Ci) |
|---|---|---|---|---|
| CS137 | 7.34E+03 | 23.154% | 1.439E-03 | 7.338E-02 |
| BA137M | 6.94E+03 | 21.901% | 1.361E-03 | 6.941E-02 |
| Y90 | 5.87E+03 | 18.525% | 1.151E-03 | 5.871E-02 |
| SR90 | 5.87E+03 | 18.519% | 1.151E-03 | 5.869E-02 |
| PU241 | 4.20E+03 | 13.256% | 8.238E-04 | 4.201E-02 |
| KR85 | 3.57E+02 | 1.125% | 1.284E+04 | 3.566E+01 |
| PM147 | 2.53E+02 | 0.797% | 4.954E-05 | 2.526E-03 |
| AM241[3] | 2.07E+02 | 0.653% | 4.059E-05 | 2.070E-03 |
| CO60 | 1.42E+02 | 0.448% | 2.785E-05 | 1.420E-03 |
| PU239 | 1.14E+02 | 0.360% | 2.236E-05 | 1.140E-03 |
| SM151 | 9.38E+01 | 0.296% | 1.839E-05 | 9.377E-04 |
| PU240 | 6.01E+01 | 0.190% | 1.178E-05 | 6.009E-04 |
| NI63 | 5.57E+01 | 0.176% | 1.093E-05 | 5.572E-04 |
| EU154 | 4.30E+01 | 0.136% | 8.438E-06 | 4.303E-04 |
| H3 | 4.27E+01 | 0.135% | 1.539E+03 | 4.274E+00 |
| EU155 | 3.10E+01 | 0.098% | 6.081E-06 | 3.101E-04 |
| PU238 | 2.95E+01 | 0.093% | 5.783E-06 | 2.949E-04 |
| SB125 | 1.93E+01 | 0.061% | 3.776E-06 | 1.925E-04 |
| CS134 | 7.96E+00 | 0.025% | 1.560E-06 | 7.957E-05 |
| I129 | 3.28E-03 | 0.000% | 6.434E-10 | 3.281E-08 |

(1) Normal operation release source for all 30 HSMs in the ISFSI.

(2) Accident release source assumes that all five HEPA filters in a single DSC fail for a period of one month.

(3) An additional 3.3 Ci of Am-241 is postulated for each of two canisters, one stored in DSC 1/HSM 4 and another stored in DSC 5/HSM 22, due to presence of AmBeCm startup source material. The corresponding additional release per canister for normal operation and accident conditions is postulated to be 2.2E-8 Ci/y and 3.2E-5 Ci, respectively [7.20].

## 7.3 Radiation Protection Design Features

### 7.3.1 Installation Design Features

The design considerations listed in Section 7.1.2 ensure that occupational exposures to radiation are ALARA and that a high degree of integrity is achieved through the confinement of radioactive materials inside the DSC. Applicable portions of Regulatory Position 2 of Regulatory Guide 8.8 [7.4] have been used as guidance.

A. Access control to the ISFSI is through a controlled gate in the perimeter fence.

B. Radiation shielding substantially reduces the exposure of personnel during system operations and storage.

C. The NUHOMS® system is a passive storage system; no process instrumentation or controls are necessary during storage. The only required surveillance or instrumentation is the periodic HEPA filter monitoring and periodic aging management inspections.

D. Airborne contaminants are confined by the high integrity welded DSC assembly and the integral DSC HEPA filtration system.

E. No crud is produced by the NUHOMS® system.

F. The necessity for decontamination is reduced by maintaining the cleanliness of the DSC and the cask during fuel loading and unloading operations (see Section 5.1). Additionally, the DSC and cask surfaces are smooth, nonporous, and are generally free of crevices, cracks, and sharp corners.

### 7.3.2 Shielding

#### 7.3.2.1 Radiation Shielding Design Features

Radiation shielding is an integral part of all NUHOMS® component designs. The features described in this section assure that doses to personnel and the public are ALARA. The following paragraphs and figures describe the radial and axial shielding provided by the NUHOMS®-12T system.

Radial shielding during loading and transfer is provided primarily by the cask. This shielding includes a stainless-steel inner liner, lead, and a stainless steel structural shell. Neutron shielding in the radial direction is provided by an outer metal jacket which forms an annulus with the cask structural shell. This annulus is filled with a solid neutron absorbing material to provide neutron dose attenuation. During storage, radial shielding is primarily provided by the thick concrete walls of the HSM.

Axial shielding during loading and transfer is provided by the thick steel DSC shield plugs and the cask steel top and bottom cover plates. Two penetrations in the top shield plug provide a means for evacuating and venting the DSC. The penetrations are located on the perimeter of the DSC away from the TMI-2 canisters and contain sharp bends to minimize radiation streaming. During storage, axial shielding is provided by the concrete HSM rear wall, the concrete and steel HSM door, and the DSC shield plugs. A thick steel cover provides shielding for the ventilation system.

The shielding geometry for the HSM and the cask are shown in Figure 7.3-1 and Figure 7.3-2, respectively. The geometry of the DSC vent is shown in Figure 7.3-3. Additional portable shielding during DSC handling, transport and transfer operations will be used by the DOE contractor as needed in accordance with existing ALARA practices. If used, the base of the welding machine used to place the top cover welds includes an integral neutron and gamma shield to minimize exposures during closure operations.

#### 7.3.2.2 Shielding Analysis

This section describes the radiation shielding analytical methods and assumptions used in calculating the NUHOMS® system dose rates during the handling and storage operations. The dose rates of interest are calculated at the locations listed in Table 7.3-1 for the design basis TMI-2 core debris. These results are shown graphically in Figure 7.3-4 for the HSM. The computer codes used for analysis are described below, each with a brief description of the input parameters generic to its use. Descriptions of the individual analytical models used in the analysis are also provided.

A. **Computer Codes:** Surface dose rates for the HSM and cask were calculated using the two-dimensional discrete ordinates transport computer code DORT [7.10]. The CASK cross section library, which contains 22 neutron energy groups and 18 gamma energy groups, is applied in an $S_{16}P_3$ approximation in DORT. Calculated radiation fluxes are multiplied by flux-to-dose conversion factors (Table 7.2-1 and Table 7.2-2) to obtain final dose rates. The DORT calculations use coupled neutron and gamma libraries. Therefore, dose rates from both primary and secondary gammas are calculated in each run.

   Dose rates around the DSC vent and the DSC top covers were calculated using the three- dimensional monte-carlo computer code MCNP [7.11]. The MCNP code uses continuous energy cross section data for both neutron and gamma-ray transport. Calculated fluxes were converted to dose rates using the ANSI/ANS 6.1.1-1977 flux-to- dose factors [7.8]. The MCNP code was also used for the site dose calculations presented in Section 7.4.2.

   Exposures due to postulated radioactive releases are calculated using the RSAC-5 computer code [7.13]. The RSAC code was developed at INL to calculate the consequences of the release of radionuclides to the atmosphere. The code has the capability to generate a fission product inventory, decay and ingrow the inventory during transport through processes, facilities, and the environment, model the downwind dispersion of the activity, and calculate doses to downwind individuals. Atmospheric dispersion was calculated using the Markee model of Gaussian plume diffusion, as included in the RSAC code.

B. **HSM Surface Dose Rates:** Three DORT models were used to model the fuel debris, DSC, and HSM. Two axisymmetric models, shown in Figure 7.3-1, model the HSM roof, front wall, and back wall. The roof model includes everything in Figure 7.3-1 above the DSC centerline and the floor model includes the remainder. The fuel debris is modeled as a homogenous cylinder and no credit is taken for the DSC basket. The 1/4 inch thick debris canister shell has been added to the DSC shell thickness. Using axisymmetric models for the HSM results in conservative dose rates over the bulk of the HSM surface. A third, Cartesian model, is used to estimate the dose rates along the gap between modules. Material properties for concrete are taken from ANSI/ANS-6.4 [7.14] and no credit is taken for reinforcing bars. The neutron and gamma-ray dose rate results for the HSM are reported in Table 7.3-1 and Figure 7.3-4.

C. **Cask Dose Rates:** The NUHOMS®-MP187 cask will be used during loading, on-site transfer, and off-site transportation of the DSCs. An axisymmetric DORT model of the MP187 cask, using dimensions and materials described in the Rancho Seco ISFSI SAR [7.15], was generated to calculate the neutron and gamma dose rates on the surface of the cask. The fuel debris is modeled as a homogenous cylinder and no credit is taken for the DSC basket. The 1/4 inch thick debris canister shell has been added to the DSC shell thickness. Cask surface dose rates are reported in Table 7.3-1.

D. **DSC Vent and Cover Dose Rates:** The DSC top shield plug and vent port dose rates were calculated using a three-dimensional MCNP model of the fuel debris, canisters, DSC basket, and DSC. The geometry of the model in the vicinity of the vent opening is shown in Figure 7.3-3. The TMI-2 debris is assumed to be homogenized within each TMI-2 canister, and the design basis gamma source term is applied to each canister. Neutron dose rates were not calculated because the HSM results provide assurance that the neutron doses are negligible. Dose rates at the surface of the vent were calculated using a point detector located as shown in Figure 7.3-3. Dose rates on the shield plug surface were calculated using a surface crossing tally located at the center of the plug.

Dose rates on the surface of the DSC shield plug and external to the vent and purge ports are provided in Table 7.3-1. The DSC shield plug dose rates are well below those of the Standardized NUHOMS® system [7.3]. The peak dose rate external to the vent and purge ports is of a magnitude similar to the annulus dose rates previously observed during loading of NUHOMS® DSCs. Therefore, operational exposures for welding the NUHOMS®-12T DSC will be similar to those observed for the Standardized NUHOMS® system. To keep exposures ALARA, temporary shielding and remote handling equipment may be utilized when access to the vent and purge ports is required.

### 7.3.3 Ventilation

As stated in Section 7.2.2, all process flows from the DSC during loading operations at TAN will be handled in accordance with DOE-ID's current practices. During storage, the

DSC will be vented to the atmosphere through the venting system described in Section 4.3.1. As stated in Section 7.2.2 and in Section 4.3.1, no significant radioactive releases are expected through the venting system. The majority of volatile fission products were released during the TMI-2 accident or during the 10 years of storage at TAN. Because the DSC is vented to the atmosphere, there is no driving pressure to force material into the environment. In the event that any material does escape the DSC, the venting system includes HEPA grade filters (removal efficiencies of 99.97% for 0.3 micron particles).

Although no significant releases are expected from the DSC, Section 7.6 includes an exposure contribution from a postulated normal operation release. Chapter 8 includes an evaluation estimating the consequences of a total failure of the DSC and/or HEPA filters. The results of both analyses show exposures to employees and the general public well below the applicable limits.

### 7.3.4 Area Radiation and Airborne Radioactivity Monitoring Instrumentation

Radiological monitoring and contamination control at ISFSI are performed to ensure that radiation exposure and release limits contained in 10 CFR Part 20 are not exceeded. The ISFSI was added to the existing INL radiological control program which monitors, as appropriate, radiation levels, contamination levels and airborne radioactivity.

**Table 7.3-1**
**Shielding Analysis Results**

| Location | Peak Dose Rate (mrem/hr) Neutron | Peak Dose Rate (mrem/hr) Gamma | Average Dose rate (mrem/hr) Neutron | Average Dose rate (mrem/hr) Gamma |
|---|---|---|---|---|
| **DSC in HSM** | | | | |
| Roof/Side Walls | 0.002 | 10.4 | 0.001 | 6.5 |
| Front Wall | 0.081 | 12.7 | 0.027 | 4.9 |
| Rear Wall | 0.235 | 104.5 | 0.034 | 7.6 |
| Module Gap | 0.002 | 4.34 | n/a | n/a |
| **DSC in Cask** | | | | |
| Cask Side | 1.35 | 1.85 | 0.16 | 0.66 |
| Cask Top | 0.33 | 0.30 | 0.24 | 0.21 |
| Cask Bottom | 0.31 | 0.16 | 0.22 | 0.11 |
| DSC Shield Plug | n/a(1) | 17.1 | n/a | n/a |
| DSC Vent Port | 42(2) | 926 | n/a | n/a |

(1) Analysis performed only for gamma-ray doses. Neutron doses represent less than 1.5% of the total doses at these locations based on the DORT model described in section 7.3.2.2(C) and have been neglected.

(2) An estimated peak neutron dose equivalent rate for each of two canisters, one stored in DSC 1/HSM 4 and another stored in DSC 5/HSM 22, attributed to AmBeCm startup source material [7.20].

**Figure 7.3-1**
**DSC and HSM Shielding Geometry**

Security-Related Information
Figure Withheld Under 10 CFR 2.390.

**Figure 7.3-2**
**DSC and Cask Shielding Geometry**

Security-Related Information
Figure Withheld Under 10 CFR 2.390.

**Figure 7.3-3**
**DSC Ventilation Port Geometry**

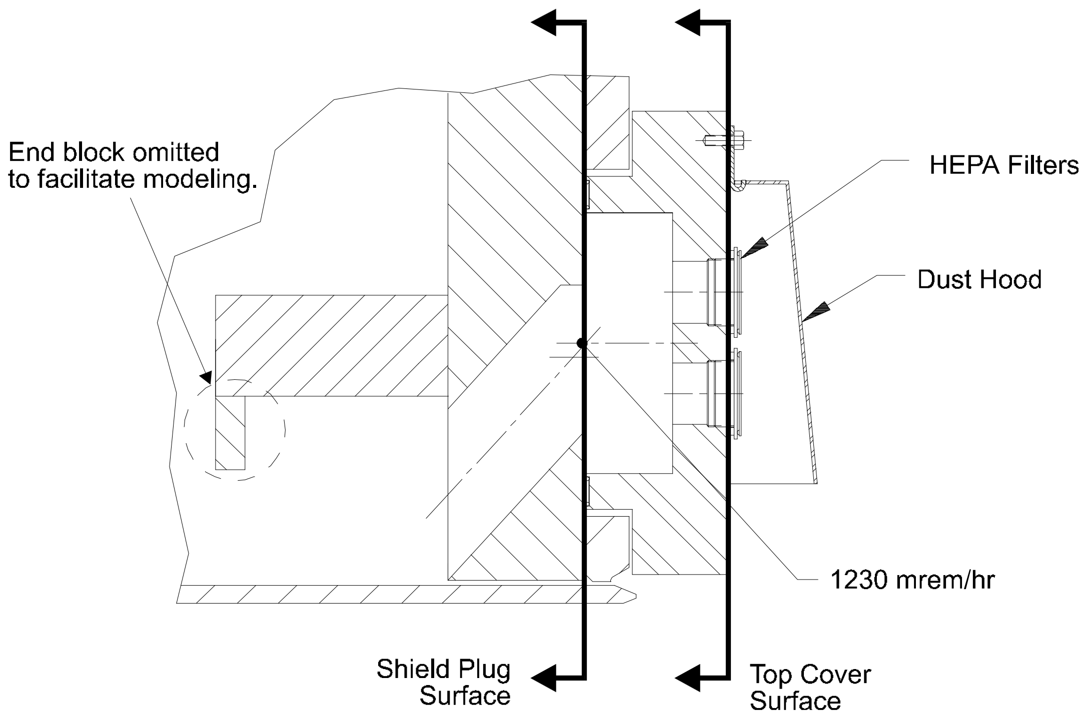
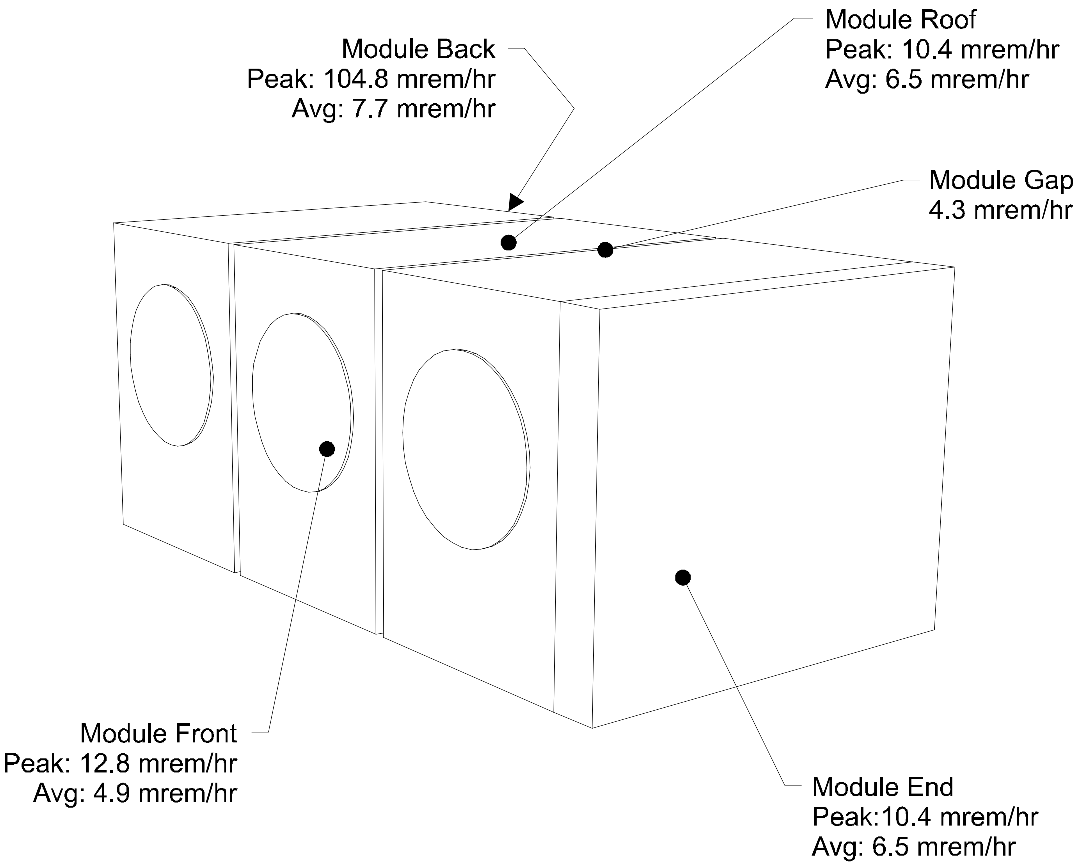


**Figure 7.3-4**
**HSM Shielding Analysis Results**



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## 7.4 Estimated On-Site Collective Dose Assessment

### 7.4.1 Operational Dose Assessment

This SAR section establishes the anticipated cumulative dose exposure to site personnel during the fuel handling and transfer activities associated with utilizing one NUHOMS® HSM for storage of one DSC. Chapter 5 describes in detail the NUHOMS® operational procedures, a number of which involve potential radiation exposure to personnel.

A summary of the operational procedures which result in radiation exposure to personnel is given in Table 7.4-1. The cumulative dose can be calculated by estimating the number of individuals performing each task and the amount of time associated with the operation. The resulting man-hour figures can then be multiplied by appropriate dose rates near the transfer cask surface, the exposed DSC top surface, or the HSM front wall. Dose rates are referenced in Table 7.3-1 for the DSC, cask, and HSM.

Every operational aspect of the NUHOMS® system, from canister loading through sealing, transport, transfer, and operation is designed to assure that exposure to personnel is ALARA. Dose rates are kept ALARA by the shielded DSC end plugs and shielded cask. The vent and purge ports have been designed with bends and shield plates to minimize streaming during DSC sealing and filter change-outs. Exposures are kept ALARA by performing most operations remotely as follows: (1) The debris canister drying and loading into the DSC were performed remotely in TAN Hot Shop; (2) If it is used, the welding machine was pre-installed on the top shield plug and top cover plates, away from the DSC, and operated remotely; (3) Transfer operations were performed inside the heavily shielded MP187 cask and trailer; and (4) Cask alignment operations were performed using a remote hand-held pendant and the hydraulic ram was operated using a remote power unit. In addition, many engineered design features are incorporated into the NUHOMS® system which minimize occupational exposure to plant personnel during placement of fuel in dry storage as well as off-site dose to the nearest neighbor during storage. The resulting dose at the ISFSI site boundary is well within the limits specified by 10 CFR 72.

Because the predicted dose rates for the NUHOMS®-12T system are well below those predicted for previous NUHOMS® systems, occupational exposures for the TMI-2 ISFSI will be bounded by those observed at other installations. Based on experience from operating NUHOMS® systems at Oconee, Calvert Cliffs, and Davis-Besse, the occupational dose for placing a DSC with TMI-2 core debris into dry storage for the operational steps listed in Table 7.4-1 will be much less than one person-rem. With the use of effective procedures and experienced ISFSI personnel, the total accumulated dose can be reduced below 500 person-mrem per DSC.

If a DSC vent housing seal fails a leak rate test, the estimated collective occupational exposure to reseat or replace the seals while the DSC remains in place is 60 person-mrem [7.22]. Subsequent to a DSC failing a seal leak rate test, any decontamination in a DSC vent housing area will increase the collective occupational exposure at a rate of 5 person-mrem/h.

### 7.4.2 Site Dose Assessment

A site dose assessment for the ISFSI has been performed using the average HSM surface dose rates presented in Figure 7.3-4 as input. Locations of interest for the assessment include the INTEC fence and the INL site boundary. The INTEC fence serves as the restricted area boundary for demonstrating compliance with the 10 CFR 20.1502 limit 0.5 rem per year, requiring individual monitoring of external and internal occupational dose. The INL site boundary serves as the controlled area boundary for compliance with 10 CFR 20.1301, 10 CFR 72.104, and 10 CFR 72.106 [7.2]. Direct and air-scattered radiation doses from all 30 HSMs at these locations have been calculated using the MCNP monte-carlo computer code [7.11].

The MCNP code was used to model the two rows of HSMs, the concrete basemat, and the surrounding land and air as shown in Figure 7.4-1. Source particles are started on the surfaces of the arrays which are modeled as solid concrete simply to account for self-shielding and scattering. No credit is taken for shielding by nearby structures or terrain.

Based on the low neutron dose rates on the surface of the HSMs (less than 0.05 mrem/hr as shown in Table 7.3-1), only gamma-rays are considered in the site dose assessment. Source particles are assumed to leave the surfaces of the modules with an angular distribution approximating a cosine function and an energy distribution of photons shielded by three feet of concrete. The total activity (photons/second) of each array face is used as input to the MCNP surface source.

Dose rates have been calculated using point detectors located around the ISFSI fence and the INTEC restricted area fence. Ring detectors are used to calculate dose rates at distances from 200 meters to 1000 meters. The 1000-meter dose rate is assumed to apply to all distances greater than 1000 meters. Ring detectors have been used at these distances to improve the statistical accuracy of the calculations. Because the air-scattered dose rates, which are relatively independent of the orientation relative to the ISFSI, dominate the results at these distances, this assumption has little effect on the results. Dose rates at various locations around the ISFSI and the acceptance criteria are provided in Table 7.4-2. As can be seen in Table 7.4-2, the site dose rates are well below the applicable 10 CFR 20.1301, 10 CFR 20.1502, and 10 CFR 72.104 limits. Figure 7.4-2 provides the dose rate as a function of distance from the ISFSI.

**Table 7.4-1**
**NUHOMS® System Operations Enveloping Times for Loading One DSC**

| | Number of Workers | Completion Time[3] (hours) |
|---|---|---|
| **Location: TAN Facility Hot Cell** | | |
| Vacuum dry 12 TMI-2 Canisters[4] | 2 | 168 |
| Ready the DSC and MP187 Cask for Service | 4 | 4.0 |
| Place the DSC into the Cask in the TAN Hot Shop | 4 | 1.0 |
| Verify and Load the Dry TMI-2 Canisters[4] into the DSC | 2 | 2.0 |
| Open the TMI-2 canister vents[4] | 2 | 1.0 |
| Place the Top Shield Plug on the DSC[4] | 3 | 1.0 |
| Decontaminate the Outer Surface of the Cask[2] | 3 | 1.0 |
| Set-up Welder | 3 | 1.5 |
| Weld the Shield Plug to the DSC Shell and Perform Examination [1] | 3 | 6.0 |
| Purge the DSC and Backfill the DSC with Helium[1][5] | 2 | 4.0 |
| Helium Leak Test the Shield Plug Weld[5] | 2 | 1.0 |
| Weld the Top Cover Plate to DSC Shell and Perform Examination[1] | 3 | 16.0 |
| Install Vent System Filter Assemblies and Transportation Covers | 2 | 1.0 |
| Evacuate and Backfill DSC with Helium[5] | 2 | 4.0 |
| Install the Top Internal Spacer and Cask Lid | 2 | 1.0 |
| Ready the Cask Support Skid and Transport Trailer for Service[2] | 2 | 2.0 |
| Place the Cask onto the Skid and Trailer | 4 | 0.5 |
| Install Skid Frame, Impact Limiters, and Personnel Barrier[5] | 4 | 2.0 |
| | | |
| **Location: ISFSI Site** | | |
| Ready the HSM and Hydraulic Ram System for Service[2] | 2 | 2.0 |
| Transport the Cask to the ISFSI | 4 | 6.0 |
| Remove Impact Limiters[5] | 3 | 1.0 |
| Position the Cask in Close Proximity with the HSM | 3 | 1.0 |
| Remove the Cask Lid (note: the top internal spacer is attached to the lid) | 3 | 1.0 |
| Align and Dock the Cask with the HSM | 3 | 2.0 |
| Position and Align Ram with Cask | 3 | 1.0 |
| Transfer the DSC from the Cask to the HSM | 3 | 0.5 |
| Move the Ram Clear of Cask and Un-Dock the Cask from the HSM | 3 | 1.0 |
| Install the HSM Access Door and Seismic Restraint | 3 | 1.0 |
| Open Rear Wall Access Door, Remove the Filter Transportation Covers[5], Visually Check HEPA Filters, and Close Access Door | 2 | 1.0 |
| Perform Radiation Survey | 2 | 1.0 |

(1) Monitoring operation - personnel may leave the radiation work area.
(2) Operation may be performed in parallel with other activities.
(3) Time shown for each operation is enveloping (i.e. these are operational times and not necessarily exposure time). Actual times for similar operations have been considerably less.
(4) Performed remotely in TAN Hot Shop.
(5) Operations applicable only to the MP-187 transportation cask.

**Table 7.4-2**
**Dose Rates in the Vicinity of the TMI-2 ISFSI at INL**

| Location | Dose Rate (mrem/hr ) | Acceptance Criteria |
|---|---|---|
| INTEC Fence (max)[1] | 4.78E-02 | 0.5 rem/year (10 CFR 20.1502) |
| INL Site Boundary[2] | 3.68E-06 | 0.1 rem/yr (10 CFR 20.1301) |

Notes:

[1] Represents the restricted area boundary. Individuals entering this area are monitored for exposure to radiation and radioactive material at levels sufficient to demonstrate compliance occupational dose limits.

[2] Represents the controlled area boundary. Maximally Exposed Individual (MEI) is assumed to be at this location. This area is controlled by DOE-contracted security forces and DOE exercises authority over its use.

**Figure 7.4-1**
**ISFSI Geometry for Site Dose Calculations**

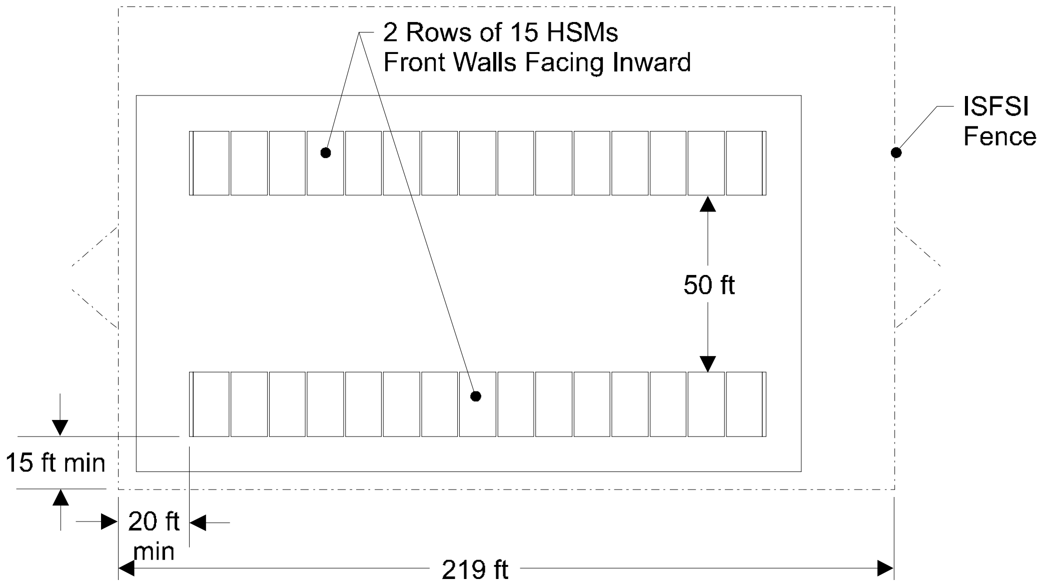
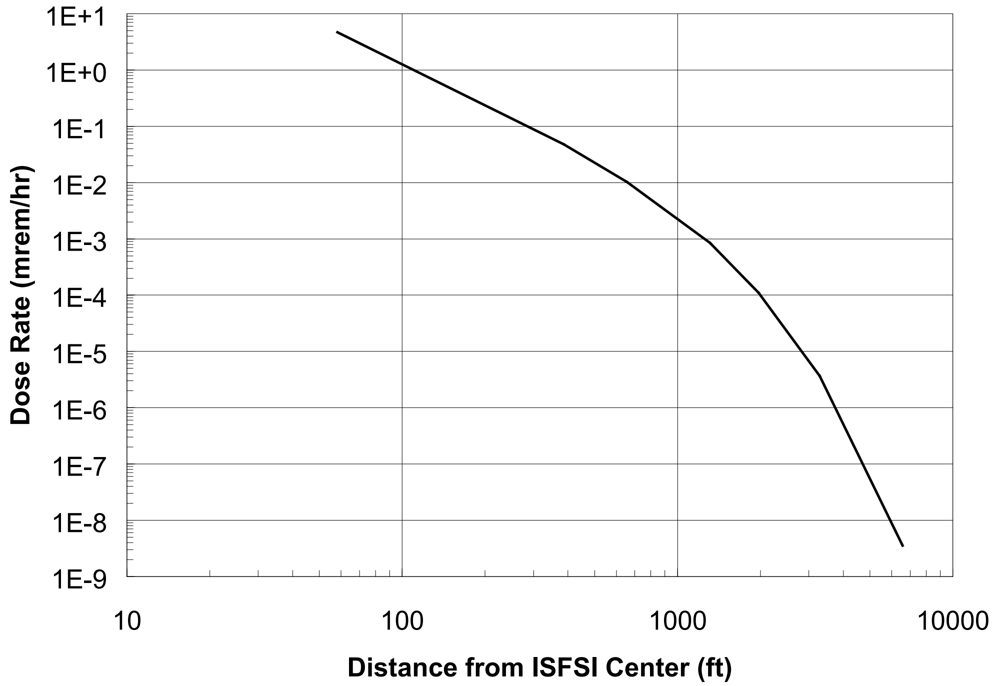


**Figure 7.4-2**
**Dose Rate Versus Distance from the ISFSI**



## 7.5 Health Physics Program

### 7.5.1 Organization

The radiation protection program at the TMI-2 ISFSI is described in a Radiation Protection Plan (RPP) which has been developed in accordance with 10 CFR 20 [7.1]. The program is implemented through the RPP as well as several INL radiation protection procedures referenced in the RPP. A Facility Safety Officer (FSO) that is independent of the facility operations organizational element is responsible for implementation of the radiation protection program. The FSO reports to the program manager at a professional level equivalent to operations personnel. Several matrixed radiation protection professionals (engineers and technicians) maintain qualification to implement the radiation protection program under the direction of the FSO.

### 7.5.2 Equipment, Instrumentation, and Facilities

The radiological equipment, instrumentation, and facilities for the ISFSI will be those currently used at INL, which includes complete health physics facilities to support the ISFSI radiation protection program. Properly selected, operated, maintained, and calibrated radiological instrumentation is employed at TMI-2 ISFSI in order to implement an effective radiation protection program. Typical instruments include:

A. A variety of portable beta-gamma detectors and suitable rate meters are used for surface contamination monitoring.

B. Low-background alpha-beta counters indicate the level of contamination on smears and air filters.

C. A variety of portable gamma and neutron dose rate meters.

D. Gamma spectroscopy capability for radionuclide identification.

The INL radiation protection procedures referenced in the RPP provide requirements for the calibration, response check, operational inspection, maintenance, and repair of standard radiological instruments used at INL. These procedures are applicable to both fixed and portable instruments.

### 7.5.3 Procedures

#### 7.5.3.1 Radiation Protection Practices

Radiation protection practices employed at the TMI-2 ISFSI include personnel protective equipment and permanent or temporary shielding as necessary to minimize the potential for personnel contamination and radiation exposure. Additional radiation protection practices used to control exposure include posting radiological areas, controlling activities within the radiologically controlled areas, and ensuring entry and exit control.

**Posting Areas:** NRC-approved signs, labels, and radiation symbols are conspicuously posted for radiologically controlled areas as required by 10 CFR Part 20. Each access point to a radiologically controlled area is posted. The size of the area is determined using the guidelines in 10 CFR Part 20. The alteration or removal of control barriers is performed by, or under the direction of, the RCTs.

**Radiological Work Permits (RWPs):** The RWP is an administrative mechanism used to establish radiological controls for intended work activities. The RWP informs workers of area radiological conditions and entry requirements and provides a mechanism to relate worker exposure to specific work activities. An RWP contains pertinent information for performing the intended work safely and within ALARA guidelines. The information includes, but is not limited to, a description of the work, radiological conditions of the work area, dosimetry requirements, stay time controls, and special dose or contamination reduction considerations.

RWPs are required to enter high and very high radiation areas, high contamination areas, and airborne radioactivity areas. Job-specific RWPs are used for non-routine operations or work in areas with changing radiological conditions. Job-specific RWPs remain in effect only for the duration of the job, whereas general RWPs are used for routine or repetitive activities, such as tours and inspections, or minor work activities in areas with well-characterized and stable radiological conditions. General RWPs can be approved for periods of up to one year.

The RWP is approved by the job supervisor and job controller for the work area as well as the appropriate radiation protection management representative and facility/area manager. Radiological surveys are reviewed to evaluate the adequacy of the RWP requirements. Workers acknowledge that they have read, understand, and will comply with the RWP before their initial entry to the area and after any revisions to the RWP.

**Entry and Exit Control:** Personnel entry control is maintained for each radiological area per 10 CFR Part 20. The degree of control is commensurate with existing and potential radiological hazards within the area. One or more of the following methods are used to ensure control: radiological posting, control devices on entrances, conspicuous visual and audible alarms, locked entrance ways, or administrative controls.

#### 7.5.3.2 Dosimetry

10 CFR Part 20 establishes the policy, requirements, and training necessary for assignment and use of external dosimetry. External dosimetry devices used for monitoring occupational whole body exposure are accredited by the DOE Laboratory Accreditation Program for the appropriate radiation types and categories.

External dosimetry provides indication of the radiation exposures received by personnel, equipment, and the environment. External dosimetry devices are capable of indicating both penetrating and nonpenetrating radiation exposure that contribute to a person's occupational exposure. External dosimetry for equipment and the environment provides an indication of the general radiation field in the ISFSI areas. All external dosimetry devices used at the ISFSI are analyzed by the INL Radiation Dosimetry and Records group.

Personnel dosimetry badges (either direct reading and/or electronic dosimetry) are issued to all personnel performing radiological work within the ISFSI, as required by the applicable RWP. Each employee is responsible for wearing his or her assigned badge while within the ISFSI. The optically stimulated luminescence (OSL) badges are analyzed to provide input into a computerized record system that accumulates employee and visitor exposure information. The reports are transmitted to management to inform them of the exposure status of all employees. In addition to external dosimetry, all ISFSI personnel who are likely to receive intakes resulting in an effective dose equivalent greater than 100 mrem undergo initial, periodic, and termination baseline whole-body counts or bioassays.

#### 7.5.3.3 Respiratory Protection

The RPP and referenced procedures provide guidelines for selecting respiratory equipment for protection against airborne radioactivity. These documents incorporate the requirements of ANSI Z88.2, "Practices for Respiratory Protection." Respirators for radiological exposure control are used in accordance with 10 CFR Part 20. ISFSI personnel are formally trained and qualified before using respiratory equipment.

#### 7.5.3.4 Radiological Protection Training

All individuals requiring access to radiological controlled areas at the ISFSI receive training that emphasizes the nature of radiological conditions and the control of radiation exposure. 10 CFR Part 20 provides instructions for determining the training requirements, based on activities and responsibilities of the workers.

Levels of training for RCTs are commensurate with the technician's assignment. Qualifications for RCT's consist of standardized course material, on-the-job training, and both a comprehensive written examination and final examination with the Oral Examination Board. The level of RCT qualification is based on the education, experience, training, orientation, and other qualification achieved and maintained by the individual. RCT qualifications follow a 2-year cycle of continuing training evaluated with a written and oral examination.

Training is also provided for other radiological support personnel (who provide health physics and radiological engineering, dosimetry, bioassay, independent oversight, and instrumentation calibration functions) to ensure that these personnel have the technical qualifications pertinent to their assigned duties. Health physicists and other radiation protection professionals with baccalaureate degrees in science or engineering or equivalent work experience are available to support the radiation protection program.

## 7.6 Estimated Off-Site Collective Dose Assessment

### 7.6.1 Effluent and Environmental Monitoring Program

The INL environmental surveillance program maintains a network of low-volume samplers to monitor for airborne radioactivity [7.21]. The network includes 14 onsite locations, 7 INL boundary locations, and 6 offsite locations. One of the onsite samplers is located about 1100 feet northwest of the ISFSI site near the INTEC entrance and west perimeter road. Samplers for monitoring tritium in the atmosphere are also located at two onsite and four offsite locations.

The INL environmental surveillance program also includes direct measurements of ambient (environmental) radiation levels using OSL dosimeters. These devices measure ionizing radiation exposure rates due to the combined sources of natural radioactivity in the air and soil, cosmic rays, residual fallout from nuclear weapons tests, and radioactivity from INL site operations. Dosimeters are located at 135 onsite locations and 27 offsite locations.

The ISFSI specific radiological environmental monitoring program includes monthly airborne radioactivity sampling within the ISFSI perimeter fence, direct radiation monitoring with OSL dosimetry placed along the ISFSI perimeter fence, and periodic loose surface radioactive contamination monitoring adjacent to each DSC vent and purge port and each HSM drain line.

The INL meteorological and environmental surveillance programs will be continued through the life of the TMI-2 ISFSI. The ISFSI specific radiological environmental monitoring program will also continue through the life of the TMI-2 ISFSI. Only the results of the ISFSI specific radiological environmental monitoring program will be reported to meet the 60-day reporting requirement of 10 CFR 72.44. The results of the INL environmental surveillance program will be available under separate cover, but outside the scope of the 60-day reporting requirement of 10 CFR 72.44.

### 7.6.2 Analysis of Multiple Contribution

The annual dose calculated for the maximally exposed individual as a result of current and projected INL sitewide emissions is about 0.05 mrem. Adding the ISFSI direct and air-scattered radiation at the controlled area boundary from Table 7.4-2 and the estimated effective dose equivalent from effluents from Section 7.6.3 (total of 0.04 mrem) results in a total annual exposure to the MEI of 0.09 mrem, well below the 25 mrem 10 CFR 72.104 limit and 10 mrem 40 CFR 61.92 limit [7.12].

### 7.6.3 Estimated Dose Equivalents

Dose equivalents from effluents as a function of distance from the ISFSI have been calculated using the RSAC-5 computer code [7.13]. Meteorological parameters were generated using hourly meteorological data taken over a six-year period. The greatest annual average atmospheric dispersion factor over this period is $4.81 \times 10^{-8}$ s/m$^3$, established for Atomic City, 18 km from the INTEC. This corresponds to a 4.45 m/s wind speed with class C stability, which was assumed for the effluent dose calculations. Normal operation nuclide releases from Table 7.2-3 were input to the code for calculations of exposure from inhalation, ingestion, ground surface dose, and immersion.

RSAC-5 calculates inhalation doses using the ICRP 30 [7.16] and DOE [7.17] dose conversion factors. The Committed Dose Equivalent (CDE) for each organ or tissue is multiplied by the appropriate weighting factor and summed to determine the Committed Effective Dose Equivalent (CEDE). Ingestion doses are calculated based on the models and equations of Regulatory Guide 1.109 [7.18]. The dose from radioactivity deposited on the ground surface is calculated using DOE [7.19] dose-rate conversion factors. Immersion doses are calculated using a finite plume model.

Table 7.6-1 provides the thyroid organ dose and the effective dose equivalent (applicable to the whole body) at distances of 100 meters (assumed INTEC boundary), 1000 meters, 13.7 km (INL boundary, controlled area boundary, and assumed MEI), and 18 km (Atomic City). Table 7.6-1 also provides the calculated $\chi/Q$ for each distance. Doses for the on-site locations are calculated assuming 40 hours per week of occupancy with no contribution from the ingestion pathway. The calculated annual dose at the INTEC boundary is 7 mrem, a small fraction of the 10 CFR Part 20 limits for either occupational exposure or dose to members of the public. The calculated effective dose equivalent at the INL boundary is well below the 25 mrem 10 CFR 72.104 limit for the whole body.

The estimated effluent dose equivalent rate for the MEI resulting from an unfiltered bounding DSC vent path is calculated to be 0.17 mrem/y [7.22]. The effective dose equivalent rate for INTEC boundary, calculated based on the ratio of the TEDE values at 100 meters and 13,700 meters documented in Table 7.6-1 and the same assumptions discussed above, is 130 mrem/y.

### 7.6.4 Liquid Release

Even though the HSM is provided with a drain to remove any moisture that may get into the HSM, no liquids are expected to be released from the TMI-2 ISFSI.

**Table 7.6-1**
**Estimated Effluent Dose Equivalents**
(rem per year, 100% occupancy for off-site locations)

| | | Distance (meters) | | | |
|---|---|---|---|---|---|
| **Location** | **Pathway** | **100** | **1,000** | **13,700** | **18,000** |
| | X/Q (s/m$^3$) | 5.06E-04 | 7.68E-06 | 6.14E-08 | 4.81E-08 |
| Thyroid | Inhalation (CDE) | 1.62E-06 | 2.26E-08 | 4.53E-10 | 3.35E-10 |
| | Ingestion (CDE) | 0.00E+00 | 0.00E+00 | 2.26E-08 | 1.67E-08 |
| | Ground Service (DE) | 8.66E-06 | 1.21E-07 | 3.03E-09 | 2.24E-09 |
| | **Total Organ Dose** | **1.03E-05** | **1.44E-07** | **2.61E-08** | **1.93E-08** |
| EDE (whole body) | Inhalation (CEDE) | 7.21E-03 | 1.08E-04 | 2.74E-06 | 2.13E-06 |
| | Ingestion (CEDE) | 0.00E+00 | 0.00E+00 | 6.51E-06 | 5.09E-06 |
| | Ground Surface (EDE) | 7.84E-06 | 1.09E-07 | 2.75E-09 | 2.03E-09 |
| | Cloud Gamma (EDE) | 4.36E-05 | 3.88E-06 | 3.24E-07 | 2.53E-07 |
| | **Total EDE** | **7.26E-03** | **1.12E-04** | **9.58E-06** | **7.48E-06** |

1) No ingestion exposure is assumed within the INL site boundary
2) On-site exposures calculated using 40 hours per week of occupancy

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## 7.7 References


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7.2 Title 10, “Energy,” Code of Federal Regulations, Part 72, “Licensing Requirements for the Independent Storage of Spent Nuclear Fuel and High-Level Radioactive Waste.”

7.3 “Safety Analysis Report for the Standardized NUHOMS® Horizontal Modular Storage System for Irradiated Nuclear Fuel,” VECTRA Technologies, Inc., Revision 4A, File Number NUH003.0103, 1996.

7.4 U. S. Nuclear Regulatory Commission, "Information Relevant to Ensuring that Occupational Radiation Exposures at Nuclear Power Stations Will be As Low As Reasonably Achievable," Regulatory Guide 8.8.

7.5 U. S. Nuclear Regulatory Commission, "Operating Philosophy for Maintaining Occupational Radiation Exposures As Low as is Reasonably Achievable," Regulatory Guide 8.10.

7.6 “ORIGEN2.1 - Isotope Generation and Depletion Code - Matrix Exponential Method,” CCC-371, Oak Ridge National Laboratory, RSIC Computer Code Collection, August 1991.

7.7 Staley, C. S., “Doses to Maximally Exposed Individuals due to Potential Airborne Releases from the INEL Storage of the TMI-2 Fuel Project,” EDF Serial Number EMA-96-001, File Number 219-02.0034, February 16, 1996.

7.8 American Nuclear Society Standards Committee Working Group ANS-6.1.1, "American National Standard Neutron and Gamma-Ray Flux-to-Dose-Rate Factors," ANSI/ANS-6.1.1-1977, American Nuclear Society, 1977.

7.9 Radiation Shielding Information Center, "CASK81 22 Neutron, 18 Gamma Ray Group, P3 Cross-Sections for Shipping Cask Analysis," DLC-23, June 1987.

7.10 “DORT-PC - Two-Dimensional Discrete Ordinates Transport Code System,” CCC-532, Oak Ridge National Laboratory, RSIC Computer Code Collection, October 1991.

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7.15 "Rancho Seco Independent Spent Fuel Storage Installation Safety Analysis Report," Sacramento Municipal Utility District, Docket No. 72-11.

7.16 ICRP, "Limits for Intakes of Radionuclides by Workers," Part 1, ICRP Publication 30, Pergamon Press, Oxford, Great Britain, 1979.

7.17 U.S. Department of Energy, "Internal Dose Conversion Factors for Calculation of Dose to the Public," DOE/EH-0071, Washington D.C., 1988.

7.18 U. S. Nuclear Regulatory Commission, "Calculation of Annual Doses to Man from Routine Releases of Reactor Effluents for the Purpose of Evaluating Compliance with 10 CFR Part 50, Appendix I," Regulatory Guide 1.109.

7.19 U.S. Department of Energy, "External Dose-Rate Conversion Factors for Calculation of Dose to the Public," DOE/EH-0070, Washington D.C., 1988.

7.20 Hall, G. G., "Impact of AmBeCm Sources on the TMI-2 ISFSI Design Basis", Engineering Design File No. 1793, Revision 4, March 15, 2001.

7.21 DOE/ID-12082. Idaho National Engineering and Environmental Laboratory Site Environmental Report, Calendar Year 2001. December 2002.

7.22 Hall, G. G., "Radiological Evaluation of TMI-2 ISFSI Technical Specification 3.1.1", Engineering Design File No. 4728, Revision 1, July, 2006.