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TABLES

FIGURES

7. RADIATION PROTECTION

7.1. Ensuring that Occupational Radiation Exposures are As Low As Reasonably Achievable (ALARA)

7.1.1. Policy Considerations

The design parameters and operating characteristics of the ISFSI allow doses to remain ALARA during operation. In order to maintain personnel exposures low, an ALARA Program will be implemented that includes the following key elements:

1. Design and procedural ALARA reviews by qualified staff and committees.

2. Pre and post job reviews including the establishment of person rem goals, planning for special tools, ventilation, shielding, services, and communications equipment.

3. Trending of radiological performance factors including worker exposures, personnel contamination, waste generation, and area contamination

4. ALARA committee reviews of selected activities including 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.

5. Training for selected jobs.

6. Management review of Radiation Protection Program effectiveness.

7.1.2. Design Considerations

The design of the MVDS complies with the requirements to maintain radiation exposure ALARA. The objectives of Regulatory Position 2 of Regulatory Guide 8.8 (Ref. 1) have been used for guidance and have been applied throughout the design of the MVDS.

The MVDS is designed to maintain the spent fuel within the FSC and provides bulk radiation shielding (i.e. the civil structure, CHM shielding, and the transfer cask) ensuring that the radiation exposures to personnel during MVDS operation are maintained ALARA, in addition to satisfying the requirements of 10 CFR Part 20 (Ref. 2).

Listed below are specific considerations to ensure that the annual dose to individual personnel and the annual collective dose to personnel are maintained ALARA.

1. The application of a system of maximum acceptable design dose rates, related to access requirements.

2. The provision of radiation shielding, designed using methods known to yield accurate results, and based on demonstrably conservative assumptions.

3. The MVDS is a passive system requiring minimal maintenance. Maintenance is based on a unit replacement philosophy. Component maintenance is performed in a low dose rate area. Wherever possible, operating mechanisms and drives requiring maintenance are positioned outside of the shielding envelope.

4. Radiation scatter and streaming are minimized by providing labyrinths and stepped streaming paths.

5. A prime consideration in the MVDS design is the provision of adequate access for personnel to perform operations carried out in radiation areas.

6. The use of existing transfer cask handling procedures and experience minimize radiation exposure and eliminate the spread of contamination

7.1.3. Operational Considerations

7.2. Operational considerations that reflect on ALARA are listed in the previous section.

7.2. Radiation Sources

7.2.1. Characterization of Sources

The design basis for the radiological assessment for the storage facility is for HTGR fuel that has been irradiated to 52,000 MWdays/MT of heavy metal and has decayed 600 days, as discussed in Section 3.1.1.2. The gamma and neutron source terms were generated with the ORIGEN-S computer code (Refs. 3 and 4) using the above burnup criteria and the actual initial fuel loading.

The spent fuel and any waste associated with the spent fuel are bounded by the source terms determined in the above mentioned analysis. The gamma and neutron source spectra are shown in Table 3.1-2. A description of the physical characteristics of the spent fuel is included in Table 3.1-1. The total gamma source strength is 2.97 E+14 photons/sec. The total neutron source strength is 3.31 E+05 neutrons/sec.

The neutron energy source spectrum includes both alpha-neutron and spontaneous fission sources. In addition to the intrinsic neutron sources discussed above, the facility is designed to accommodate the storage of Cf-252 neutron sources. The maximum design strength of the Cf-252 neutron sources is 4.000 E+05 micro Curies per source. No Cf-252 neutron sources are stored at the FSV ISFSI (see Section 1.1.1).

The gamma and neutron sources originating in the fuel are considered to be the primary sources. Secondary gammas generated in the shielding are considered in the shielding calculations. The shielding calculations are detailed in Appendix A7.1-1.

Tables 7.2-1 and 7.2-2 provide fuel element source characterization information.

7.2.2. Airborne Radioactive Material Sources

The design of the MVDS is such that in normal operation there are no sources of radioactive material that may become airborne.

Table 7.2-1. Total Gamma Source Terms for an Average Fuel Element.

Gamma Energy Boundaries (MeV)	(MeV) Mean	Gamma Spectra (photons/sec) 400 Day	600 Day	900 Day
4.0 - 3.5	3.75	3.03E+05	2.11E+05	1.24E+05
3.5 - 3.0	3.25	6.62E+08	4.54E+08	2.58E+08
3.0 - 2.6	2.80	1.50E+10	1.45E+10	1.45E+10
2.6 - 2.2	2.40	4.45E+10	2.90E+10	1.53E+10
2.2 - 1.8	2.00	2.34E+12	1.44E+12	6.95E+11
1.8 - 1.34	1.57	4.02E+12	3.08E+12	2.14E+12
1.34 - 0.92	1.13	9.06E+12	7.47E+12	5.90E+12
0.92 - 0.38	0.65	3.47E+14	2.68E+14	2.11E+14
0.38 - 0.22	0.30	2.57E+13	1.72E+13	1.03E+13
0.22 - 0.12	0.17	8.92E+08	8.93E+08	9.82E+08
Total		3.88E+14	2.97E+14	2.30E+14

Table 7.2-2. Total Neutron Source Terms for an Average Fuel Element.

Neutron Energy Boundaries (MeV)	(MeV) Mean	Neutron Spectra (neutrons/sec) 400 Day	600 Day	900 Day
20.0 - 6.43	13.22	5.41E+3	5.14E+3	4.89E+3
6.43 - 3.00	4.72	7.26E+4	6.89E+4	6.59E+4
3.00 - 1.85	2.43	9.74E+4	9.36E+4	9.07E+4
1.85 - 1.40	1.63	4.59E+4	4.40E+4	4.24E+4
1.40 - 0.90	1.15	5.60E+4	5.34E+4	5.12E+4
0.90 - 0.40	0.65	5.77E+4	5.48E+4	5.23E+4
0.40 - 0.10	0.25	1.12E+4	1.07E+4	1.02E+4
Total		3.46E+5	3.31E+5	3.18E+5

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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 are ALARA and that a high degree of integrity is obtained for the confinement of radioactive materials.

7.3.1.1. Design Features Relevant to Maintaining Exposures ALARA

The applicable design feature guidance given in Regulatory Position 2 of Regulatory Guide 8.8 (Ref. 1) is discussed below.

7.3.1.1.1. Access Control of Radiation Areas

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Access to the ISFSI is controlled in accordance with 10 CFR Part 72 (Ref. 5) (shown in Figure 4.0-1).

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Normal access to the MVDS is through a single access point that includes a Health Physics Control Station when radiological conditions warrant.

7.3.1.1.2. Radiation Shields and Geometry

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Fixed radiation shielding constitutes the primary method of reducing personnel exposure to radiation.

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The design of radiation shielding is based on the results of shielding analyses, which are discussed in detail in Appendix A7.1-1. Calculation methods known to provide reliable and accurate results have been employed, including Point Kernel Integration and Monte Carlo analyses. The application of these methods to specific shield designs and the verification of the computer codes used are fully described in Appendix A7.1-1.

Conservative assumptions have been used throughout. In particular, radiation exposure is calculated on the bases that all spent fuel elements handled and stored have the average source strengths, defined in Sections 3.1.1.3 and A7.1.2 which is based on the fuel segment with the highest heat generation rate at 600 days after shutdown. Integrated operator exposures are therefore maximum values, although higher local dose rates from groups of higher rated individual fuel elements could occur.

The main components of the shielding system shown in Figures 1.1-1, 1.1-2 and 1.1-3 are as follows:

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a. The transfer cask.

b. The CLUP and its isolation valve (including a cask shield ring) at the TCRB, which maintains the shielding envelope while a FSC containing the spent fuel is lowered from the CHM into the transfer cask.

c. The CHM bulk shielding, which provides a massive radiation barrier surrounding the spent fuel within the CHM.

d. The vault module charge face structure, in which the tops of the FSCs are located. Stepped streaming paths and the charge face shield plugs maintain the shielding integrity of the composite steel and concrete filled charge face structure.

e. The massive concrete vault civil structure surrounding the array of FSCs. The cooling air inlet duct provides a multiple scatter path for radiation from spent fuel stored in the FSCs, the shielding effectiveness of which is enhanced by concrete radiation collimating slabs. In the same way, collimating slabs at the outlet duct, combined with its dog-leg geometry, maintain the integrity of the shielding envelope surrounding the FSC array.

7.3.1.1.3. Control of Airborne Contaminants and Gaseous Radiation Sources

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The MVDS has no sources of airborne contaminants or gaseous radiation during normal operation, therefore no specific control systems are required. All radiation sources except for sealed instrument check sources are contained within the sealed FSCs. The radioactive sources are described in Section 7.2.1.

The spent fuel was sealed in FSCs at the Reactor Building. This containment prevented the spread of contamination during handling and storage at the MVDS.

7.3.1.1.4. Provisions for Spent Fuel Transfer to the MVDS

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Transfer of spent fuel to the MVDS used the existing transfer cask and procedures. This permitted the direct application of previous operational experience in controlling radiation exposure and spread of contamination, maintaining the ALARA principle as it relates to these operations.

7.3.1.2. Installation Layout

The general layout of the MVDS is shown in Figures 1.1-1, 1.1-2 and 1.1-3. The radiation zone designations applied to specific areas and the location of areas and equipment that are relevant to the radiological protection program are indicated in Table 3.3-2. The allocation of radiation zone designations to these areas is shown in Table 3.3-3.

7.3.2. Shielding

The design of shielding for each of the radiation sources identified in Section 7.2 is described in Appendix A7.1-1.

7.3.3. FSC Integrity

DOE commits to performing a leak test of the interspace between the double metal ‘O’ ring on six (6) FSCs at five year intervals following initial loading of the MVDS with spent fuel in accordance with the ISFSI Technical Specifications.

7.4. Estimated Onsite Collective Dose Assessment

The MVDS is surrounded by an ISFSI fence and access to the MVDS is only possible via the security access control point. Access is primarily restricted to personnel required for inspection, monitoring, and maintenance activities and in the longer time scale for fuel unloading and decommissioning activities.

During the inspection/monitoring/maintenance phase of operations of the MVDS only infrequent access to the facility will be required. The dose rates prevailing in the key facility areas for fuel decay periods of 600 days and 900 days are presented in Table 7.4-1.

General area radiation and contamination levels inside the charge face have historically been less than 0.2 mrem/hour and less than 1,000 dpm/100 cm^2, respectively. Consequently, individual and collective annual occupational radiation exposure during normal operation and maintenance has been, and will continue to be, less than 10 mrem.

During March 1996 FSC seal integrity verification, the localized radiation level above each FSC undergoing testing increased to less than 5 mrem/hour. The increased radiation level occurs only during the test period because of the changed configuration of the shield plug. Contamination levels remained less than 1,000 dpm/100 cm^2. Individual and collective annual occupational radiation exposure received during the FSC seal integrity verification and projected during subsequent verifications was, and will continue to be, less than 10 mrem.

Actions to be performed by workers at the MVDS for defueling operations are identified in Section 5.1.1. Since the time at which the spent fuel will be transferred out of the ISFSI is not known, a detailed dose assessment is not provided. The 6.5 person-mrem average actual exposure to workers at the MVDS per FSC transfer during fuel loading operations provides a reasonably conservative exposure estimate for defueling operations. Using this value, transfer of 244 FSCs from the ISFSI would result in total exposures to workers at the MVDS of approximately 1.59 person-rem.

Table 7.4-1. Dose Rates at MVDS after Completion of Fuel Loading.

Area	Dose Rate (mrem/h) at Decay Periods	
	600 Day	900 Day
Controlled area boundary fence (100m)	0.02	0.02
Surface of outlet duct wall	2.0	1.5
Vault wall inside TCRB	6.9	5.3
Surface of outer TCRB wall	1.0	0.8
Surface of storage well wall	2.8	2.2
Inlet duct mesh	21.0	16.0
Inlet duct exclusion fence	1.3	1.0

7.5. Estimated Offsite Collective Dose Assessment

The design features described in Section 7.3.1 ensure that, during normal operation, there are no effluent streams from the MVDS. The only offsite dose due to the MVDS installation, therefore, will be from direct and scatter radiation.

The variation of calculated dose rate with distance from the MVDS installation is shown graphically in Figure 7.5-1. The dose rate is highest adjacent to the cooling air inlet structure (east side). However, at the controlled area boundary the dose rate is highest on the west side. The dose rates for other directions are lower. This effect is due to the layout of the outlet duct, TCRB, inlet duct and storage wells.

There are five significant contributions to the controlled area boundary dose rate, arising from:

1. Direct radiation through the vault module shield walls.

2. Direct radiation from a FSC resident in the CHM (short term).

3. Scattered radiation from the beam emerging from the inlet and outlet ducts.

4. Scatter of radiation arising from the charge face.

5. Direct radiation from a FSC resident in the transfer cask in the TCRB (short term).

All five contributions were analyzed using the Point Kernel Integration code RANKERN, which has the facility to perform both direct penetration and scatter/skyshine calculations. (See Appendix A7.1-1).

The conservative dose rates shown in Figure 7.5-1 make no allowance for personnel occupancy factors and are based on 600 day decay spent fuel. The maximum dose rate to the nearest resident [797meters (2,600 feet)] is 0.15 mrem/year.

The variation of the off-site dose rates, at distances up to 950 meters (3,100 feet), are presented in Figures 7.5-1(a), 7.5-l(b) and 7.5-1(c). The dose rates are presented assuming a 10% occupancy factor for a member of the public, which is appropriate for this rural location.

The dose rates have been assessed based on actual fuel irradiation parameters and based on a reactor shutdown in September 1989 and fuel loading completion in July 1992.

Figure 7.5-1(a) presents the dose rates at the controlled area boundary on the west side of the MVDS. These annual dose rates are for the first year of operation.

Figures 7.5-1(b) and 7.5-1(c) are to be used in combination to provide the total annual dose rates for the north, south, east and west sides of the MVDS as a function of both distance (0 to 950m) and storage time (0 to 50 years). The first figure presents the dose rate against distance for the four sides, based on a hypothetical assumption that the complete MVDS is loaded instantaneously with fuel at 760 days decay. The second figure presents factors by which the north/south and east/west dose rates should be multiplied to give the average annual dose rate for storage periods up to 50 years. The two figures must only be utilized in this manner.

The maximum annual dose rate to a member of the public at the controlled area boundary is 13 mrem/year and occurs on the west boundary. The maximum annual dose rate at the controlled area boundary, averaged over all four sides of the MVDS, is 4 mrem/year. Both of these dose rates are based on a 10% occupancy factor.

No real individual member of the public located beyond the controlled area boundary will receive a dose rate in excess of 25 mrem/year, as required in 10 CFR 72.104. The maximum and averaged annual dose rates at the controlled area boundary are 13 mrem/year and 4 mrem/year respectively. Based on the occupancy levels of 10%, these controlled area boundary radiation levels are well within the regulatory requirement and consistent with the ALARA philosophy.

The calculation for the FSV off-site dose rate involves many conservative assumptions, which include:

a. All fuel elements were assumed to be from the hottest segment.

b. The assumed fuel burnup is much higher than the actual value. The calculation assumed operation through 300 Effective Full Power Days (EFPD), plus a coastdown period of 170 EFPD. The actual fuel burnup is 232 EFPD, which is 238 EFPD less than the value used in the calculations.

c. A conservative peaking factor of 1.76 was used.

d. 10% public occupancy.

e. Maximum dose rate with no azimuthal averaging.

f. The presence of six Cf-252 neutron source elements loaded in the ISFSI. As explained in Section 1.1.1, Cf-252 neutron sources were removed from the fuel elements before the elements were transferred to the ISFSI.

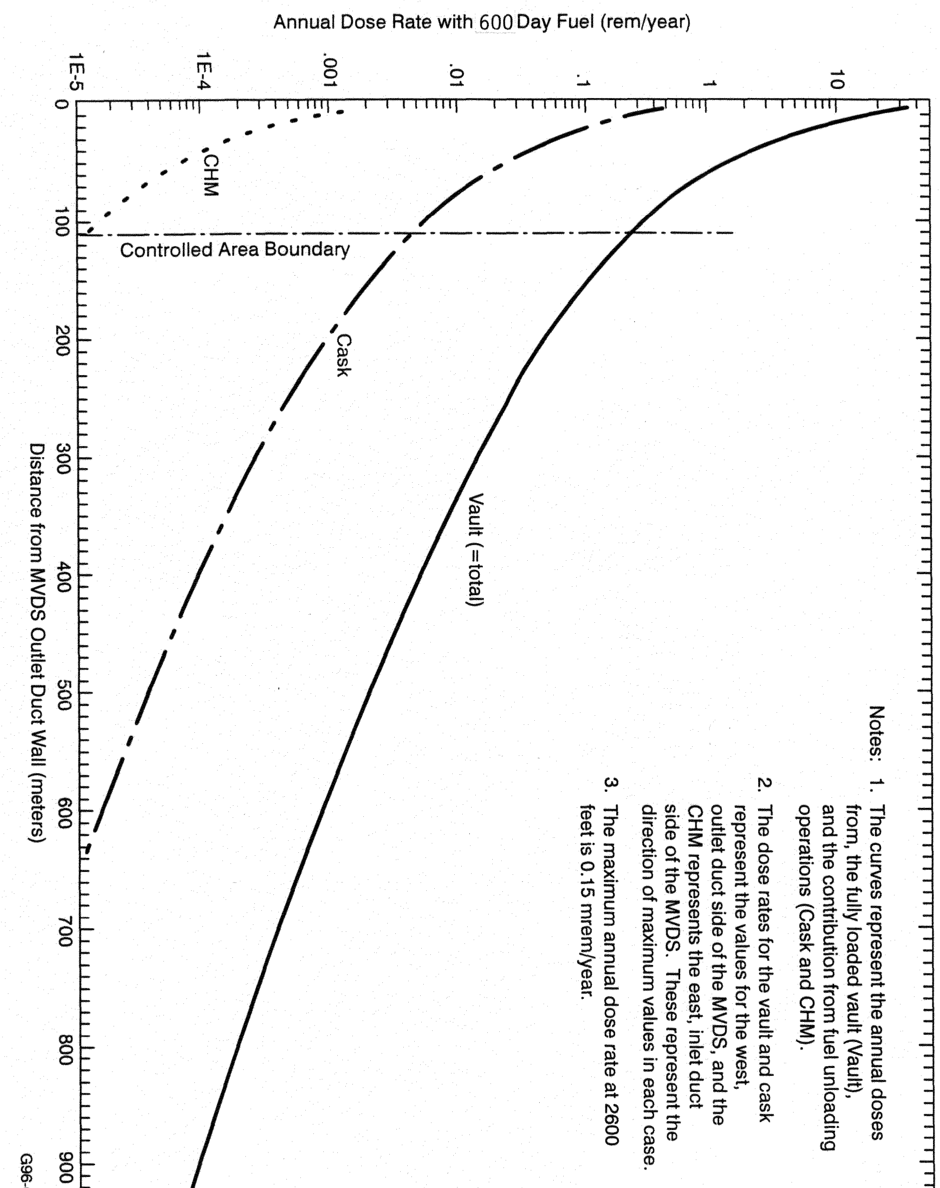

Figure 7.5-1. Variation of Dose Rate with Distance from the MVDS.

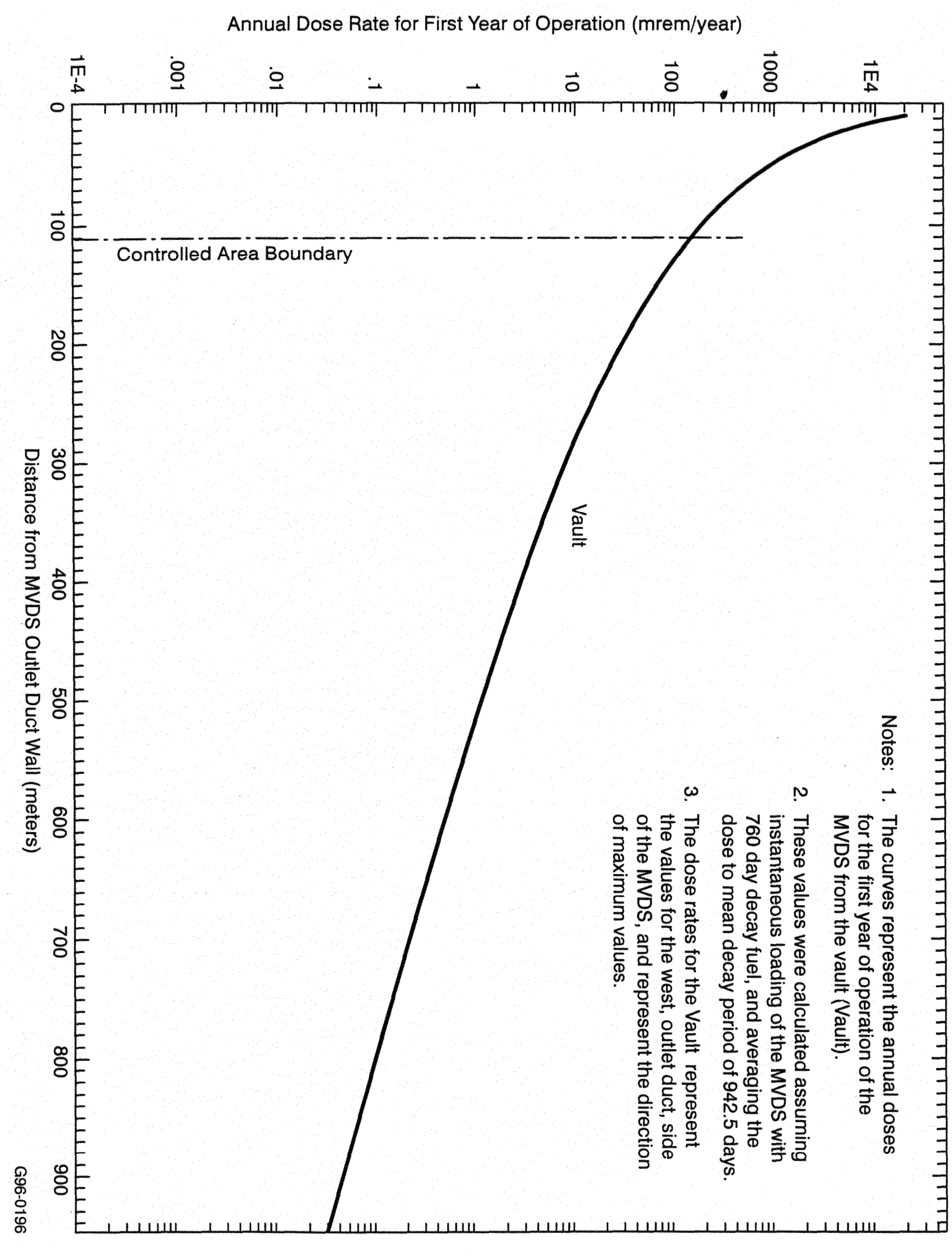

Figure 7.5-1(a). Variation of Dose Rate with Distance from the MVDS.

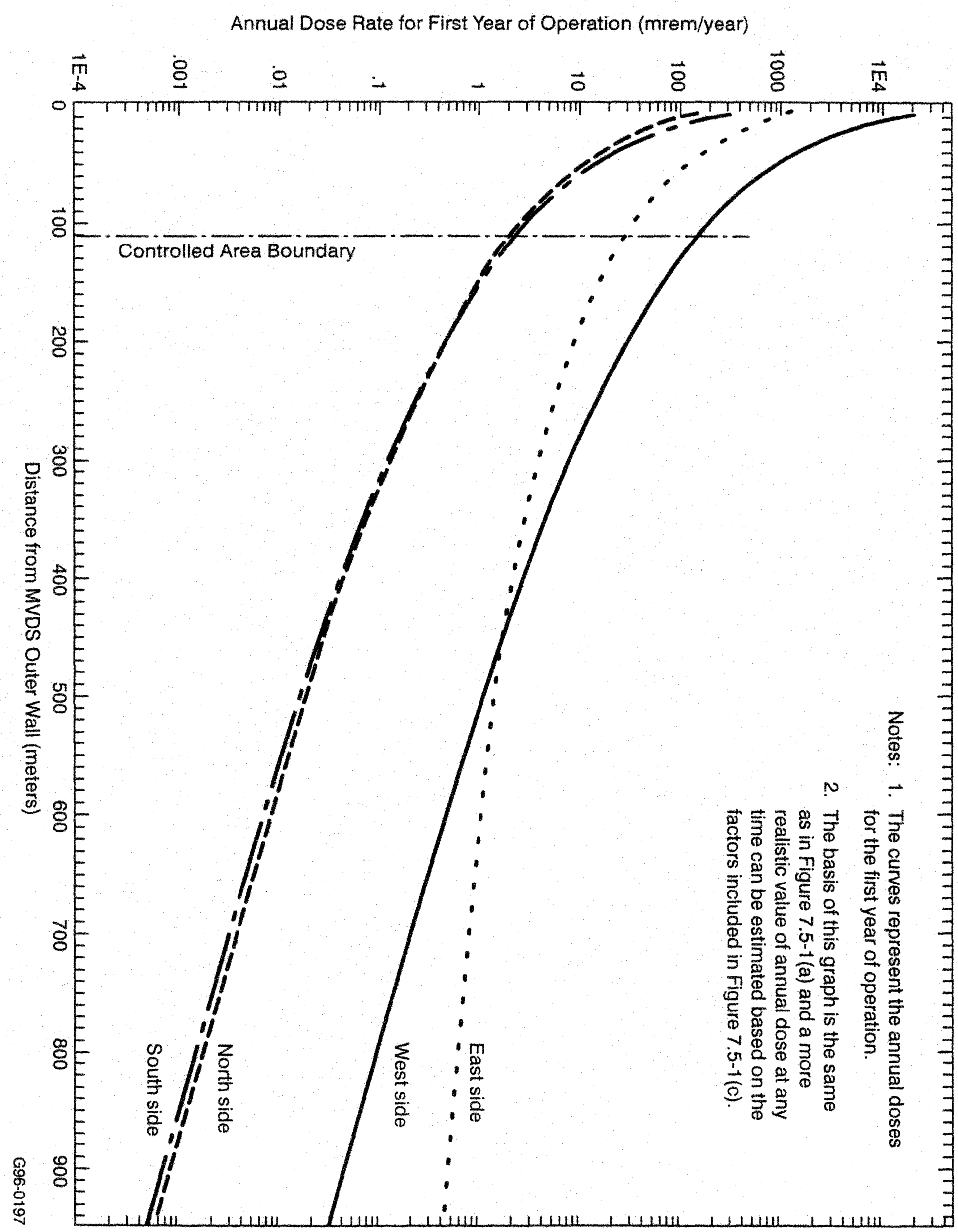

Figure 7.5-1(b). Variation of Dose Rate with Distance from the MVDS.

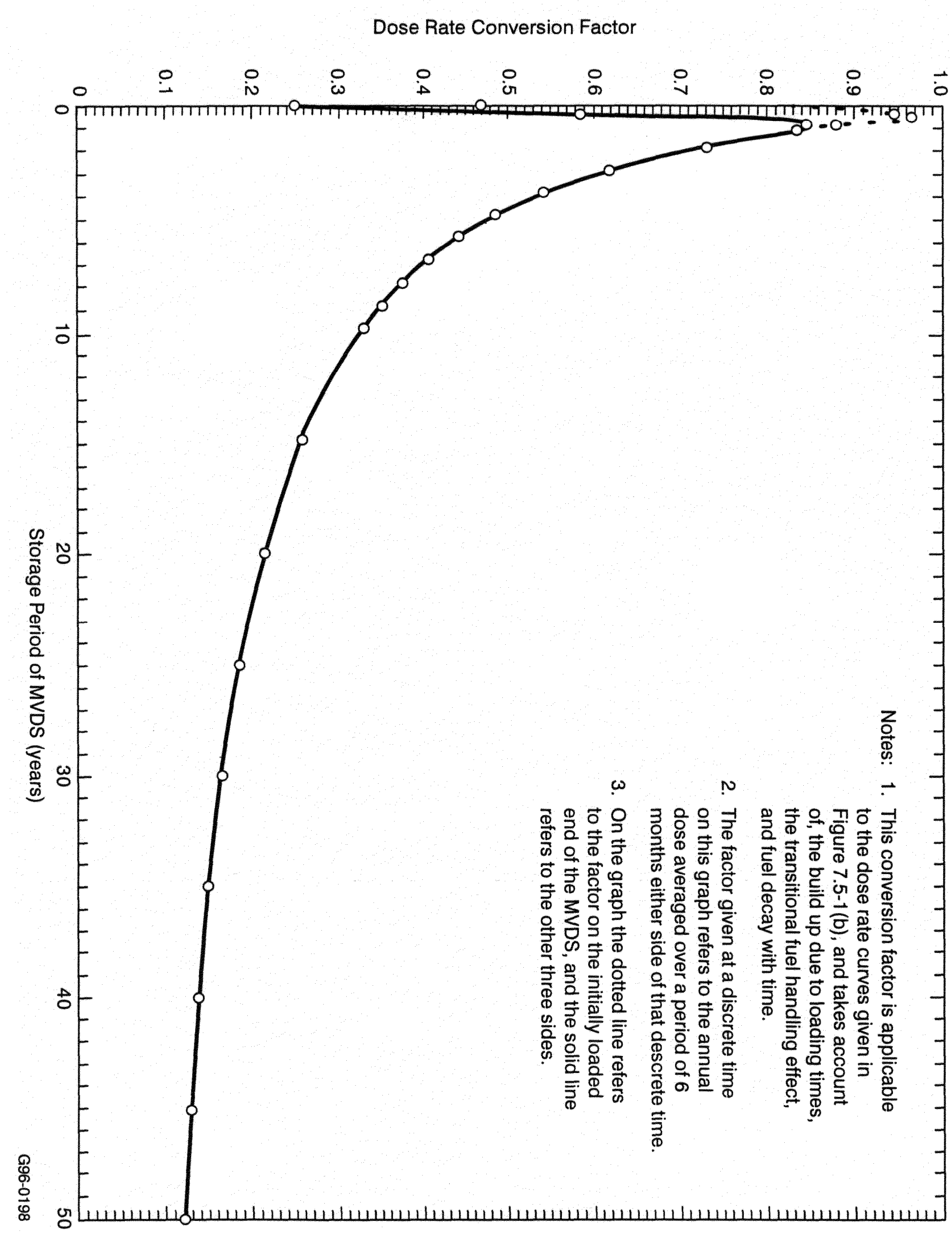

Figure 7.5-1(c). Conversion Factor for Storage Time at the MVDS.

7.6. Health Physics Program

This section comprises DOE's FSV ISFSI Occupational Radiation Protection Program. The procedural requirements of this FSV ISFSI Occupational Radiological Protection Program are implemented by contractor procedures.

7.6.1. This Health Physics Program is implemented by the contractor Management Control Procedures for Radiation Protection developed for use at the INL and by Technical Procedures developed specifically for use at the FSV ISFSI. The effectiveness and implementation of the FSV ISFSI Radiation Protection Program is reviewed annually by DOE or it's contractor. Organization

The FSO is responsible for all radiation protection activities at the ISFSI with implementation performed by contractor personnel or subcontractors working under the direction of the contractor in accordance with approved procedures.The FSV FSO acts as the ISFSI representative on the INTEC ALARA Committee.Radiation safety issues that arise are brought to the attention of the INTEC ALARA Committee and the ISFSI Safety Review Committee by the FSO. Stop work authority is granted to any individual who observes an actual or potential unsafe radiological condition.

7.6.2. Equipment

Sufficient numbers and types of radiation surveillance instrumentation and other general radiological control supplies such as anti-contamination clothing, decontamination materials, etc., are available to perform assessment of radiological conditions and perform normally expected duties during routine activities at the ISFSI.If non-routine, unplanned, or otherwise large scale maintenance or surveillance activities are required, adequate equipment will be obtained through the INL.

7.6.3. Administrative Limits

7.6.3.1. Occupational Radiation Exposure

Occupational radiation exposure at the FSV ISFSI is controlled in accordance with 10 CFR 20, Subpart C, Occupational Dose Limits. With the exception of planned special exposure, occupational radiation exposure is administratively limited to 20% of the limits specified in 10 CFR 20. The need for planned special exposure at the FSV ISFSI is not anticipated.

7.6.3.2. Radiation Exposure to Public

Radiation exposure to individual members of the public is limited to an combined annual dose equivalent of 25 mrem in accordance with 40 CFR 190.03, which also achieves the dose limitation of 40 CFR 72.104. The results of the FSV ISFSI Radiological Environmental Monitoring Program are used to verify compliance with the exposure limit.

7.6.3.3. Control of Exposure from External Sources

During routine operations at the FSV ISFSI, there are no personnel access points to High Radiation Areas (HRA) and Very High Radiation Areas (VHRA) where posting and access control of such areas is warranted. Any HRA's and VHRA's created during fuel storage container (FSC) movement will be posted and access controlled in accordance with 10 CFR 20.1601 and 20.1602 respectively.

7.6.3.4. Respiratory Protection and Controls to Restrict Internal Exposure

Process and engineering controls inherent in the design of the FSV ISFSI minimize the potential for generation of airborne radioactivity. During FSC movement and FSC seal integrity checks, the potential for intake of airborne radioactivity is minimized through HEPA filtration and/or containment of breached air volumes. Access control, exposure time limitation, and use of respiratory protection equipment are secondary means for maintaining the Total Effective Dose Equivalent (TEDE) ALARA. Use of respiratory protection equipment is performed in accordance with 10 CFR 20.1703 through services provided by the INL.

7.6.4. Procedures

Procedures and engineering controls are used, to the extent practicable and based upon sound radiation protection principles, to achieve occupational radiation doses and radiation doses to members of the public that are ALARA.

7.6.4.1. Surveys and Monitoring

Internal and external occupational radiation exposure monitoring thresholds specified in 10 CFR 20.1502 are not anticipated to be exceeded during routine operation of the FSV ISFSI. However, capability for providing such monitoring is maintained.

Tritium monitoring has been instituted within the ISFSI as a means of monitoring the effects of facility aging during the prolonged storage period and future fuel movements. Such tritium monitoring is performed on the ISFSI charge face and in each of the six ISFSI chimneys. The charge face monitoring is intended to detect gross failures related to the FSC O-ring seals while the chimney monitoring is intended to detect failures related to the FSCs themselves. Desiccant samples installed on the charge face and in the chimneys are retrieved periodically and the absorbed atmospheric water vapor is analyzed for tritium.

Radiation surveys are performed periodically in the FSV ISFSI and within the 100 meter perimeter fence. Contamination surveys are performed periodically in the ISFSI. Radiation and contamination monitoring instrumentation is calibrated periodically for the types of radiation measured.

When occupational radiation exposure monitoring is required, direct reading and/or electronic dosimetry is utilized. Radiation and contamination surveys in the ISFSI are used to confirm when the occupational radiation exposures are low enough that personal radiation dosimetry is not required. Personnel thermoluminescent dosimetry, when used, is processed at the INL by a DOELAP accredited processor. DOELAP accreditation is considered a NVLAP accreditation equivalent for the radiation performance categories that approximate the radiation environment at the FSV ISFSI. An exemption from the NVLAP requirement in 10 CFR 20.1501(c) was granted by the NRC on March 12, 1999 and made effective upon the transfer of the Materials License SNM-2504 to DOE.

All material leaving a posted contamination area is surveyed prior to unrestricted release to uncontrolled areas.

7.6.4.2. Storage and Control of Licensed Material

Licensed material at the FSV ISFSI is controlled in accordance with 10 CFR 20.1801 and 20.1802.

7.6.4.3. Posting, Labeling, and Receipt of Licensed Material

Radiological areas characterized as areas defined in 10 CFR 20.1003 are posted and controlled in accordance with 10 CFR 20.1901 through 20.1903.

Containers of licensed material are labeled and controlled in accordance with 10 CFR 20.1904 and 20.1905.

Receipt and opening of packages containing quantities of radioactive material exceeding Type A quantities is performed in accordance with 10 CFR 20.1906.

7.6.4.4. Control of Radioactive Material Other Than Spent Fuel

The NRC amended the ISFSI license to authorize receipt, possession, storage and the transfer of certain radioactive materials at the ISFSI other than spent fuel (Ref. 6). These radioactive materials include radioactive sources (calibration disks or sealed sources which are used for sample analyses and instrument calibrations), low-level radioactive waste, contaminated equipment items, and the depleted uranium plugs (DUPs), all associated with spent fuel storage activities.

The radioactive sources are controlled in accordance with the requirements in 10 CFR 20. The sources are stored in a fire-rated storage cabinet located at the south end of the ISFSI charge face. The cabinet is locked when unattended, with keys controlled by the Facility Safety Officer. It is restrained in a manner such that it will not topple in the event of a design basis earthquake. The radioactive sources, or packaging containing the sources, are labeled and the cabinet posted in accordance with 10 CFR 20 requirements. Radiation protection personnel are responsible for receipt, storage, use, inventory, performance of source leak checks, and final disposition of sources, in accordance with approved procedures. Radiation Protection personnel that use the sources are trained in their proper handling.

The ISFSI License, as amended (Ref. 6), provides for the receipt, possession, storage and transfer of low-level radioactive waste and contaminated equipment/materials associated with spent fuel storage activities. Radioactive waste generated during maintenance, surveillance, defueling or decommissioning operations is expected to consist primarily of dry radioactive waste such as rags or paper wipes, and anti-contamination clothing. The waste will be packaged in 55 gallon drums and temporarily stored at the ISFSI while awaiting shipment for disposal. Staging of low-level waste in the transfer cask reception bay is not permitted when a cask containing spent fuel is in the cask load/unload port, in order to assure a potential fire does not impact a loaded cask. In addition, staging of low-level waste is not permitted within 20 ft. of the DUPs, as discussed below. Section 8.2.1 postulates that a tornado breaches drums of low-level waste, and assesses dose consequences. Section 8.2.3 assesses effects of postulated drop of contaminated equipment.

Three DUPs, necessary for spent fuel transfer operations, are stored on the ISFSI charge face. A DUP is positioned on the lid of a FSC when the FSC is placed in the transfer cask. This assures adequate shielding to workers performing operations on top of the transfer cask, such as removal of the cask closure. The depleted uranium is radioactive, with dose rates of approximately 10 mrem/hour and 3 mrem/hour on contact and at 12 inches respectively, and the DUP storage areas are posted in accordance with 10 CFR 20 requirements. In the event the DUPs could be exposed to a fire, no significant release of U-238 would be anticipated since uranium metal is pyrophoric only when finely divided. The DUPs consist of solid, machined, depleted uranium that is nickel plated. However, in order to reduce the potential for a low-level radioactive waste fire from involving the DUPs, a minimum of 20 ft. horizontal separation is required between the DUPs and any drums containing low-level waste stored at the ISFSI (Ref. 7).

7.6.4.5. Waste Disposal

Licensed material designated as radioactive waste is presently neither generated nor disposed at the FSV ISFSI. Temporary on-site storage of such material will be the interim option until disposal procedure approval is obtained.

The need for discharge of licensed material into sanitary sewerage is not anticipated.

If and when transfer of low-level radioactive waste for disposal at a land disposal facility becomes necessary, a shipping manifest tracking system will be established in accordance with 10 CFR 20.2006. Control of such transfers will also be implemented in accordance with 10 CFR 20.2006.

7.6.4.6. Records

Records of the provisions of the FSV ISFSI Radiation Protection Program; survey results used in the determination of dose from external radiation sources; measurement results and calculations used to determine individual intakes of radioactive material and used in the assessment of internal dose; results of air sampling, surveys, and bioassays to substantiate the respiratory protection program; and measurement results and calculations used to evaluate the release of radioactive effluents to the environment are maintained for the duration of the license.

Records of audits and reviews of the program content and implementation, radiological surveys, and radiological monitoring equipment calibrations are maintained for a minimum of three years.

Prior occupational dose for individuals who are likely to exceed annual occupational exposure above required monitoring thresholds while employed at the FSV ISFSI is determined and documented in accordance with 10 CFR 20.2104. Information documented on the NRC Form 4 or equivalent is maintained for the duration of the license. Records used in preparing the NRC Form 4 are maintained for a minimum of three years.

Records of individual occupational radiation monitoring results, planned special exposures, and exposure to individual members of the public will be documented and controlled in accordance with 10 CFR 20.2105, 20.2106, and 20.2107 and maintained for the duration of the license.

Records of licensed material disposal will be retained for the duration of the license.

7.6.4.7. Reports

The theft or loss of licensed material will be reported in accordance with 10 CFR 20.2201.

Any event involving byproduct, source, or special nuclear material in possession at the FSV ISFSI that may have caused, or threatens to cause, exposure and intake thresholds established in 10 CFR 20.2202(a) to be exceeded will be reported immediately.

Any event involving loss of control of licensed material in possession at the FSV ISFSI that may have caused, or threatens to cause, exposure and intake thresholds established in 10 CFR 20.2202(b) to be exceeded will be reported within 24 hours of discovery of the event.

Any event resulting in exposure, radiation level, or radioactive material concentration thresholds in 10 CFR 20.2203 to be exceeded will be reported within 30 days after learning of the event.

The occurrence of any planned special exposure will be reported within 30 days following the exposure.

Prior to April 30 of each year, an annual report for the preceeding year of the results of individual monitoring carried out at the FSV ISFSI for each individual for whom monitoring was required will be submitted in accordance with 10 CFR 20.2206.

7.6.5. Occupational Radiation Exposure Monitoring

A DOELAP accredited thremoluminescence dosimetry (TLD) processor will be used to process TLDs worn by workers at the FSV ISFSI. The processor will have demonstrated successful proficiency testing in those radiation performance categories that approximate the radiation environment at the FSV ISFSI.

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7.7. Radiological Environmental Monitoring Program

No planned radioactive effluents are released from the FSV ISFSI during storage conditions. The only offsite dose due to the FSV ISFSI is from direct and scattered radiation. Environmental monitoring is performed to detect unplanned radioactive effluents from the FSV ISFSI.

This section of the SAR comprises DOE's FSV ISFSI Radiological Environmental Monitoring Program (REMP).

The REMP is conducted to provide data on levels of radiation and radioactive material in the site environs. The program discriminates between those changes in environmental radiation and radioactivity levels resulting from radioactive releases from the FSV ISFSI and those changes attributed to other sources such as worldwide fallout from weapons testing. The results of this program are used to verify the effectiveness of measures applied to prevent the release of radioactive materials. A preoperational radiological environmental monitoring program for the FSV ISFSI environs was conducted by Colorado State University. Operational data will continue to be compared to the preoperational data to characterize the environmental impact of FSV ISFSI operation.

Additional monitoring in the vicinity of the facility may be conducted or coordinated by other organizations, notably the Colorado Department of Public Health and Environment.

7.7.1. Organization

The Facility Safety Officer will oversee the conduct of the REMP Sampling

Radiological environmental monitoring dosimeters will be collected from the specific locations given in Table 7.7-1.

7.7.2. Processing

Radiological environmental monitoring dosimeters will be processed pursuant to the requirements of Table 7.7-1 with a demonstrated minimum detectable dose of no greater than 10 mrem.

7.7.3. Annual Reports

An Annual Radiological Environmental Monitoring Report shall be submitted by DOE-ID within 60 days of January 1 of each year. The report shall include summaries, interpretations, and analyses or trends of the results of the REMP for the reporting period, the quantity of each of the principal radionuclides released to the environment in liquid and gaseous effluents in the previous 12 months, and any other information required to estimate the maximum potential dose commitment to the public resulting from effluent releases and direct radiation exposure.

Each radiological environmental monitoring report shall contain a map that presents detailed information regarding monitoring station locations. Deviations are permitted from the required monitoring schedule if dosimeters are unobtainable due to hazardous conditions, seasonal unavailability, schedule conflict with unforseen fuel handling operations, and other legitimate reasons. If dosimeters are unobtainable due to equipment malfunction, every effort shall be made to complete corrective action before the end of the next monitoring period.

All changes in monitoring station locations and schedule that occur through the year shall be explained in each annual report.

Table 7.7-1. Radiological Environmental Monitoring Program.

Exposure Pathway	Number of Monitoring Stations and Locations	Dosimeter Collection Frequency	Type and Frequency of Processing
Direct and scattered radiation	16 stations with one or more dosimeters to be placed as follows: a ring of stations around the controlled area boundary; others may be placed in special interest areas such as population centers, nearby residences, and schools as desired. One control station with one or more dosimeters at a location at least 10 miles from the FSV ISFSI	Quarterly Approximately one third of the dosimeters will be checked and replaced each month.	Gamma exposure rate on each dosimeter collected.

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7.8. References

1. "Information Relevant to Insuring that Occupational Radiation Exposures at Nuclear Power Stations will be As Low As Is Reasonably Achievable," USNRC Regulatory Guide 8.8, Revision 3, June 1, 1978.

2. 10 CFR Part 20, "Standards for Protection Against Radiation."

3. NUREG/CR-0200, Volume 2, Section F7, "ORIGEN-S, Scale System Module to Calculate Fuel Depletion, Actinide Transmutation, Fission Product Buildup and Decay, and Associated Radiation Source Terms," Oak Ridge National Laboratory, Oak Ridge, Tennessee, December, 1984.

4. Engineering Evaluation, "Use of Origin-S Computer Code for Fort St. Vrain Applications," EE-DEC-0026, Rev. A, September 25, 1989.

5. 10 CFR Part 72, "Licensing requirements for the independent storage of spent nuclear fuel and high-level radioactive waste, and reactor-related greater than Class C waste."

6. NRC letter dated March 21, 1996 (G-96020), Travers to Crawford; Subject: "Organizational Changes and Revised Possession Limits."

7. PSCo letter dated December 12, 1995 (P-95103), Crawford to Travers (NRC); Subject: "Request License Amendment to Permit Certain Radioactive Materials at the FSV ISFSI, Necessary to Conduct Licensed Activities."

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