Private Fuel Storage, LLC

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U.S. Nuclear Regulatory Commission
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August 31, 2000

LICENSE APPLICATION AMENDMENT No. 17
DOCKET NO. 72-22/TAC NO. L22462
PRIVATE FUEL STORAGE FACILITY
PRIVATE FUEL STORAGE L.L.C.

This letter submits Amendment No. 17 to the Private Fuel Storage Facility (PFSF) License Application. This amendment updates the PFSF Safety Analysis Report (SAR), Environmental Report (ER), Emergency Plan (EP), and License Application (LA) to remove discussion of BNFL Fuel Solutions Corporation's (BFS) TranStor systems, since BFS has withdrawn their application for licensing of the TranStor shipping and storage systems. In those infrequent instances where reference continues to be made to information that was associated with TranStor, it is characterized as a representative spent fuel storage system. Changes to the Proposed Technical Specifications (Appendix A of the LA) were also made to address several comments received from the NRC in telephone conversations between the NRC (Mark Delligatti and Marissa Bailey) and PFS personnel that took place on August 18 and 21, 2000. Other miscellaneous changes have been made to the SAR, ER, EP, and LA for clarity, accuracy and completeness.

If you have any questions regarding this submittal, please contact me at 608-787-1236 or Mr. J. L. Donnell, Project Director, at 303-741-7009.

Sincerely,

John D. Parkyn, Chairman
Private Fuel Storage L.L.C.

JDP:JRJ
Enclosure

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PREFACE

PRIVATE FUEL STORAGE FACILITY

LICENSE APPLICATION

AMENDMENT 17

Enclosed are the following revisions to the Private Fuel Storage Facility License Application documents:

Safety Analysis Report – Revision 17

Environmental Report – Revision 11

Emergency Plan – Revision 10

License Application – Revision 10

PRIVATE FUEL STORAGE FACILITY

LICENSE APPLICATION AMENDMENT NO. 17

REPLACEMENT PAGE INSTRUCTIONS

With the exception of Chapter 7 of the Safety Analysis Report (SAR), remove individual pages and replace with the revised pages enclosed using the updated document control lists for guidance. Due to the large fraction of Chapter 7 of the SAR that is being revised by this Amendment No. 17, the entire Chapter is enclosed, including those pages that have not been revised. Therefore, for SAR Chapter 7, simply remove the existing Chapter in its entirety, including tables and figures, and replace it with the enclosed Chapter 7.

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CHAPTER 1

INTRODUCTION AND GENERAL DESCRIPTION OF FACILITY

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1.2 GENERAL DESCRIPTION OF FACILITY

The PFSF is designed to store up to 40,000 Metric Tons of Uranium [2](MTU) of spent fuel from U.S. commercial power reactors in sealed metal canisters (approximately 4,000 storage casks). A detailed description of the fuel types that can be stored is provided in Chapter 3 and 10. The canister-based spent fuel storage system selected for use at the PFSF utilizes sealed metal canisters to store multiple spent fuel assemblies. Each canister is placed inside of a storage cask. The storage system is passive and relies on natural convection for cooling. The system is an integral part of the facility "Start Clean / Stay Clean" philosophy which eliminates the need to handle individual fuel assemblies at the site. The system assures there is negligible contamination or radioactive waste generated at the site and facilitates the ease of decommissioning at the end of the life of the facility. Design criteria are described in more detail in Chapter 3.

The passive nature of the storage systems results in a relatively simple facility as shown on Figure 1.2-1. The Restricted Area (RA), is approximately 99 acres and is surrounded by a chain link security fence and an outer chain link nuisance fence with an isolation zone and intrusion detection system between the two fences. The cask storage area within the RA is surfaced with compacted gravel that slopes slightly to allow for runoff of storm water. The cask storage area consists of concrete cask storage pads that support the storage casks. Each pad is designed to support up to 8 storage casks in a 2 by 4 array. The Canister Transfer Building, as well as the Security and Health Physics Building, are also located within the RA. An overhead bridge crane and a semi-gantry crane are located within the Canister Transfer Building to facilitate shipping cask load/unload operations and canister transfer operations.

[2] Metric Tons of Uranium (initial uranium). This includes the small amount of mixed oxide fuels that are anticipated to require storage.

A main gate is provided for vehicular access to the RA. Light poles are located within the RA inside the security fence. Outside the RA is an Administration Building, and an Operations and Maintenance Building. The overall site or owner controlled area (OCA) is approximately 820 acres and is bounded by a range fence. The fence will be a typical 4-strand wire range fence, which will serve to identify the limit of PFSF activities and to keep out any stray livestock. Specifications for the fence, such as wire type and spacing, and pole type and spacing will meet the requirements of the BLM Manual Handbook H-1741-1 for Fencing and/or other applicable requirements identified by the BLM and BIA. PFS will consult with the BLM and BIA prior to construction of the fence to make sure the fence meets the latest BLM/BIA requirements.

The PFSF is designed to accommodate the storage system and the transportation of spent fuel canisters to and within the facility. The amount of yard area provided within the RA is sized to limit the radiation dose outside of the RA from the storage casks to less than 2 mrem/hr per 10 CFR 20.1301(a)(2). The yard area is also sized to provide adequate space for maneuvering the onsite cask transporter used during storage cask placement. The area of the OCA is based on 10 CFR 72.104(a) requirements of maintaining the annual dose to any real individual outside the OCA during normal operations to less than 25 mrem whole body, as well as 10 CFR 72.106(b) requirements of maintaining a minimum distance of 100 meters (328 ft) from spent fuel storage and handling areas to the OCA boundary and limiting the dose to 5 rem, to the whole body or any organ, from any design basis accident.

In compliance with 10 CFR 72.122 (d), the PFSF does not share structures, systems, or components with other facilities.

1.3 GENERAL SYSTEMS DESCRIPTION

In order to store spent fuel at a centralized facility, an appropriate system is required to accommodate transfer of the spent fuel from nuclear power plants to the PFSF and to provide storage of the spent fuel at the PFSF.

The PFSLLC has selected the canister-based system for use at the PFSF because it eliminates the need to handle individual spent fuel assemblies once a canister is loaded and is sealed at the originating nuclear power plant. In addition, the canister-based system is expected to be compatible with the final DOE system for spent fuel management (Reference 4).

The canister-based system utilizes a sealed metal canister to store multiple spent fuel assemblies in a controlled environment. The sealed metal canister is placed in casks that provide radiation shielding and physical protection of the fuel during fuel transfer, transportation, storage, and disposal. The vendor selected to provide canister-based storage systems at the PFSF is Holtec International (Holtec). Holtec is supplying its Holtec International Storage and Transfer Operation Reinforced Module Cask System (HI-STORM 100) (Reference 5). This canister-based storage system and its components are shown on Figure 1.3-1.

The metal canister is a cylindrical shell with a lid and closure ring, shell assembly, and bottom plate, that houses a spent fuel basket. The canister is designed to accommodate PWR or BWR fuel types, including failed and mixed oxide fuel. The spent fuel basket provides structural support for the spent fuel assemblies and a path for the transfer of heat generated by the spent fuel to the canister shell. The spent fuel basket also provides criticality control to ensure that a nuclear fission reaction can not be sustained. The cylindrical shell provides structural support for the fuel basket structure and spent fuel assemblies. During storage, the cylindrical shell, bottom plate, and the canister lid provide the primary confinement boundary to prevent the release of radioactive material from the spent fuel. The canister is designed to provide radiation shielding and physical protection for the spent fuel and uses casks to provide additional shielding and protection. Spent fuel assembly capacities of the canisters are as follows:

	Holtec HI-STORM 100 Cask System
PWR fuel	24
BWR fuel	68

After the spent fuel is loaded into a canister at the originating nuclear power plant, the canister is partially drained and the lid is welded to the canister. The canister is then drained and vacuum dried, filled with helium, and sealed closed by welding the vent and drain port cover plates. The closure ring is then welded onto the top of the canister, providing a redundant closure. Additional details of the canister design are discussed and shown in Chapter 4.

There are three types of casks used to transfer and store the canister. These are a shipping cask, transfer cask, and storage cask.

The shipping casks are used to transport the spent fuel canisters from the originating power plants to the PFSF. The shipping casks are designed to provide a complete confinement barrier, canister cooling, and to protect the canister from the effects of environmental conditions and natural phenomena. Each shipping cask is fitted with impact absorbing devices and is designed to withstand postulated transportation accidents. After the spent fuel canister is unloaded, the empty shipping cask is returned to the power plants for reloading with another canister of spent fuel.

The metal transfer cask is used to provide radiation shielding, physical protection, and canister cooling for the spent fuel canister when transferring the canister from the shipping cask to the storage cask at the PFSF.

The storage cask is a concrete and steel cylindrical structure which provides structural support for the spent fuel canister, physical protection, radiation shielding, and provides for natural convection cooling of the canister to remove decay heat while in storage at the PFSF.

At the PFSF, the shipping cask is lifted off the transport vehicle and placed in a shielded area of the Canister Transfer Building, called a transfer cell, using the overhead bridge crane. The canister is transferred from the shipping cask to the transfer cask, then transferred from the transfer cask into the storage cask using either the overhead bridge crane or the semi-gantry crane. The storage cask, loaded with the canister, is then sealed closed and moved to the storage area using a cask transporter and placed on a concrete pad. Details of the HI-STORM 100 cask are discussed and shown in Chapter 4.

The design of the canister system and the loading procedures minimizes the potential for contamination of the canister at the originating nuclear power plant. Contamination that does occur is removed to within acceptable limits at the power plant. The system assures there is negligible contamination or radioactive waste generated at the site and facilitates the ease of decommissioning at the end of the life of the facility. Site generated waste confinement and management is discussed in Chapter 6.

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1.4 SPENT FUEL TRANSPORTATION TO THE PFSF

Although transportation activities are not part of the 10 CFR 72 license application for the PFSF, the transportation process is briefly described herein to provide an understanding of the overall program. Spent fuel enroute to the PFSF will be transported in accordance with applicable U.S. Department of Transportation (DOT) regulations (49 CFR 173-Shippers General Requirements for Shipments and Packaging, Subpart A-"General" and Subpart I - "Radioactive Materials", 49 CFR 171-"General Information, Regulations, and Definitions", 49 CFR 172-"Hazardous Materials Tables and Hazardous Materials Communications Regulations", 49 CFR 174-"Carriage by Rail", 49 CFR 177-"Carriage by Public Highway"), and NRC regulations (10 CFR 71 - "Packaging and Transportation of Radioactive Material"). Holtec is supplying its Holtec International Storage, Transport, and Repository Cask System (HI-STAR 100) (Reference 7).

As a result of adherence to strict controls, utilities and carriers have a long history of safe spent fuel transportation. In more than 30 years of shipping fuel in the United States, no accident has caused a release of radioactive material. Moreover, no deaths or serious injuries to the public or to transportation industry personnel have ever occurred as the result of the radioactive nature of any radioactive material shipment (Reference 9).

Currently there is no direct rail line to the PFSF. Therefore the PFSF will be designed to employ two transport modes to ship the cask from the railroad mainline to the site. The preferred mode is to ship the shipping cask the final 32 miles by rail on a new rail line. The alternate mode is to transfer the shipping cask from the rail car to a heavy haul transport tractor/trailer at an intermodal transfer point located 1.8 miles West of Timpie and haul the shipping cask the final 26 miles by road to the PFSF. The PFSF is expected to handle approximately 100 to 200 shipments of loaded spent fuel canisters annually. The PFSF will accept delivery and perform receipt inspection of the spent fuel shipping casks at the PFSF.

PFS may contract directly with its utility customers to perform as a rail or motor carrier to transport the casks from the main rail line to the PFSF. To the extent that PFS acts as a carrier, PFS will comply with the applicable DOT statutes and regulations pertaining to rail or motor carriers, as appropriate, and the related hazardous material transportation requirements. Specifically, if PFS operates as a rail carrier for the utility customers, PFS will meet the requirements applicable to rail carriers, including, without limitation, the applicable requirements set forth in 45 U.S.C. Chapters 2, 8, 9,and 11; 49 U.S.C. Subtitle IV (Part A); Subtitle V, Subtitle X; and associated implementing regulations contained in 49 C.F.R. Parts 200, 1000 through 1300. If the heavy haul option is used and PFS operates as a motor carrier for the utility customers, PFS will meet the requirements for motor carriers including, without limitation, the applicable requirements set forth in 49 U.S.C. Subtitle Part IV Part B; Subtitle VI; Subtitle X and associated implementing regulations contained in 49 C.F.R. Parts 300, 1000, 1090-1099, 1200 and 1300. The commitments in this paragraph are made as part of a Settlement between PFS and the State of Utah in the NRC licensing proceeding for the PFSF and, as agreed to by PFS, the State and the NRC Staff, do not constitute a license condition or licensing commitment under the 10 CFR Part 72 license for the PFSF and, as further agreed, their incorporation into the SAR does not render the commitments subject to 10 CFR 72.48, or obligate the NRC Staff to enforce the above requirements, or undertake enforcement action with respect to violation of these requirements, under the 10 C.F.R. Part 72 license for the PFSF.

1.5 IDENTIFICATION OF AGENTS AND CONTRACTORS

Holtec International is providing the spent fuel storage systems for use at the PFSF. Holtec has a NRC-approved QA program and is responsible for the design and licensing of their canisters, casks, and transfer equipment.

Stone & Webster Engineering Corporation is providing the engineering and supporting preparation of the 10 CFR 72 license documentation of the PFSF site.

The PFSLLC has responsibility for construction management and construction of the PFSF. The prime construction contractor for the PFSF, with experience on nuclear projects, will either have its own NRC-approved QA program, or work under the PFSLLC NRC-approved QA program. This contractor, and other subcontractors for non-nuclear portions of the facility, will be selected at an appropriate time to support construction of the PFSF.

The PFSLLC will be responsible for operation of the PFSF.

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1.6 MATERIAL INCORPORATED BY REFERENCE

The Safety Analysis Report for the Holtec HI-STORM 100 Storage System (Reference 5) was submitted to the NRC for approval and is incorporated by reference into this document. The NRC has issued a Certificate of Compliance for Holtec's HI-STORM 100 storage cask system (Reference 12), as well as a Certificate of Compliance for its HI-STAR 100 shipping cask system (Reference 13).

The PFSLLC QA Program (Reference 11) was approved by the NRC on November 3, 1996 for use under 10 CFR 71, Subpart H (Docket 71-0829) and is incorporated by reference into this document. It is proposed that this PFSLLC QA Program be used to satisfy the requirements of 10 CFR 72, Subpart G. Chapter 11 provides a detailed discussion of the PFSLLC QA program.

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1.7 REFERENCES

1. 10 CFR 72, Licensing Requirements for the Independent Storage of Spent Nuclear Fuel and High Level Radioactive Waste.

2. U.S. Nuclear Regulatory Commission, Regulatory Guide 3.48, Standard Format and Content for the Safety Analysis Report for an Independent Fuel Storage Installation or Monitored Retrievable Storage Installation (Dry Storage), August 1989.

3. U.S. Nuclear Regulatory Commission, NUREG-1567, draft Standard Review Plan for Spent Fuel Dry Storage Facilities, October, 1996.

4. DOE/RW-0445, Multi-Purpose Canister System Evaluation, U.S. Department of Energy Civilian Radioactive Waste Management, September 1994.

5. Final Safety Analysis Report for the Holtec International Storage and Transfer Operation Reinforced Module Cask System (HI-STORM 100 Cask System), Holtec Report HI-2002444, Revision 0, NRC Docket 72-1014, July 2000.

6. (deleted)

7. Topical Safety Analysis Report for the Holtec International Storage, Transport, and Repository Cask System (HI-STAR 100 Cask System), Holtec Report HI-951251, Revision 9, Docket 71-9261, April 2000.

8. (deleted)

9. U.S. Department of Energy, "Transporting Spent Nuclear Fuel: An Overview, DOE/RW-0065 (1986).

10. (deleted)

11. PFSLLC QA Program Manual, Current Revision, Docket 71-0829.

12. 10 CFR 72 Certificate of Compliance 1014, Rev. 0, HI-STORM 100 System, May, 2000.

13. 10 CFR 71 Certificate of Compliance 9261, Rev. 0, HI-STAR 100 System, March, 1999.

(deleted)

Figure 1.3-2

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PRIVATE FUEL STORAGE FACILITY
SAFETY ANALYSIS REPORT

based on the borings and confirmed by the laboratory tests performed specifically for the purpose of classifying the soil types. Conservatively adjusting these data by subtracting 1 from the SBTs that were reported to be greater than 5 (i.e., "sandy" soils), as discussed in Section 2.6.1.6, results in the SBTs presented on the pad emplacement area foundation profiles, Sheets 2 through 14 of Figure 2.6-5. As shown on these figures, the subsurface soils that were reported in ConeTec (1999) as being silty sand/sand and sands are more correctly described as silts with some sandy silts. The following discussion is included to demonstrate that even if these soils are cohesionless soils, the factor of safety against a bearing capacity failure is much greater than that reported above for the clayey soils identified in the borings.

Whereas the bearing capacity of cohesive soils is a function of the strength of the soil, that of cohesionless soils is also a function of the width of the foundation. The foundations in question for this project have widths that are 30 ft or greater. Such large foundations, supported by soils having Standard Penetration Test blow counts that were measured for these soils, have much greater bearing capacities if they are founded on cohesionless soils than if supported by undrained cohesive soils. Therefore, characterizing the soils in the upper layer as cohesive even though some of these may be cohesionless provides a conservative estimate of the bearing capacity.

Analyses of bearing capacity were made in Calculation 05996.02-G(B)-4, (SWEC, 2000b), based on the assumption that the entire upper layer, approximately 25 to 30-ft thick, was comprised of cohesive soils similar to those tested at depths of 10 to 12 ft. In these analyses, the strength of the soils in the entire upper layer (~25 to 30-ft thick) was set equal to the minimum value measured in the UU tests (s_u = 2.2 ksf) that were performed on samples obtained from depths of approximately 10 to 12 feet. As indicated on Table 2.6-6 for Case IA, the factor of safety of the cask storage pad foundation is 6.7 using this undrained strength for the cohesive soils. The results for Case IB Table 2.6-6 illustrates that the factor of safety against a bearing capacity failure increases to greater than 14 when the effective-stress strength of ϕ = 30° is used.

The friction angle used in the effective-stress strength analyses discussed above is less than the friction angle shown for the soils that behave as sandy soils (SBT>5) based on the CPT data presented in Appendix D of ConeTec (1999). These plots illustrate that most of the "Phi" values are between 35° and 40° for these soils, with very few values that are slightly less than 35°. Therefore, assuming that all of the soils underlying the cask storage pads are cohesionless, as represented by the preponderance of soils that behave as "sandy" soils based on the uncorrected CPT SBT data, the factor of safety against a bearing capacity failure will be much greater than 14.

Static Settlements of the Cask Storage Pads

Analyses were performed to estimate the settlement of the cask storage pads as a result of the weight of the pad and the weight of eight, fully loaded, Holtec HI-STORM casks (356.5 K) in Calculation 05996.02-G(B)-3 (SWEC, 1999e). The actual bearing pressure for this case was about 1.9 ksf, and the estimated total settlement of the pad was determined to be about 3.3 inches. The total settlement consists of the following three components:

• Elastic settlement	0.5 inch
• Primary consolidation settlement	1.7 inches
• Secondary compression	1.1 inches
• Total estimated settlement	3.3 inches

LOAD COMBINATION				DISPLACEMENT
Case IIIA	40% N-S,	-100% Vertical,	40% E-W.	0.1 inches
Case IIIB	40% N-S,	-40% Vertical,	100% E-W.	0.4 inches
Case IIIC	100% N-S,	-40% Vertical,	40% E-W.	0.4 inches

The estimated relative displacement of the cask storage pads ranges from ~0.1 inches to 0.4 inches. Because there are no connections between the pads or between the pads and other structures, displacements of this magnitude, were they to occur, would not adversely impact the performance of the cask storage pads. There are several conservative assumptions that were made in determining these values and, therefore, the estimated displacements represent upper-bound values.

The soils in the layer that are assumed to be cohesionless (the one ~10 ft below the pads) are clayey silts and silts, with some sandy silt. To be conservative in this analysis, these soils are assumed to have a friction angle of 30°. However, the results of the cone penetration testing (Appendices D & F of ConeTec, 1999) indicate that these soils have ϕ values that generally exceed 35 to 40°. These high friction angles likely are the manifestation of the cementation that was observed in many of the specimens obtained in split-barrel sampling and in the undisturbed tubes that were obtained for testing in the laboratory. Possible cementation of these soils is also ignored in this analysis, adding to the conservatism.

In addition, this analysis postulates that cohesionless soils exist directly at the base of the pads. In reality, the surface of these soils is 10 ft or more below the pads, and it is not likely to be continuous, as the soils in this layer are intermixed. For the pads to slide, a surface of sliding must be established between the horizontal surface of the "cohesionless" layer at a depth of at least 10 ft below the pads, through the overlying clayey layer, and daylighting at grade. As shown in the analysis preceding this section, the overlying clayey layer is strong enough to resist sliding due to the earthquake forces. The contribution of the shear strength of the soils along this failure plane rising from the horizontal surface of the "cohesionless" layer at a depth of at least 10 ft to the resistance to sliding is ignored in the simplified model used to estimate the relative displacement, further adding to the conservatism.

These analyses also conservatively ignore the presence of the soil cement under and adjacent to the cask storage pads. As shown above, this soil cement can easily be designed to provide all of the sliding resistance necessary to provide an adequate factor of safety, considering only the passive resistance acting on the sides of the pads, without relying on friction or cohesion along the base of the pads. Adding friction and cohesion along the base of the pads will increase the factor of safety against sliding.

2.6.1.12.2 Stability and Settlement Analyses—Canister Transfer Building

Stability and settlement analyses of the Canister Transfer Building were performed to confirm the adequacy of the structure and its foundation. Calculation 05996.02-G(B)-13 (SWEC, 2000c) evaluated the stability of the Canister Transfer Building and determined it is stable with respect to bearing capacity, overturning, and sliding due to static and dynamic load conditions. Calculation 05996.02-G(C)-14 (SWEC, 2000e) evaluated the soil settlements due to static load conditions and found the resulting building settlements to be uniform and small and to have little effect on the structure. Calculation 05996.02-G(B)-11 (SWEC, 2000d) evaluated the soil settlements due to dynamic load conditions and also found the resulting building settlements to be small and to have little effect on the structure. In summary, the stability and settlement analyses of the Canister Transfer Building indicate that the building is stable and will retain its structural integrity and the performance of the structure will not be adversely affected.

Chapter 4 describes the structural analyses of the Canister Transfer Building. The analyses used a finite element model approach and considered the effects of soil-structure interaction. The structural analyses take into account the flexibility of the soil underlying the building by the use of finite elements with the stiffness properties of the soil. Non-uniform elastic deformation of the soil, which results in bending moments and shear forces in the base mat are accounted for. This analysis is performed in Calculation 0599602-SC-6 (SWEC, 1998a). Induced stresses resulting from these non-uniform displacements were accommodated in the design of the structure.

The Canister Transfer Building is a large and massive building consisting of exterior reinforced concrete walls 2'-0" thick, a reinforced concrete roof 1'-0" thick, and a solid

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reinforced concrete mat foundation 5'-0" thick. The interior partitions that make up the low level waste holding area will be constructed of concrete or concrete masonry. The equipment and office areas on the east side of the building will utilize steel-framed partition walls covered with gypsum board. The total weight (static load) of the building and foundation is approximately 73,000 kips (Calculation 05996.02-SC-5, SWEC, 1999f) or 36,500 tons.

Bearing Capacity of the Canister Transfer Building

The bearing capacity of the Canister Transfer Building foundation was determined using the general bearing capacity equation and associated shape, depth, and inclination factors, as presented in Winterkorn and Fang (1975). Refer to Calculation 05996.02-G(B)-13 (SWEC, 2000c) for details. These analyses are based on the strength parameters for the silty clay/clayey silt layer directly underlying the mat. Conservatively ignoring the presence of the denser layers, which start at a at a depth of ~25 to 30 ft, these analyses demonstrate that there is an adequate factor of safety against a bearing capacity failure for both static and dynamic loadings. They included determination of factors of safety against a bearing capacity failure of the foundation due to static loads and due to static plus dynamic loads from the design basis ground motion (PSHA 2,000-yr return period earthquake). The dynamic bearing capacity analyses are discussed in detail in the section below titled *"Dynamic Bearing Capacity of the Canister Transfer Building."*

Static Bearing Capacity of the Canister Transfer Building

Table 2.6-9 presents the results of the bearing capacity analyses for the following static load cases. As indicated above, the minimum factor of safety required for static load cases is 3.

- Case IA Static using undrained strength parameters (ϕ = 0° & c =3.18 ksf).
- Case IB Static using effective-stress strength parameters (ϕ = 30° & c = 0).

As indicated in this table, the gross allowable bearing pressure for the Canister Transfer Building to obtain a factor of safety of 3.0 against a shear failure from static loads is greater than 6 ksf. However, loading the foundation to this value may result in undesirable settlements. This minimum allowable value was obtained in analyses that conservatively assume $\phi = 0°$ and c = 3.18 ksf, to model the end of construction. This is the average undrained strength of the soils in the upper ~25 to ~30 ft, as discussed in Section 2.6.1.11.2. Using the estimated effective-stress strength of $\phi = 30°$ and c = 0, results in higher allowable bearing pressures. As shown in Table 2.6-9, the gross allowable bearing capacities of the Canister Transfer Building for static loads for this soil strength is 45 ksf.

Settlement of the Canister Transfer Building

Analyses were performed to estimate the settlement of the Canister Transfer Building for the static dead and live loads in Calculation 05996.02-G(C)-14 (SWEC, 2000e). A total building settlement of approximately 3 inches is estimated over the life of the building. The settlement will be generally uniform. Of the total building settlement, approximately 1.9 inches will occur within a few years after construction and an additional 1.1 inches will occur during the life of the building. These analyses were performed using the results of the consolidation tests that are included in Attachment 2 of Appendix 2A.

As indicated in Section 2.6.1.12.1 regarding the settlement analyses of the storage pads, this settlement represents an upper-bound estimate of the settlement, because it was developed assuming that the consolidation characteristics that were measured for the clayey soils at a depth of about 10 ft are applicable for the entire upper layer. The SPT data from the borings and the CPT results indicate that the soils become stiffer within the 10 to 20 ft depth zone. Additional consolidation tests performed on samples

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CHAPTER 3

PRINCIPAL DESIGN CRITERIA

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CHAPTER 3

PRINCIPAL DESIGN CRITERIA

This chapter identifies the principal design criteria for the Private Fuel Storage Facility (PFSF). The principal design criteria provide a record of design bases derived from 10 CFR 72 and applicable industry codes and standards referenced herein for comparison with the actual design, which is presented in subsequent chapters.

3.1 PURPOSES OF INSTALLATION

The purpose of the PFSF is to provide interim storage for up to 40,000 MTU of pressurized water reactor (PWR) or boiling water reactor (BWR) spent fuel from commercial nuclear power plants throughout the United States.

The PFSF shall utilize canister-based dry cask storage systems, where multiple spent fuel assemblies are stored in a dry inert environment inside a sealed metal canister that is placed inside a storage cask and stored outdoors on a concrete pad. The storage system shall provide physical protection, heat removal, radiation shielding, criticality control, and confinement for the safe storage of spent fuel. The storage systems shall be designed to maintain retrievability of the canister for future removal offsite.

The dry cask storage system used at the PFSF shall be the HI-STORM 100 Cask System (HI-STORM) designed by Holtec International (Holtec). Holtec has submitted a Safety Analysis Report (SAR) to the U.S. Nuclear Regulatory Commission (NRC) for the HI-STORM 100 Cask System (Reference 1). The NRC has issued a Certificate of Compliance for Holtec's HI-STORM 100 storage cask system (Reference 34).

3.1.1 Materials to be Stored

The PFSF shall be designed to store commercial BWR and PWR spent nuclear fuel with zircaloy or stainless steel cladding including failed fuel, BWR fuel channels, PWR control components, and mixed oxide (MOX) fuel. The spent fuel characteristics from these plants shall be encompassed by the design fuel characteristics that are established by the storage systems used at the PFSF.

The types of fuel to be stored at the PFSF are based on the types of fuel each storage system is licensed to store and PFSF design requirements. A summary of the fuel types that can be stored at the PFSF is shown in Appendix B of the HI-STORM Certificate of Compliance (Reference 34) for the HI-STORM 100 system.

The bounding design fuel characteristics for the PFSF, which are based on the capabilities of the storage system utilized at the PFSF, are summarized in Appendix B of the HI-STORM Certificate of Compliance for the HI-STORM 100 system.

3.1.2 General Operating Functions

3.1.2.1 Transportation and Storage Operations

The PFSF shall be designed to use a passive dry storage technology. Canister transfer and cask placement or removal operations are the major activities.

Prior to receipt at the PFSF, the spent fuel is loaded in a canister at the originating nuclear power plant. The canister is surveyed for contamination, decontaminated if necessary, drained, vacuum dried, filled with helium, and sealed closed prior to shipping. The canister is then loaded into a shipping cask. The shipping cask is protected by impact limiters and mounted on a shipping cradle, and attached to a rail car and shipped to the PFSF.

The PFSF shall be designed to utilize two transport modes to haul the shipping cask from the railroad mainline to the site. The preferred mode is to haul the shipping cask by rail on a railroad line to be constructed from Low Junction to the PFSF. The railroad line and associated equipment shall be designed in accordance with railroad industry standards. The alternate mode is to haul the shipping cask by highway on a heavy haul tractor/trailer from an intermodal transfer point, located next to the railroad mainline 1.8 miles West of Timpie, to the PFSF via Skull Valley Road. The intermodal transfer point shall include the necessary components (crane, rail siding, and truck access area) to accommodate the rail to tractor/trailer transfer.

At the PFSF the canister shall be transferred from the shipping cask to the storage cask. The shipping cask shall be off-loaded from the transport vehicle inside the Canister Transfer Building using an overhead crane and placed in a shielded transfer cell. Once the shipping cask has been opened a transfer cask shall be placed on top of the shipping cask and the canister hoisted up and secured into the transfer cask. The transfer cask shall then be moved by crane onto the top of a storage cask and the canister shall be lowered into the storage cask. The storage cask lid shall be installed and bolted. The storage cask shall then be moved to the cask storage pad using a cask transporter. Storage of the loaded storage cask shall include temperature monitoring and periodic surveillance of the storage casks.

When the fuel is to be shipped offsite, the storage cask shall be moved back into the Canister Transfer Building using the cask transporter. The transfer cask shall be placed on top of the storage cask and the canister lifted up and secured into the transfer cask. The transfer cask shall then be moved by crane onto the top of a shipping cask. The canister shall be lowered into the shipping cask, which shall be closed and shipped offsite.

The PFSF shall be designed with the necessary equipment (such as, the Canister Transfer Building, cranes, cask transporter, storage area) to accommodate shipping cask receipt, canister transfer from the shipping cask to the storage cask, cask transport to and from the storage pads as detailed above with provisions for security, health physics, maintenance, document control, and inventory management.

3.1.2.2 Onsite Generated Waste Processing, Packaging and Storage

The selected canister-based storage systems shall be designed to confine spent fuel within a sealed canister at the originating nuclear power plant. Therefore, handling of spent fuel is not required and no radioactive waste is generated at the PFSF.

Health physics survey material (i.e. smears, disposable clothing) shall be collected, identified, packaged in low level waste (LLW) containers, marked in accordance with 10 CFR 20 requirements, and temporarily stored in the LLW holding cell of the Canister Transfer Building while awaiting shipment to an offsite low-level radioactive disposal facility.

There shall be no other systems or facilities for processing, packaging, storing, or transporting any other type of radioactive waste at the PFSF.

3.1.2.3 Utilities

The PFSF shall be designed to include utilities necessary for facility operation. These utilities include (1) electrical power for operation of equipment, lights, monitoring equipment, communication systems, security systems; (2) backup electrical power for operation of security systems, emergency lights, monitoring equipment, and communication systems; and (3) mechanical systems for operation of fire protection equipment, building HVAC systems, compressed air systems, water supply systems, and septic systems. Utilities do not need to be classified as Important to Safety unless their function could affect the safe operation of a SSC that is classified as Important to Safety.

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3.2 STRUCTURAL AND MECHANICAL SAFETY CRITERIA

This section of the principal design criteria establishes requirements that satisfy 10 CFR 72.122(b), which identifies the general design criteria that requires structures, systems, and components (SSCs) classified as Important to Safety be designed to withstand the effects of environmental conditions and natural phenomena in their structural and mechanical design. SSCs classified as Important to Safety shall be designed with sufficient capability to withstand the worst-case loads under normal, off-normal, and accident-level conditions such that their capability to perform safety functions is not impaired. Accident-level conditions include credible accidents, natural phenomena, and hypothetical events. Loads considered for the PFSF are categorized as follows:

Load	Normal	Off-normal	Accident-Level
Dead Loads	x		
Live Loads	x		
Handling Loads	x	x	
Snow and Ice Loads	x		
Wind Loads	x		
Internal/External Pressure	x	x	
Lateral Soil Pressure	x	x	x
Thermal Loads	x	x	x
Accident Loads			
Explosion Overpressure			x
Drop/Tipover			x
Accident Pressurization			x
Fire			x
Tornado Winds/Missiles			x
Floods			x
Earthquake			x

Design criteria for these loads are described in this chapter and shall be used in the design of all SSCs classified as Important to Safety.

The SSCs that are classified as Important to Safety include:

- The Dry Cask Storage Systems - The dry cask storage system (HI-STORM) shall consist of metal canisters for spent fuel storage, storage casks, a metal transfer cask, lifting.attachments, and associated equipment.

- Cask Storage Pads - The cask storage pads shall provide a stable and level support surface for the storage casks.

- Canister Transfer Building - The Canister Transfer Building shall be a reinforced concrete, one-story, high-bay structure that houses the canister transfer cranes and supports shipping cask receiving and canister transfer operations. The Canister Transfer Building shall use cells designed for canister transfer operation with thick concrete walls to shield personnel from radiation doses.

- Canister Transfer Cranes - The overhead bridge and semi-gantry cranes shall be single-failure-proof. The overhead bridge crane shall have a maximum capacity of 200 tons and shall be used to load and unload shipping casks from the shipment vehicle or transfer the canisters between the shipping cask and the storage cask. The semi-gantry crane shall have a minimum capacity of 150 tons and shall be used to transfer the canisters between the shipping cask and the storage cask.

The HI-STORM storage system design criteria are fully described in its SAR. Where the storage system design criteria do not bound the PFSF design criteria, the storage system design shall be shown in subsequent chapters as complying with the PFSF site-specific design criteria. The storage system design parameters that require site-specific analysis and/or design and the Sections where they are addressed are as follows:

Site Specific Design Criteria	Section Addressed
• Cask stability during a seismic event	4.2.1.5.1(H)
• Radiation doses for 4000 cask array to the RA, OCA, and nearest residence	7.3.3.5 and 7.6
• Off-normal contamination release event	8.1.5
• Hypothetical storage cask tipover onto a PFSF concrete storage pad	8.2.6
• Hypothetical loss of confinement	8.2.7
• Fire	8.2.5

3.2.1 Dead Load

Dead load is defined as the self weight of the structure, including all permanently installed equipment, and loads due to differential settlement, creep and shrinkage.

3.2.2 Live Loads

Live loads are defined as all equipment not permanently installed, lift loads, and all loads other than dead loads that might be experienced that are not separately identified and used in the applicable load combinations. These include normal and off-normal handling and impact loads from equipment. Impact loads for the cranes include equipment loads imposed on the crane through supporting members of the building and loads induced by the acceleration and deceleration of the crane bridge, gantry, or trolley.

3.2.3 Snow and Ice Loads

Snow loads, which are considered as live loads, shall be determined in accordance with ASCE-7 (Reference 3). The site is located in an area designated as CS on ASCE-7 Figure 7-1. Areas designated as CS require site-specific Case Studies to establish ground snow loads. In lieu of site-specific analysis, the ground snow load (P_g) is based on recommendations from the Tooele County Building Department for design of structures per the Uniform Building Code (UBC). UBC figure A-16-1 designates that the ground snow load for the entire State of Utah be established by the building official. The Tooele County Building Department recommends a 43 psf ground snow load for the reservation. This value is rounded up to a 45 psf ground snow load for a conservative design value. Design snow loads and placement of loads on structures shall follow the procedures outlined in ASCE-7.

3.2.8.2 Determination of Forces on Structures

Forces resulting from the design basis wind and the design basis tornado shall be considered in the design. The method used to convert wind loading into forces on a structure shall be in accordance with NUREG-0800 (Section 3.3.1, Wind Loadings, and Section 3.3.2, Tornado Loadings).

3.2.8.3 Ability of Structure to Perform Despite Failure of Structure Not Designed for Tornado Load

The PFSF shall be designed to ensure that SSCs that are not designed for tornado loads do not adversely affect the safety functions of SSCs that are classified as Important to Safety.

SSCs that are classified as Important to Safety but not designed for tornado loads shall be located so as to be protected by a SSC that is classified as Important to Safety and designed for tornado loads.

The Canister Transfer Building shall be designed to withstand tornado-generated wind loadings and missiles in order to protect Important to Safety SSCs housed within the building that are not designed for tornado loads.

3.2.8.4 Tornado Missiles

SSCs that are classified as Important to Safety shall be designed for tornado-generated missiles except as noted in Section 3.2.8.3.

The loaded storage casks shall remain stable and the confinement boundary not breached when subjected to tornado-generated missiles.

The storage pads and Canister Transfer Building shall remain stable and structurally intact when subjected to tornado-generated missiles.

Tornado-generated missiles need not be considered in the design of the canister, overhead bridge and semi-gantry cranes, or transfer cask since the canister is protected by the storage cask and the cranes and transfer cask are protected by the Canister Transfer Building.

NUREG-0800, Section 3.5.1.4 requires that postulated tornado missiles include at least three objects: a massive high kinetic energy missile which deforms on impact, a rigid missile to test penetration resistance, and a small rigid missile of a size sufficient to just pass through any openings in protective barriers. To bound these three objects, NUREG-0800, Section 3.5.1.4 requires the applicant analyze the specific missiles defined as "Spectrum I" or "Spectrum II". Spectrum II missiles are used for the Canister Transfer Building since the type and velocity of the missiles specified are representative of the types of objects which might be found near the PFSF site. Therefore the postulated tornado missiles for the design of the Canister Transfer Building shall be in accordance with NUREG-0800, Section 3.5.1.4, for Spectrum II missiles for Region III. The tornado-generated missiles shall include:

A. 115 lb. wood plank (3.6" x 11.4" x 12' long) with horizontal velocity of 190 ft/sec.
B. 287 lb. 6" schedule 40 pipe with horizontal velocity of 33 ft/sec.
C. 9 lb. 1" diameter steel rod with horizontal velocity of 26 ft/sec.
D. 1124 lb. 13.5" diameter wooden utility pole with horizontal velocity of 85 ft/sec.
E. 750 lb. 12" schedule 40 pipe with horizontal velocity of 23 ft/sec.
F. 3990 lb. automobile with horizontal velocity of 134 ft/sec.

NOTES: Vertical velocities are 70% of horizontal velocities except for missile C. Missile C has the same velocity in all directions. Missiles A, B, C, and E are

3.2.10.1.1 Design Response Spectra

The design basis ground motion for the PFSF is described by site-specific response spectrum curves anchored at 0.53 g in two directions of the horizontal plane and 0.53 g in the vertical plane. The response spectra curves are free field at the ground surface and account for the local soil conditions. The horizontal and vertical design response spectra curves for the site are shown in Figure 5 of Reference 27.

3.2.10.1.2 Design Response Spectra Derivation

Site-specific horizontal and vertical design response spectra curves for the facility are developed using probabilistic seismic hazard analysis methodology in accordance with Regulatory Guide 1.165 (Reference 25), as described in References 27 and 28.

3.2.10.1.3 Design Time History

Design time histories shall be used in the cask stability analyses and in the storage pad design. Statistically independent artificial time histories shall be developed in accordance with NUREG-0800, Sections 3.7.1 and 3.7.2 shall be shown to envelope the site-specific response spectra.

3.2.10.1.4 Use of Equivalent Static Loads

The HI-STORM storage system is dynamically analyzed and does not use equivalent static loads.

Equivalent static loads are not used for onsite structures since dynamic analyses are used in the seismic analysis and design.

3.2.10.1.5 Critical Damping Values

Critical damping values shall be in accordance with Regulatory Guide 1.61 (Reference 13) for a SSE.

3.2.10.1.6 Basis for Site-Dependent Analysis

Site-specific vibratory ground motion is established through evaluation of the seismology, geology, and the seismic and geologic history of the site and surrounding region. This information is contained in the site-specific probabilistic seismic hazard analysis (References 27 and 28).

3.2.10.1.7 Soil-Supported Structures

The soil-supported structures that shall be analyzed for the ISFSI design basis ground motion are the concrete cask storage pads and the Canister Transfer Building. These structures shall be founded on in-situ soil at a minimum depth of 2 ft 6 inches for frost protection. The depth of soil over bedrock is between 520 ft and 880 ft below the surface of the site (Reference 9).

3.2.10.1.8 Soil-Structure Interaction

Soil-structure interaction shall be considered in the design of soil-supported structures by including the effects of the soil properties established during the geotechnical investigation program and as represented by discrete soil springs or a finite element layered system as described in ANSI/ANS 57.9, Appendix C.

Soil boring logs and soil properties of the PFSF site are contained in Chapter 2, Appendix 2A.

3.2.10.2 Seismic-System Analysis

3.2.10.2.1 Seismic Analysis Methods

Seismic analysis methods shall be in accordance with standard practices and methods as described in ANSI/ANS 57.9, NUREG-0800, ASCE-4 (Reference 14), and others referenced herein.

The seismic response of each structure shall be determined by preparing a mathematical model of the structure and calculating the response of the model to the prescribed seismic input.

The HI-STORM storage system seismic loadings and analysis methods are described in the HI-STORM SAR, Sections 2.2.3.7 and 3.4.7, respectively. Site-specific cask stability analysis shall be performed to account for the site-specific seismic response spectra curves, soil-structure interaction, and the actual PFSF pad size and arrangement.

The concrete storage pads shall be analyzed with a dynamic seismic time history analysis using a finite element model with soil-structure interaction considered by the use of dynamic soil springs. Various combinations of cask placements shall be considered to determine the controlling load case.

The Canister Transfer Building shall be analyzed for seismic loads using a frequency response analysis and considering soil-structure interaction.

The overhead bridge and semi-gantry cranes shall be analyzed considering the Maximum Critical Load (maximum lifted load whose uncontrolled movement or release could adversely affect the operation of SSCs classified as Important to Safety) in combination with a seismic event in accordance with NUREG-0554 (Reference 15). A set of amplified response spectra curves at the crane rail locations shall be developed for use in the crane seismic analysis and design.

3.2.10.2.2 Natural Frequencies and Response Loads

The modal analysis considers the natural frequency of the system as well as the other significant modes of vibration. Response loads are determined from the appropriate response spectra at the calculated frequencies.

3.2.10.2.3 Procedure Used to Lump Masses

The inertial mass properties of each structure shall be modeled using the discretization of mass formulation whereby the structural mass and associated rotational inertia are discretized and lumped at node points of the model. Node points where masses are lumped shall be located at the center of gravity of the member or component represented in the model.

3.2.10.2.6 Methods Used to Account for Torsional Effects

The storage pads and the Canister Transfer Building shall be modeled and analyzed as 3-dimensional multimass systems. Therefore, torsional effects due to eccentricities of the mass are taken into account in the analysis.

3.2.10.2.7 Methods for Seismic Analysis of Dams

There are no dams onsite or in the immediate area.

3.2.10.2.8 Methods to Determine Overturning Moments

Overturning stability shall be assured for the storage casks on the pads.

Overturning stability of loaded storage casks located on a storage pad shall be proved with a dynamic analysis using the site-specific seismic design parameters and considering soil-structure interaction.

3.2.10.2.9 Analysis Procedure for Damping

Critical damping values shall be developed in accordance with Regulatory Guide 1.61 for a SSE.

3.2.10.2.10 Seismic Analysis of Overhead Cranes

The overhead bridge and semi-gantry cranes shall be analyzed for seismic effects in accordance with the requirements of NUREG-0554 for single-failure-proof cranes. The seismic analysis of the cranes shall include the Maximum Critical Load in the lifted position during a seismic event. The seismic analysis methods shall be in accordance with ASME NOG-1 (Reference 16). A set of amplified response spectra curves at the crane rail locations shall be developed for use in the crane seismic analysis and design.

3.2.10.2.11 Seismic Analysis of Specific Safety Features

SSCs classified as Important to Safety shall meet the requirements of 10 CFR 72.122(b)(2), which requires SSCs be designed such that design basis ground motion will not result in an uncontrolled release of radioactive material or increased radiation exposure to workers or members of the general public.

3.2.11 Combined Load Criteria

The design shall consider all appropriate loads and load combinations as required by the specific SSC design code(s). Design loads shall be determined from normal, off-normal, and accident-level conditions. Design loads shall be combined to simulate the most adverse load conditions.

3.2.11.1 HI-STORM Storage System Load Combinations

Loads and load combinations used in the design of the HI-STORM 100 Cask System are identified in the HI-STORM SAR, Sections 2.2.7 and 3.1.2.1.2. Exceptions to the various code criteria are shown in HI-STORM SAR, Table 2.2.15.

HI-STORM Canister

The canister shell and internals are required by the HI-STORM SAR to be designed to the applicable requirements of Subsections NB and NG of the ASME BPVC, Section III (Reference 17). The load combinations for all normal, off-normal and accident conditions and corresponding Service Levels of the canister design are as follows:

W' = Tornado wind

M = Tornado Missile Loads

The stress intensity limits for the steel structure of the HI-STORM storage cask and HI-TRAC transfer cask (governed by Subsection NF of the ASME BPVC, Section III for plate and shell components) are shown in Table 3.2-2. Limits for the Level D condition are obtained from Appendix F of the ASME BPVC, Section III for the steel structure of the storage cask. The storage cask concrete structure design is governed by ACI-349.

The ASME BPVC is not applicable to the HI-TRAC transfer cask for accident conditions, service level D conditions. The HI-TRAC cask shall be shown by analysis to not deform and cause an applied load to the canister, have any shell rupture, or have the top lid or transfer lid detach. The HI-TRAC lifting trunnion design is governed by ANSI N14.6.

3.2.11.2 (deleted)

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3.2.11.3 Cask Storage Pad Load Combinations

The cask storage pads shall be conventional mat foundations of reinforced concrete construction.

Loads and load combinations used in the design of the concrete storage pads shall be in accordance with ANSI/ANS 57.9 and ACI-349 (Reference 18) and shall include skip loading conditions to account for incremental cask placement.

Load factors and allowable stresses used in the design shall be in accordance with ACI-349.

The concrete storage pad design shall consider the following load combinations as included in, or derived from, ANSI/ANS 57.9 and ACI-349:

Normal Conditions

$U_c > 1.4D + 1.7L$

$U_c > 1.4D + 1.7L + 1.7H$

Off-Normal Conditions

$U_c > 0.75\ (1.4D + 1.7L + 1.7H + 1.7T)$

$U_c > 0.75\ (1.4D + 1.7L + 1.7H + 1.7T + 1.7W)$

Accident-Level Conditions

$U_c > D + L + H + T + (E \text{ or } A \text{ or } W_t \text{ or } F)$

$U_c > D + L + H + T_a$

Where:

U_c = Minimum available strength capacity of a cross section or member calculated per the requirements and assumptions of ACI-349

D = Dead load

L = Live load

H = Lateral soil pressure

W = Wind loads

W_t = Tornado wind and missile loads

E = ISFSI Design Earthquake load

T = Thermal loads

T_a = Accident-level thermal loads

A = Accident loads

F = Flood loads (not applicable to this site)

The allowable soil bearing pressures beneath the cask storage pad are described in Section 2.6.1.12.

Requirements for Concrete Storage Pads Associated with Cask Drop/Tipover Accident

In addition to the above load combination criteria, the concrete storage pads and foundation are required to have the following characteristics to assure the validity of Holtec's analysis of the HI-STORM storage cask tipover/drop onto the storage pad:

a. Concrete Thickness: ≤ 36 inches

b. Concrete Compressive Strength: ≤ 4,200 psi at 28 days

c. Reinforcement top and bottom (both directions):

d. Reinforcement area and spacing determined by analysis

e. Reinforcing bar shall be 60 ksi Yield Strength ASTM Material

f. Soil Effective Modulus of Elasticity: ≤ 28,000 psi

(Measured prior to ISFSI pad installation)

An acceptable method of defining the soil effective modulus of elasticity applicable to the drop and tipover analyses is provided in Table 13 of NUREG/CR-6608 (February, 1998) with soil classification in accordance with ASTM D2487-93, Standard Classification of Soils for Engineering Purposes (Unified Soil Classification System, USCS) and density determination in accordance with ASTM D1586-84, Standard Test Method for Penetration Test and Split/Barrel Sampling of Soils.

These storage pad requirements are identified in Appendix B of the HI-STORM 100 System Certificate of Compliance (Reference 34).

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3.3 SAFETY PROTECTION SYSTEMS

3.3.1 General

The PFSF shall be designed for safe containment and storage of the spent fuel. The PFSF shall withstand normal, off-normal, and postulated accident conditions without release of radioactive material. The major design elements that assure that the safety objectives are met are the storage system, the cask storage pads, the Canister Transfer Building, and the canister transfer cranes.

The primary safety functions of the storage system principal components (canister, storage cask, and transfer cask) are as follows:

1. Canister

    - Provides confinement of the spent nuclear fuel and associated radioactive material.
    - Provides criticality control.
    - Provides heat transfer capability so that the fuel clad temperature does not exceed allowables.
    - Provides radiation shielding (together with a storage cask or transfer cask).

2. Storage cask

    - Protects the canister from weather and postulated environmental events such as earthquakes and tornado missiles.
    - Facilitates heat transfer (ventilated) of the canister.
    - Provides radiation shielding.

3. Transfer cask

- Serves as a special transfer and lifting device for movement of the spent fuel canister.
- Provides physical protection of the canister during canister transfer operations.
- Provides radiation shielding to minimize exposure rates during transfer operations.
- Facilitates heat transfer of the canister.

The primary safety function of the cask storage pads is to:

- Provide a stable and level surface for the storage casks.
- Provide required yielding for drop/tipover of the storage casks.

The primary safety functions of the Canister Transfer Building are to:

- Provide tornado and wind protection during transfer operations.
- Provide protection from tornado-generated missiles.
- Provide radiation shielding during transfer operations.
- Provide the support for the canister transfer cranes.
- Provides fire suppression

The primary safety function of the overhead bridge and semi-gantry cranes is to:

- Provide the single-failure-proof lifting capability for shipping cask load/unload operations and canister transfer operations.

As discussed in the following sub-sections, the PFSF design shall incorporate design features addressing each of the above functions to assure safe execution of PFSF operations.

3.3.2 Protection By Multiple Confinement Barriers and Systems

This section of the principal design criteria establishes requirements that satisfy 10 CFR 72.122(h), which identifies general design criteria requirements to protect and confine the spent fuel.

3.3.2.1 Confinement Barriers and Systems

The primary confinement barrier for spent nuclear fuel is the canister. The canister is required to maintain confinement for normal storage conditions and all postulated accidents with the protection of the storage cask or transfer cask.

The canister shall be designed to provide a confinement barrier for spent nuclear fuel. The canister confinement barrier shall be designed in accordance with ASME Boiler and Pressure Vessel Code, Section III.

The canister internals, which are used to constrain fuel assemblies during storage, shall be designed in accordance with ASME Section III, Subsection NG.

The canister shall be designed to withstand credible drop accidents (drops less than 10 inches while in the storage cask) without impairing fuel retrievability. The canister shall also be designed to maintain leak tightness and ensure that there is no leakage of radioactive material under all postulated loading conditions.

3.3.2.2 Ventilation Offgas

There are no ventilation offgas systems at the PFSF. The welded sealed canister precludes the need for offgas systems.

3.3.3 Protection by Equipment and Instrumentation Selection

3.3.3.1 Equipment

The SSCs that have been identified as Important to Safety, per Section 3.4, for the PFSF are:

- Storage cask system canister, storage cask, transfer cask, and lifting devices.
- Cask Storage Pads.
- Canister Transfer Building.
- Canister Transfer Cranes (overhead bridge and semi-gantry).

The design criteria for these components are summarized in Section 3.6.

3.3.3.2 Instrumentation

This section of the principal design criteria establishes requirements that satisfy 10 CFR 72.122(i), which identifies general design criteria that requires instrumentation and control systems be provided to monitor systems that are classified as Important to Safety. These systems shall be monitored over the anticipated ranges for normal and off-normal operation.

3.4 CLASSIFICATION OF STRUCTURES, SYSTEMS, AND COMPONENTS

The SSCs of the PFSF are classified as Important to Safety or Not Important to Safety. A tabulation of the SSCs by their classification is shown in Table 3.4-1. The criteria for selecting the classification for particular SSCs are based on the following definitions:

Important to Safety

A classification per 10 CFR 72.3 for any structure, system, or component whose function is to maintain the conditions required to safely store spent fuel, prevent damage to spent fuel containers during handling and storage, and provide reasonable assurance that spent fuel can be received, handled, packaged, stored, and retrieved without undue risk to the health and safety of the public.

Not Important to Safety

A quality classification for items or services that do not have a safety related function and that are not subject to special utility requirements or NRC imposed regulatory requirements.

SSCs classified as Important to Safety shall be designed, constructed, and tested in accordance with the Quality Assurance (QA) Program described in Chapter 11. The level of importance to safety for each SSC shall be based on QA classification categories as detailed in NUREG/CR-6407 (Reference 24). The classifications are intended to standardize the QA control applied to activities involving spent fuel storage systems. These classifications are defined as follows:

Classification Category A - Critical to Safe Operation

Category A items include SSCs whose failure or malfunction could directly result in a condition adversely affecting public health and safety. The failure of a single item could cause loss of primary containment leading to release of radioactive material, loss of shielding, or unsafe geometry compromising criticality control.

Classification Category B - Major Impact on Safety

Category B items include SSCs whose failure or malfunction could indirectly result in a condition adversely affecting public health and safety. The failure of a Category B item, in conjunction with the failure of an additional item, could result in an unsafe condition.

Classification Category C - Minor Impact on Safety

Category C items include SSCs whose failure or malfunction would not significantly reduce the packaging effectiveness and would not be likely to create a situation adversely affecting public health and safety.

The QA determination for the SSCs that are classified as Important to Safety are discussed in the following sections. A QA classification for these SSCs establishes the requirements that satisfy 10 CFR 72.122(a) general design criteria, which specifies SSCs Important to Safety be designed, fabricated, erected, and tested to quality standards.

3.4.1 Spent Fuel Storage Systems

3.4.1.1 Canister

The canister is classified as Important to Safety, Classification Category A since it serves as the primary confinement structure for the fuel assemblies and is designed to remain intact under all accident conditions analyzed in Chapter 8.

3.4.1.2 Storage Cask

The storage cask is classified as Important to Safety, Classification Category B since it is designed to remain intact under all accident conditions analyzed in Chapter 8 and serves as the primary component for protecting the canister during storage and provide radiation shielding and canister heat rejection.

3.4.1.3 Transfer Cask

The transfer cask is classified as Important to Safety, Classification Category B since it is designed to support the canister during transfer lift operations and provide radiation shielding and canister heat rejection.

3.4.1.4 Lifting Devices

The lifting devices (lift yoke, trunnions, and canister lift attachments) are classified as Important to Safety, Classification B to preclude the accidental drop of a canister.

3.4.2 Cask Storage Pads

The cask storage pads are classified as Important to Safety, Classification Category C to ensure a stable and level support surface for the storage cask under normal, off-normal, and accident-level conditions.

3.4.3 Canister Transfer Building

The Canister Transfer Building is classified as Important to Safety, Classification Category B to protect the canister from adverse natural phenomena during shipping cask load/unload operations and canister transfer operations. The building shall provide physical protection from tornado winds and missiles, radiological shielding inside to workers during transfer operations, and support for the canister transfer cranes.

3.4.4 Canister Transfer Cranes

The overhead bridge and semi-gantry canister transfer cranes are classified as Important to Safety, Classification Category B to preclude the accidental drop of a shipping cask without impact limiters during load/unload operations or canister during the canister transfer operations.

3.4.5 Seismic Support Struts

The seismic support struts are classified as Important to Safety, Classification Category B to ensure that the casks will remain stable and will not topple in the event of an earthquake.

3.4.6 Design Criteria for Other SSCs Not Important to Safety

The design criteria for SSCs classified as Not Important to Safety, but which have security or operational importance, such as security systems, standby power systems, cask transport vehicles, flood prevention earthwork, fire protection systems, radiation monitoring systems, and temperature monitoring systems, are addressed in subsequent chapters of this SAR. These SSCs shall be required to comply with their applicable codes and standards to ensure compatibility with SSCs that are Important to Safety and to maintain a level of quality that shall ensure that they will mitigate the effects of off-normal or accident-level events as required.

The cask transporter is classified as not Important to Safety but is designed with several features that assure safety while transporting spent nuclear fuel. Potential failure mechanisms of the transporter could involve the drive-train, brakes, electrical system, or lift beam hydraulic ram. Of these potential failures, only those that could drop the cask have the possibility of damaging the cask and adversely affecting public health and safety. Because of this, the transporter is not permitted by design to lift a cask above the cask vendor's analyzed safe handling height. In addition, a Technical Specification is proposed to ensure that the casks will not be lifted above the vendor's analyzed safe handling height. Therefore, a failure of the cask transporter will not damage the spent fuel storage system or adversely affect the health and safety of the public, which is the basis for the transporter classification as Not Important to Safety.

The flood control berm is classified as not Important to Safety. Flooding due to PMF would not compromise the safety of the storage casks or the Canister Transfer Building if the berm was not constructed or if it failed since the cask systems are designed to withstand severe flooding and full submergence. The berm is provided to minimize stormwater flowing across the site for ease of operations and maintenance activities.

Complete blockage of air inlet ducts, which could be hypothesized to occur in the event that severe flooding were to result in standing water above the tops of the air inlet ducts, is described in SAR Section 8.2.8. This section indicates that the HI-STORM inlet ducts can be blocked for 33 hours before the concrete of the HI-STORM storage cask would reach its short-term temperature limit, and over 72 hours without the fuel rod cladding exceeding its short-term temperature limit. PMF flows are mitigated in the Canister Transfer Building by locating the ground floor elevation above the maximum elevation of flood water. In addition, forces due to flowing water would be insignificant and would not affect the stability to the casks due to the shallow depth of the flow across the site.

The closed circuit television (CCTV) is classified as not Important to Safety. The function of the CCTV is to assist in assessment of unauthorized penetration within the protected area as required per 10 CFR 73.51 (Reference 30). As noted in NUREG-1497 (Reference 31), adequate assessment may also be provided through onsite assessment by security personnel if an acceptable justification of timely assessment can be provided. A failure of the CCTV system would be discovered immediately by security personnel as indicated by a loss of continuously observed surveillance capabilities. Appropriate compensatory measures would then be initiated, eg, sending security personnel to CCTV observation locations to provide timely onsite surveillance.

The PFSF radiation monitors are classified as not Important to Safety since they are not needed to prevent or mitigate any credible accident that would adversely affect public health and safety. The PFSF will utilize various types of radiation monitors including area monitors, thermoluminescent dosimeters (TLD), portable hand held monitors, personnel dosimetry, and portable airborne monitors. The purpose of the area radiation monitors is to detect and alarm high radiation conditions in the canister transfer building. The purpose of TLDs is to record radiation doses received at the radiation area boundary, owner controlled area boundary, and by PFSF personnel. The purpose of the portable hand held monitors is to provide surveillance of radiation levels near worker locations during transfer operations. The purpose of the personnel dosimetry, which is worn by all workers in the canister transfer area, is to measure worker accumulated dose while in the transfer area. The purpose of the portable airborne monitors is to ensure that, although the canisters are sealed, no airborne radioactivity is present during transfer operations. The use and presence of various types of monitors during facility operations provides defense in depth and will ensure that even if one fails, other monitors would detect high radiation conditions and alarm to provide safe working conditions for onsite personnel.

The temperature monitoring system is classified as not Important to Safety. The purpose of the temperature monitoring system is to provide continuous surveillance of each cask's temperature to ensure proper operation. In the event of a temperature monitor failure, the monitoring computer would not receive a signal. This would create an alarm informing personnel of a potential cask temperature problem. A temperature monitor system failure would alarm in the security monitoring area and security personnel would contact operations personnel. As discussed in SAR Section 8.2.8, under worst case conditions, cask temperature increases occur over a relatively long period of time, with no temperature limits reached for over a day. Assuming complete duct blockage of a HI-STORM storage cask, it would take 33 hours for the limiting component, the cask concrete, to reach its short-term temperature limit, while the fuel cladding and canister confinement boundary temperatures are substantially below their respective short-term limits at 72 hours. This would give operations personnel ample time to assess and resolve the problem.

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3.5 DECOMMISSIONING CONSIDERATIONS

This section of the principal design criteria satisfies 10 CFR 72.130, which requires provisions be made to facilitate decontamination of structures and equipment, minimize the quantity of radioactive wastes and contaminated equipment, and facilitate the removal of radioactive wastes and contaminated materials.

The PFSF shall be designed to facilitate safe and economical decommissioning activities in an expedient manner. Canister-based dry cask storage systems shall be used at the site because the canisters are designed to confine the spent fuel and facilitate its removal offsite. The spent fuel shall be sealed within the canister at the originating power plant to preclude contaminating other equipment and to enable the sealed canisters to be shipped and stored without having to open the canister or handle fuel assemblies. The PFSF shall be required to operate in a manner that supports decommissioning activities throughout the life of the facility.

The PFSF shall be designed to minimize the quantity of radioactive wastes generated and the amount of equipment that becomes contaminated. The canisters are not expected to have external surface contamination since measures are employed at the originating power plant to assure the external surfaces of the canisters are maintained in a clean condition. This minimizes the possibility of contaminating the Canister Transfer Building, canister transfer equipment, and storage casks. The Canister Transfer Building concrete floor, interior surfaces of the concrete transfer cells walls, and the low level waste holding cell shall be coated with paints or epoxy that accommodate and facilitate decontamination. Activation of the storage casks and concrete storage pads following long-term storage are expected to be negligible, allowing the release of the storage casks and pads as uncontrolled material. As canisters are shipped offsite and storage casks become available, the casks shall be reused for storage of any new incoming spent fuel canisters in order to minimize potential future waste.

The PFSF site will not use site drainage collection systems that would require decommissioning since there are no liquid effluents at the site.

Solid LLW created from health physics survey materials and dry decontamination shall be disposed of in LLW containers authorized for transport to a LLW disposal facility.

The PFSF shall be designed to facilitate the removal of radioactive wastes and contaminated materials. When the storage period for any particular canister of spent fuel is completed, the canister shall be transferred into a shipping cask and shipped offsite. The storage cask shall then be surveyed, and any contamination or activation products removed for disposal as LLW. The design of the storage casks, with the internal surfaces completely lined with steel, facilitates any decontamination efforts which may be required. Storage Cask components which are determined to be below specified activation and contamination levels shall be segregated for disposal as uncontrolled material.

The fences, electrical support structures, and other storage area equipment will not require special decommissioning activities since no contamination is expected to be transferred to these structures.

The PFSF shall be designed and operated to maintain radiation exposures ALARA during all decommissioning and decontamination activities.

Further decommissioning considerations are addressed by the storage system vendor in Section 2.4 of the HI-STORM SAR and in Appendix B of the PFSF License Application, "Preliminary Decommissioning Plan."

3.7 REFERENCES

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2. (deleted)

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4. ANSI/ANS 57.9, Design Criteria For An Independent Spent Fuel Storage Installation (Dry Storage Type), 1984.

5. Local Climatological Data, Annual Summary with Comparative Data for 1991, Salt Lake City, Utah, National Oceanic and Atmospheric Administration, National Environmental Satellite Data and Information Service, National Climatic Data Center, March 1992.

6. Ashcroft, G.L., D.T. Jensen and J.L. Brown, 1992, Utah Climate; Utah Climate Center, Utah State University.

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8. Regulatory Guide 1.76, Design Basis Tornado for Nuclear Power Plants, U.S. Nuclear Regulatory Commission, April 1974.

9. Private Fuel Storage Facility Storage Facility Design Criteria, Section 4.0, Geotechnical Design Criteria, Revision 2.

10. Regulatory Guide 1.29, Seismic Design Classification, U.S. Nuclear Regulatory Commission, September 1978.

11. 10 CFR 100, Appendix A, Seismic and Geologic Siting Criteria for Nuclear Power Plants.

12. Deterministic Earthquake Ground Motion Analysis, Private Fuel Storage Facility, prepared by Geomatrix Consultants, Inc. and William Lettis & Associates, SWEC Report No. 0599601-G(P05)-1, Revision 0.

13. Regulatory Guide 1.61, Damping Values For Seismic Design Of Nuclear Power Plants, U.S. Nuclear Regulatory Commission, October 1973.

14. ASCE-4, Seismic Analysis of Safety-Related Nuclear Structures and Commentary on Standard for Seismic Analysis of Safety-Related Nuclear Structures, American Society of Civil Engineers, 1986.

15. NUREG-0554, Single-Failure-Proof Cranes for Nuclear Power Plants, U.S. Nuclear Regulatory Commission, 1979.

16. ASME NOG-1, Rules for Construction of Overhead and Gantry Cranes (Top Running Bridge, Multiple Bridge), 1989.

TABLE 3.1-3
(Sheet 1 of 2)

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TABLE 3.1-3
(Sheet 2 of 2)

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TABLE 3.2-3

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TABLE 3.4-1

QUALITY ASSURANCE CLASSIFICATION OF STRUCTURES, SYSTEMS, AND COMPONENTS

IMPORTANT TO SAFETY	NOT IMPORTANT TO SAFETY
Classification Category A	Storage Facility Infrastructure
Spent Fuel Canister	Security and Heath Physics Building
	Administration Building
Classification Category B	Operations and Maintenance Building
Storage Cask	Intrusion Detection System
Transfer Cask	CCTV System
Associated Lifting Devices	Restricted Area Lighting
Canister Transfer Building	Security Alarm Stations
Canister Transfer Overhead Bridge Crane	Electrical Power - UPS
Canister Transfer Semi-gantry Crane	Electrical Power - Backup Diesel Generator
Seismic Support Struts	Electrical Power - Normal
	Yard/Building Lighting
Classification Category C	Cask Transporter
Cask Storage Pads	Radiation Monitors
	Temperature Monitoring System
	Communication Systems
	Fire Detection/Suppression
	Water Supply Systems
	Septic Systems
	Access Road
	Road Transport Components
	Railroad Line Components

TABLE 3.6-1
(Sheet 1 of 5)

SUMMARY OF PFSF DESIGN CRITERIA

DESIGN PARAMETERS	DESIGN CONDITIONS	APPLICABLE CRITERIA AND CODES
GENERAL		
PFSF Design Life	40 years	PFSF Specifications
Storage Capacity	40,000 MTU of commercial spent fuel	PFSF Specifications
Number of Casks	approximately 4,000 casks	PFSF Specifications
SPENT FUEL SPECIFICATIONS		
Type of Fuel	See Appendix B of HI-STORM C. of C.	Reference 34
Fuel Characteristics	See Appendix B of HI-STORM C. of C.	Reference 34
STORAGE SYSTEM CHARACTERISTICS		
Canister Capacity	<u>HI-STORM</u> 24 PWR assemblies/canister 68 BWR assemblies/canister	HI-STORM SAR, Section 1.1
Weights (maximum)	<u>HI-STORM</u> Storage Cask - 268,334 lbs. Loaded Canister - 87,241 lbs. Transfer Cask - 152,636 lbs. Shipping Cask - 153,080 lbs.	HI-STORM SAR, Table 3.2.1 " HI-STORM SAR, Table 3.2.2 Shipping SAR, Table 2.2.1

TABLE 3.6-1
(Sheet 2 of 5)

SUMMARY OF PFSF DESIGN CRITERIA

DESIGN PARAMETERS	DESIGN CONDITIONS	APPLICABLE CRITERIA AND CODES
STRUCTURAL DESIGN		
Wind	90 mph, normal speed	ASCE-7
Tornado	240 mph, maximum speed 190 mph, rotational speed 50 mph, translational speed 150 ft, radius of max speed 1.5 psi, pressure drop 0.6 psi/sec rate of drop	Reg. Guide 1.76
Tornado Missiles	115 lb. Wood plank, 190 ft/sec 287 lb. 6" schedule 40 pipe, 33 ft/sec 9 lb. 1" diameter steel rod, 26 ft/sec 1124 lb. Wooden utility pole, 85 ft/sec 750 lb. 12" schedule 40 pipe, 23 ft/sec 3990 lb. Automobile, 134 ft/sec	NUREG-0800, Section 3.5.1.4
Flood	N/A – PFSF is not in a flood plain and is above the PMF elevation	PFSF SAR Section 2.3.2.3
Seismic	0.53g, horz.(both directions) & 0.53 g vert. Design basis ground acceleration	10 CFR 72.102, Reg. Guide 1.165
Snow & Ice	P(g) = 45 psf	ASCE-7/County
Allowable Soil Pressure	Static = 4 ksf max Dynamic = Varies by footing type/size	PFSF SAR Section 2.6.1.12
Explosion Protection	The PFSF design and layout shall assure that the peak positive incident overpressure at important to safety SSCs does not exceed 1.0 psi from credible onsite and offsite explosions.	Reg. Guide 1.91
Ambient Conditions	Low Temperature = -30°F Max. Annual Average Temp. = 51°F Average Daily Max. Temp. = 95°F Humidity = 0 to 100 %	NOAA Data-Salt Lake City UT Climate Data UT Climate Data

TABLE 3.6-1
(Sheet 3 of 5)

SUMMARY OF PFSF DESIGN CRITERIA

DESIGN PARAMETERS	DESIGN CONDITIONS	APPLICABLE CRITERIA AND CODES
HI-STORM 100 Cask System Load Criteria	Canister: Internals: Storage Cask: Transfer Cask: See HI-STORM SAR, Table 2.2.6	ASME III, NB ASME III, NG ASME III NF, ACI-349 ASME III NF, ANSI N14.6
Cask Storage Pads Load Combinations	Normal Conditions $U_c > 1.4D + 1.7L$ $U_c > 1.4D + 1.7L + 1.7H$ Off-Normal Conditions $U_c > 0.75(1.4D + 1.7L + 1.7H + 1.7T)$ $U_c > 0.75(1.4D + 1.7L + 1.7H + 1.7T + 1.7W)$ Accident-Level Conditions $U_c > D + L + H + T + (E \text{ or } A \text{ or } W_t \text{ or } F)$ $U_c > D + L + H + T_a$	ANSI/ANS 57.9 ACI-349
Canister Transfer Building Structure Load Combinations (Reinforced Concrete)	Normal Conditions $U_c > 1.4D + 1.7L$ $U_c > 1.4D + 1.7L + 1.7H$ Off-Normal Conditions $U_c > 0.75(1.4D + 1.7L + 1.7H + 1.7T)$ $U_c > 0.75(1.4D + 1.7L + 1.7H + 1.7T + 1.7W)$ Accident-Level Conditions $U_c > D + L + H + T + (E \text{ or } A \text{ or } W_t \text{ or } F)$ $U_c > D + L + H + T_a$	ANSI/ANS 57.9 ACI-349
Canister Transfer Building Structure Load Combinations (Structural Steel)	Normal Conditions S and $S_v > D + L$ or $D + L + H$ Off-Normal Conditions $1.3 (S \text{ and } S_v) > D + L + H + W$ $1.5S > D + L + H + T + W$ $1.4 S_v > D + L + H + T + W$ Accident-Level Conditions $1.6S > D + L + T + (W_t \text{ or } E)$ $S_v > D + L + T + (W_t \text{ or } E)$	ANSI/ANS 57.9 ANSI/AISC N690

TABLE 3.6-1
(Sheet 4 of 5)

SUMMARY OF PFSF DESIGN CRITERIA

DESIGN PARAMETERS	DESIGN CONDITIONS	APPLICABLE CRITERIA AND CODES
Canister Transfer Building Foundation Load Combinations	Normal Conditions $U_f > 1.4D + 1.7L + 1.7G$ $U_f > 1.4D + 1.7L + 1.7H + 1.7G$ Off-Normal Conditions $U_f > 0.75 (1.4D + 1.7L + 1.7H + 1.7T + 1.7G)$ $U_f > 0.75 (1.4D + 1.7L + 1.7H + 1.7T + 1.7W + 1.7G)$ Accident-Level Conditions $U_f > D + L + H + T + G + (E \text{ or } A \text{ or } W_t \text{ or } F)$ $U_f > D + L + H + T_a + G$	ANSI/ANS 57.9 ACI-349
Canister Transfer Crane Designs	Type I, single-failure-proof 200 ton overhead bridge crane 150 ton semi-gantry crane	ASME NOG-1, NUREG 0554, & NUREG 0612
Canister Transfer Crane Load Combinations	Normal Conditions $P_c = P_{db} + P_{dt} + (P_{lr} \text{ or } P_p)$ $P_c = P_{db} + P_{dt} + P_{lr} + (P_v \text{ or } P_{ht} \text{ or } P_{hl}) + P_{wo}$ Off-Normal Conditions $P_c = P_{db} + P_{dt} + P_a + P_{wo}$ Accident-Level Conditions $P_c = P_{db} + P_{dt} + P_{cs} + P_e + P_{wo}$ $P_c = P_{db} + P_{dt} + P_e + P_{wo}$ $P_c = P_{db} + P_{dt} + P_{wt}$	ASME NOG-1
THERMAL DESIGN		
Design Temperatures (°F) (maximum)	HI-STORM / Norm / Off-norm / Acc Stor. cask conc. 200 350 350 Outer shell steel 350 600 600 Lid top plate 350 550 550 Inner shell steel & remaining steel 350 400 400 PWR Cladding 692 1058 1058 BWR Cladding 742 1058 1058	HI-STORM SAR, Table 2.2.3

TABLE 3.6-1
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SUMMARY OF PFSF DESIGN CRITERIA

DESIGN PARAMETERS	DESIGN CONDITIONS	APPLICABLE CRITERIA AND CODES
RADIATION PROTECTION/SHIELDING DESIGN		
Storage Systems Design Dose Rate Limits	<u>HI-STORM</u> cask side surface - 40 mrem/hr cask inlet/exit vent area - 60 mrem/hr cask top surface - 10 mrem/hr	HI-STORM SAR, Section 2.3.5.2
Individual Workers Dose Rate	Total eff. dose equiv.(TEDE) - 5 rem/yr Dose to eye lens - 15 rem/yr Dose to skin & extremities - 50 rem/yr	10 CFR 20.1201
Restricted Area Boundary Dose Rate	2 mrem/hr, max.	10 CFR 20.1301
Owner Controlled Area Boundary Dose Rate	25 mrem/yr whole body & 75 mrem/yr thyroid, max. 25 mrem/yr to any other critical organ 5 rem accident dose TEDE, or total organ dose equivalent of 50 rem (one time)	10 CFR 72.104 10 CFR 72.106
CRITICALITY DESIGN		
Control Method	Incorporation of Boral in canister fuel basket walls, and favorable geometry provided by the canister basket	HI-STORM SAR, 2.3.4.1
K_{eff}	< 0.95	NUREG-1536
CONFINEMENT DESIGN		
Confinement Method	Welded closed steel canister	HI-STORM SAR, 2.3.2.1
Confinement Barrier Design	HI-STORM canister: ASME III, NB	HI-STORM SAR, 2.3.2.1
Maximum Leak Rate	5.0E-6 atm cm^3 / sec (HI-STORM)	HI-STORM SAR Section 12, Tech. Spec Table 3-1;

CHAPTER 4

FACILITY DESIGN

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CHAPTER 4

FACILITY DESIGN

4.1 SUMMARY DESCRIPTION

This chapter identifies the Facility Design for the Private Fuel Storage Facility (PFSF). The Principal Design Criteria used as a basis for the Facility Design is described in Chapter 3. The design of the structures, systems, and components (SSCs) and how the design ensures quality standards are met in accordance with 10 CFR 72.122(a) is described.

The descriptions presented in this chapter specifically focus on SSCs that are classified as being Important to Safety; SSCs that are not Important to Safety are also addressed where appropriate. The SSCs that are classified as being Important to Safety are identified in Chapter 3 as the storage systems, cask storage pads, canister transfer cranes, and Canister Transfer Building.

The PFSF utilizes the HI-STORM 100 Cask System (HI-STORM) designed by Holtec International (Holtec). Holtec submitted a Safety Analysis Report (SAR) to the U.S. Nuclear Regulatory Commission (NRC) for the HI-STORM system (Reference 1). The NRC has issued a Certificate of Compliance for Holtec's HI-STORM 100 storage cask system (Reference 79).

The HI-STORM storage system SAR contains the generic design of its storage system and transfer equipment. This chapter summarizes the generic design and how the generic design complies with the site-specific criteria at the PFSF.

The PFSF is designed in accordance with the General Design Criteria set forth in 10 CFR 72, Subpart F. Table 4.1-1 summarizes compliance with these criteria.

4.1.1 Location and Layout

The PFSF is located on the Skull Valley Indian Reservation in northwestern Utah, approximately 27 miles west-southwest of Tooele City. The site location is shown on Figure 1.1-1. The PFSF site layout is shown on Figure 1.1-2 and the PFSF general arrangement is shown on Figure 1.2-1.

4.1.2 Principal Features

The principal features of the PFSF consist of the storage area, including cask storage pads, the Canister Transfer Building, shown on Figure 4.1-1, and the Security and Health Physics Building, shown on Figure 4.1-2. The cask storage pads, the Canister Transfer Building, and the Security and Health Physics Building are located within the Restricted Area (RA). The Canister Transfer Building facilitates the transfer of the canister from the shipping cask to the storage cask and houses the overhead bridge and semi-gantry cranes used in the transfer process. The Security and Health Physics Building is the entrance point for the RA and houses offices and equipment for security and health physics personnel. The RA provides security and physical protection of spent fuel and restricts access because of potential radiation doses from the spent fuel. The RA consists of approximately 99 acres of storage area surrounded by a chain link security fence, 20 ft isolation zone, and chain link nuisance fence. The design capacity of the RA is approximately 500 concrete cask storage pads capable of storing up to 8 storage casks each for a total of approximately 4,000 storage casks. The storage pad area is surfaced with compacted gravel to enable transport of the storage casks from the Canister Transfer Building to the storage pads.

4.2 STORAGE STRUCTURES

The storage SSCs are used to safely store spent fuel at the PFSF. The storage SSCs at the PFSF consist of the following:

- HI-STORM 100 Cask System
- Cask storage pads

The storage SSCs are designed to ensure adequate safety and to mitigate the effects of site environmental conditions, natural phenomena, and accidents in accordance with 10 CFR 72.122(b) and 10 CFR 72.128(a). The SSC design is described in Chapter 8 and in the HI-STORM SAR.

The storage SSCs are designed to permit testing, inspection, and maintenance of the systems in accordance with 10 CFR 72.122(f). The acceptance test and maintenance program of the storage system is specified in HI-STORM SAR Chapter 9. Because of the passive nature of the storage system design, inspection and maintenance requirements are minimal. Surveillance requirements associated with operational control and limits are described in Chapter 10. Inspection and testing is performed in accordance with the Quality Assurance program described in Chapter 11.

Each of the storage SSCs are described in the following sections. Figures are provided to illustrate the components and their function.

4.2.1 HI-STORM 100 Cask System

The HI-STORM storage system consists of metal canisters, concrete storage casks, and associated transfer equipment. The following sections provide an analysis of the HI-STORM storage system canister and storage cask design relative to the storage requirements of the PFSF. Types and characteristics of fuel to be stored, site environmental conditions, support structures, and support systems are shown to be within the design criteria envelope of the HI-STORM SAR, thus ensuring no unanalyzed safety conditions for storage using the HI-STORM storage system exist at the PFSF. The HI-STORM canister transfer equipment is described in Section 4.7.3.

4.2.1.1 Design Specifications

The design, fabrication, and construction specifications used for the HI-STORM storage system components are identified in the HI-STORM SAR Table 2.2.6 and are summarized as follows:

- Metal canister -

Pressure boundary	ASME BPVC Section III, Subsection NB
Internal assembly	ASME BPVC Section III, Subsection NG

- Concrete storage cask -

Steel	ASME BPVC Section III, Subsection NF
Concrete	ACI-349

BWR assembly (HI-STORM SAR Tables 2.1.7 and 2.1.8). The analysis assumed HI-STORM storage casks are in an array, subjected to an 80° F annual average ambient temperature, with solar radiation. The annual average temperature takes into account both day and night, summer and winter temperatures throughout the year. The annual average temperature is the principal design parameter in the storage system design analysis because it establishes the basis for demonstration of long-term spent nuclear fuel integrity. The long-term integrity of the spent fuel cladding is a function of the averaged ambient temperature over the entire storage period, which is assumed to be at the maximum average yearly temperature in every year of storage for conservatism in the cladding service life components. The results of this analysis are presented in Tables 4.4.9 and 4.4.10 of the HI-STORM SAR for MPC-24 and MPC-68 canisters, respectively. The results, summarized in Table 4.2-3 of this SAR, indicate that temperatures of all components are within normal condition temperature limits.

Holtec considered stainless steel clad fuels in the thermal analysis, as discussed in HI-STORM SAR Section 4.3.2. Stainless steel cladding is less conductive than zircaloy clad fuel and the net thermal resistance of a basket full of stainless steel clad fuel is greater, which would result in higher cladding temperatures for stainless steel fuel assemblies having the same decay heat generation rate as zircaloy clad fuel. However, the design basis decay heat for stainless steel clad fuel is significantly lower than that of zircaloy clad fuel, as noted previously, and the allowable temperature limit for stainless steel cladding is considerably higher than for zircaloy cladding. Holtec determined that the reduction in heat duty is much more pronounced than the nominal increase in the resistance to heat transfer, and concluded that the peak cladding temperature for stainless steel clad fuel will be bounded by the results for zircaloy clad fuel and a separate analysis for stainless steel clad fuel was not required.

HI-STORM SAR Section 11.1.2 evaluates temperatures of the HI-STORM storage system for a maximum off-normal daily average ambient temperature of 100° F, an increase of 20° F from the normal conditions of storage discussed above. This off-normal temperature condition is based on a 24 hour average solar load in accordance with 10 CFR 71, which represents extreme environmental conditions or off-normal conditions. The maximum off-normal temperatures were calculated by adding 20° F to the maximum normal temperatures from the highest component temperature for MPC-24 and MPC-68. All the maximum off-normal temperatures are below the short term condition design basis temperatures (HI-STORM SAR Table 2.2.3). Therefore, all components are within allowable temperatures for the 100° F ambient temperature condition.

The thermal analysis in the HI-STORM SAR discussed above includes the following global assumptions: a) the concrete pad is assumed to be an insulated surface, i.e., no heat transfer to or from the pad is assumed to occur; b) adjacent casks are assumed to be sufficiently separated from each other (i.e., cask pitch is sufficiently large) so that their ventilation actions are autonomous from each other; c) the cask is assumed to be subject to full solar insolation on its top surface as well as view-factor adjusted solar insolation on its lateral surface. Second order effects such as insolation heating of the concrete pad, heating of feed air traveling downward between casks and entering the inlet ducts of the reference cask, and radiative heat transfer from adjacent spent fuel casks were not explicitly modeled in the HI-STORM SAR analysis.

In order to address these second order effects, PFS had the HI-STORM storage cask vendor, Holtec, perform a revised analysis (Reference 60). The revised analysis specifically applies to HI-STORM storage casks at the PFSF site and assumes the following: a) exposed areas of the storage pad and the storage casks are heated by the sun, with the intensity of radiation derived from 10 CFR 71.71(c); b) conductive heat transfer takes place between both the pad and the

4.2.1.5.5 Confinement Design

Confinement design for the HI-STORM storage system is addressed in HI-STORM SAR Chapter 7. The confinement vessel of the HI-STORM storage system is the canister, which provides confinement of all radionuclides under normal, off-normal, and accident conditions in accordance with 10 CFR 72.122(h). The canister consists of the canister shell, a bottom base plate, the canister lid, and the canister closure ring, which form a totally welded vessel for the storage of spent fuel assemblies. The canister requires no valves, gaskets, or mechanical seals for confinement. All components of the confinement system are classified as Important to Safety.

The canister is a totally seal-welded pressure vessel designed to meet the stress criteria of ASME BPVC Section III (Reference 11), Subsection NB. No bolts or fasteners are used for closure. All closure welds are examined using the liquid penetrant method and helium leak tested to ensure their integrity. Two penetrations are provided in the canister lid for draining, vacuum drying, and backfilling during loading operations. Following loading operations, vent and drain port cover plates are welded to the canister lid. A closure ring, which covers the penetration cover plates and welds is welded to the canister lid providing redundant closure of the canister vessel. The loading and welding operations are performed at the originating power plant, ensuring total confinement of the canister upon arrival to the PFSF. There are no confinement boundary penetrations required for canister monitoring or maintenance during storage.

The confinement features of the HI-STORM storage system meet the PFSF design criteria in Section 3.3.2 for confinement barriers and systems.

4.2.2 (deleted)

4.2.3 Cask Storage Pads

The design criteria for the cask storage pads are described in Chapter 3. The analysis methods and resulting design of the pads are described below.

4.2.3.1 Design Specifications

The design of the cask storage pads is in accordance with ANSI/ANS-57.9 (Reference 14) and ACI 349 (Reference 15) as identified in Chapter 3.

The cask storage pads are independent structural units constructed of reinforced concrete. Each pad is 30 ft wide by 64 ft long and 3 ft thick. The size of the pad is based on a center to center spacing of 15 ft for the storage casks. The ends of the storage pad are provided with an additional 2 ft in length to support both tracks of the cask transporter on the pad. The pads are nearly flush with grade for direct access by the cask transporter. Each cask storage pad is capable of supporting 8 loaded storage casks.

An independent modular pad design was chosen to simplify the pad analysis (i.e. minimize the number of cask placement combinations) and to minimize the effects of thermal expansion. The modular pad design also provides for ease of construction by limiting the number of concrete pad construction and/or expansion joints required and allows for staged construction of the facility.

The cask storage pad design is based on a maximum loaded storage cask weight of 360,000 lbs. This maximum weight envelopes the maximum loaded weight of the storage casks proposed for use at the PFSF.

The HI-STORM storage casks proposed for use at the PFSF are the MPC-24 (PWR fuel) and the MPC-68 (BWR fuel) with maximum weights of 348,321 lb. and 355,575 lb., respectively, as shown in the HI-STORM SAR Table 3.2-1.

The cask storage pad design also considers the weight of the loaded storage casks in combination with the seismic loads due to the site-specific probabilistic seismic hazard assessment (PSHA) design basis earthquake (0.53g horizontal in two directions and 0.53g vertical – See Section 8.2.1.1).

4.2.3.2 Plans and Sections

The site plan, which shows the locations of the concrete storage pads, is shown in Figure 1.2-1. A typical concrete storage pad plan, cross section, and details are shown in Figure 4.2-7.

4.2.3.3 Function

The function of the cask storage pads is to provide a level and stable surface for placement and storage of the storage casks containing the spent fuel canisters.

4.2.3.4 Components

The components of the cask storage pads consist of the materials of construction, which include concrete with a minimum 28-day compressive strength of 3,000 psi and reinforcing steel with a minimum yield strength of 60,000 psi.

4.2.3.5 Design Bases and Safety Assurance

The design bases for the cask storage pads are identified in Chapter 3.

The cask storage pads are classified as being Important to Safety in order to provide the appropriate level of quality assurance in the design and construction. This provides for the safety assurance that the cask storage pads will perform their intended function.

4.2.3.5.1 Storage Pad Analysis

The reinforced concrete pads were analyzed and designed in accordance with nuclear industry standard structural analysis and design methods (Reference 16). The static and dynamic analyses for evaluating the concrete pad response displacements and internal stresses have used standard finite element analysis computer programs CECSAP (Reference 17) and SASSI (Reference 18) computer codes.

Static analyses have been performed for the dead load and design live (storage cask) loads using the CECSAP computer program. Dynamic analyses have been performed for the site-specific probabilistic seismic hazard assessment (PSHA) design basis ground motion using the CECSAP computer program. In addition, a separate dynamic analysis was performed using the SASSI computer program to more rigorously account for the effect of soil-structure interaction. These static and dynamic analyses confirm the structural adequacy of the reinforced concrete storage pad for supporting the storage casks when subjected to the design loading conditions. The results of the pad dynamic analysis using SASSI confirmed validity and indicated conservatism of the corresponding results using CECSAP.

The structural analyses of the pad used a three-dimensional flat-shell finite element model for the concrete pad. The finite element model mesh developed for the pad is shown in Figure 4.2-8. A total of 264 flat-shell finite elements have been used to model the concrete pad. Gross uncracked stiffnesses have been used for the model. The finite element mesh was developed with the consideration that it would produce reasonably refined distribution of internal forces and moments. Also, the nodal points of the mesh coincided with the locations of the static and dynamic loadings associated with one to eight casks to be applied on the pad. These loadings are lumped to four points on the outer circular perimeter of each cask corresponding to the four quadrants of the cask. Various cask loading patterns were considered to determine the maximum pad internal stresses.

To represent the soil support condition of the pads for the long-term static (i.e. dead and live) load conditions, vertical boundary soil springs tributary to each node of the pad finite element model were used in the CECSAP static finite element model. The spring stiffness values for the static loading cases were developed from the modulus of subgrade reaction of the supporting soil medium (Reference 71). The spring stiffness values were varied to account for uncertainties in the soil properties by using lower and upper bound conditions (Reference 40). For the short-term PSHA design basis earthquake loading condition, three-component (two horizontal and one vertical) boundary soil springs and dashpots representing the dynamic soil stiffnesses and radial damping characteristics of the supporting soil medium were used to connect to each node of the pad model. The soil spring stiffness (and its associated soil mass), and radial damping coefficient tributary to each node were derived from the lumped soil spring stiffness, mass, and damping coefficient values based on the procedure in ASCE-4 (Reference 20). For the dynamic analysis using the SASSI computer program, the soil support to the pad was represented by three-component (two horizontal and one vertical) complex-valued, frequency-dependent, dynamic soil impedance functions that are connected to each node of the pad finite element model. The soil impedance functions were computed numerically within the SASSI computer program based on the free-field profile and dynamic properties of the soil layers underlying the pad.

The pad structural analyses included both static and dynamic analyses. The static analysis evaluated the pad response stresses due to the dead and (cask) live loads. The dynamic analysis evaluated the pad response due to the PSHA design basis earthquake loadings. The pad responses obtained from these analyses were then combined to give the combined maximum response values in accordance with the applicable load combinations. The combined response values were used for checking the structural adequacy of the concrete pad and the dynamic soil bearing capacity and overturning and sliding stability. The static and dynamic pad analyses performed for the pad are separately described below.

A. Static Analysis

The static pad analysis, using the CECSAP finite element model of the pad shown in Figure 4.2-8, was conducted for the dead load equal to the gravitational dead weight of the pad and the live load of the casks. The static loading cases were performed under a bounding range of soil spring values (stiffness) to account for potential uncertainties in the soil properties used in the analyses. Each load case was evaluated for lower and upper bound soil values using the soil property provided in Reference 40 and 71.

The live loads from three loading patterns of 2, 4, and 8 fully-loaded casks were considered. The weight of one fully-loaded cask considered was 360 kips. To simulate the condition of one fully-loaded cask being transported onto the pad, one additional cask loading pattern consisting of 7 fully-loaded casks and one fully-loaded cask being lifted by a cask transporter on the pad having a weight of 135 kips (Reference 21) was also considered. For conservatism, a dynamic impact (or amplification) factor equal to 2.0 was used for the load of one fully-loaded cask plus the transporter to account for any dynamic effect that may arise while transporting the cask.

Based on the results of this analysis, the cask-loading pattern that produces the highest pad internal stresses is that of four casks on the pad and the worst-case loading that produces the largest soil bearing pressure is that of 7 casks plus one cask being carried by the transporter. The maximum response results obtained from the static analyses are summarized in Table 4.2-7.

Soil pressures beneath the storage pad were also verified to be within the acceptance criteria for static loading conditions. Actual soil bearing pressures were calculated beneath the pad and compared to the allowable soil bearing pressures identified in Section 2.6.1.12.1 for various static load combinations. The maximum static soil pressure was calculated to be 4.0 ksf under the static (dead plus live [snow plus 7 casks plus the loaded transporter]) loading condition. The maximum calculated static soil pressure is less than the minimum allowable soil bearing pressure for static loads (4.35 ksf, as shown in Table 2.6-6).

B. <u>Dynamic Analysis</u>

The dynamic analysis was performed to determine the pad response stresses under the PSHA design basis earthquake loading. The dynamic loading cases were analyzed for a bounding range of soil spring values (stiffness) to account for potential uncertainties in the soil properties used in the analyses. Each load case was evaluated for lower bound, best estimate, and upper bound soil values using the data and soil properties provided in References 40 and 71.

The global seismic time-history response analysis was performed utilizing a series of cask-pad-soil interaction models to represent the dynamic characteristics of one to eight casks supported on the pad, which is supported on the site soil. To account for uncertainties in the frictional resistance to horizontal movements of casks on the pad, the friction coefficient between the cask base and the concrete pad considered in these analyses was varied from a lower-bound value of 0.2 to an upper-bound value of 0.8. The case with the lower-bound friction results in an upper-bound estimate of the sliding displacements of the casks on the pad and a lower-bound estimate of the cask dynamic forces acting on the pad, whereas the upper-bound friction case results in a lower-bound estimate of the sliding displacements and an upper-bound estimate of the cask dynamic forces acting on the pad. Thus, for the purpose of determining the upper-bound seismic stresses in the pad, the cask dynamic force time histories resulting from the upper-bound friction case were conservatively used as the dynamic forcing function inputs to the pad.

The dynamic forcing function inputs were obtained from the Holtec site-specific cask stability analysis for the HI-STORM storage cask (Reference 7). These dynamic forcing time histories were evaluated for each cask at four points that are equally-spaced along the circular outer perimeter of the cask base. At each point, a set of three-component (two horizontal and one vertical) dynamic forcing time histories was evaluated, which represents the lumped dynamic reaction forces of the pad to the cask within the four quadrants of each circular cask-base area.

In evaluating the pad dynamic stresses due to the dynamic forces of the casks acting on the pad, the finite element model of the pad-soil system (using CECSAP) was used and the dynamic force time histories of the casks were applied on the pad as nodal forcing functions. To reasonably bound the various cask loading patterns, the same 2, 4, and 8-cask loading configurations that were considered in the static analyses were analyzed. The maximum values of the pad response shear forces and bending moments resulting from the analysis were then evaluated and used for checking the structural adequacy of the pad design. The maximum values of the three-component (two horizontal and one vertical) soil-spring reaction forces were also evaluated and used for checking the overturning stability, the soil bearing capacity, and the sliding stability of the pad for the dynamic loads due to the design basis ground motion. Refer to Section 2.6.1.12.1 for a discussion of these analyses.

To provide a comparison and an assessment of the accuracy and conservatism of the dynamic analysis results from the CECSAP pad-soil system model, an additional dynamic analysis was also performed for a selected dynamic loading case using the SASSI computer program. The dynamic response results obtained from this SASSI finite element analysis were compared with the corresponding results obtained from the CECSAP analysis. This comparison indicates that the CECSAP analysis results are conservative relative to the corresponding SASSI results by a margin of greater than 20 percent.

The results of the maximum dynamic response values obtained from the dynamic analyses described above are summarized in Table 4.2-8. Based on these results, the loading that produces the maximum dynamic pad internal stresses and soil pressures is that of two casks, and the loading that produces the largest seismic horizontal soil reaction forces is that of 8 casks. These values were included in the analyses of dynamic bearing capacity and sliding stability of the pad, which are discussed in Section 2.6.1.12.1.

The maximum dynamic soil pressures, which include earthquake loadings, were also calculated for the pad dead load plus 2 casks, 4 casks, and 8 casks. The resulting soil pressure distribution was converted to an average soil pressure over an effective pad width and compared to the allowable dynamic soil pressures. The maximum soil pressure under dynamic (dead plus live plus the PSHA design basis earthquake) loading condition is 6.53 ksf, which is below the minimum allowable dynamic soil bearing pressure for dynamic loads (10.4 ksf, as shown in Table 2.6-7).

The sliding stability of the cask storage pads was also analyzed using the dynamic forces applicable for the PSHA design basis earthquake and found to be adequate. Refer to Section 2.6.1.12.1 for details and results of these analyses.

4.2.3.5.2 Storage Pad Design

The storage pad design is a 3-ft thick reinforced concrete slab with #10 longitudinal and transverse horizontal reinforcing bars spaced at 12 inches on center each way at the top face and #10 longitudinal and transverse horizontal reinforcing bars spaced at 6 inches on center each way at the bottom face of the pad. The top and bottom face horizontal reinforcements are tied through the thickness of the pad by #8 vertical shear reinforcing bars spaced at 12 inches on center each way in two ways uniformly distributed over the entire pad. The concrete has a minimum 28-day compressive strength of 3,000 psi and the reinforcing steel has a minimum yield strength of 60,000 psi.

Static design moments are based on the 1.4D + 1.7L + 1.7H load combination. The design provides an ultimate static moment capacity in the longitudinal pad direction of $-M_{yy}$ = 332 k-ft/ft and $+M_{yy}$ = 182 k-ft/ft. The capacities exceed the demand moments of $-M_{yy}$ = 114 k-ft/ft and $+M_{yy}$ = 124 k-ft/ft. The moment capacity to demand ratios in the transverse pad direction is less than that for the longitudinal direction.

Static design shear values are also based on the 1.4D + 1.7L + 1.7H load combination. The design provides an ultimate static beam shear capacity of 134 k/ft and an ultimate static punching shear capacity of 134 k/ft. The capacities exceed the demand shears of V_u (beam) = 19 k/ft and V_u (punching) = 9 k/ft.

Dynamic (or accident-level) design moments are based on the D + L + H + E load combination. The design provides an ultimate dynamic (impulse or impactive) moment capacity in the longitudinal pad direction of $-M_{yy}$ = 367 k-ft/ft and $+M_{yy}$ = 202 k-ft/ft. The capacities exceed the demand moments of $-M_{yy}$ = 332 k-ft/ft and $+M_{yy}$ = 131 k-ft/ft. The design also provides a moment capacity in the transverse pad direction of $-M_{xx}$ = 350 k-ft/ft and $+M_{xx}$ = 201 k-ft/ft. These exceed the demand moments of $-M_{xx}$ = 348 k-ft/ft and $+M_{xx}$ = 154 k-ft/ft.

Dynamic (or accident-level) design shear values are also based on the D + L + H + E load combination. The design provides an ultimate dynamic beam shear capacity of 148 k/ft and an ultimate dynamic punching shear capacity of 148 k/ft. The capacities exceed the demand shears of V_u (beam) = 80 k/ft and V_u (punching) = 147 k/ft.

Therefore, the storage pad as designed provides adequate strength for accommodating the design loading conditions.

4.2.3.5.3 Storage Pad Settlement

The relationship of major foundations to subsurface materials is contained in Section 2.6.1.6. Storage pad soil settlement analyses are described in Section 2.6.1.12.1.

The in situ soil is suitable for supporting the cask storage pads, but settlements are expected to occur. Analyses were performed to calculate the estimated settlement of the storage pads from the weight of the pad with 8 fully loaded casks in place. The nominal soil bearing pressure for this case is approximately 1.9 ksf, and the total estimated settlement of the pad is approximately 3.3 inches.

In order to accommodate the total estimated settlement, the storage pads will be constructed 3.5 inches above adjacent finished grade. Exposed edges of the pad will be chamfered, and the crushed rock surface materials will be feathered to meet the edges of the raised pads for transporter access.

Uniform downward settlement has no adverse effect on either the pad or the casks, it only lowers the final elevation of the storage pad. The storage pads will be constructed 3.5 inches above the surrounding grade to allow for settlements and yet maintain the surface drainage scheme at the site. The first pads constructed and loaded will be monitored for settlements to confirm the calculated settlements and make future adjustments, if necessary. The temporary uniform differential settlement of the pad from partial cask placements causes no loss of structural integrity to the pad. The storage pad is not susceptible to subsurface failures associated with liquefaction since the site is not subject to liquefaction, as discussed in Section 2.6.4.8.

4.2.3.5.4 Cask Stability

Cask stability ensures the storage casks will not tip over or slide excessively during a seismic event. The generic cask stability analyses in the HI-STORM SAR does not consider soil-structure interaction, which can amplify seismic accelerations. Consequently, site-specific cask stability analyses, performed by Holtec (Reference 61) demonstrate the storage casks will not tip over or slide excessively during the PFSF design basis ground motion. The cask stability analyses are described in detail in Section 8.2.1. The cask storage pad is designed for the loads generated from the site-specific cask stability analyses and will provide the required support for the storage casks.

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4.5 SHIPPING CASKS AND ASSOCIATED COMPONENTS

Spent fuel shipping casks are used to transport the spent fuel canisters from the originating power plants to the PFSF and later offsite. The shipping casks are designed to protect the canisters from the effects of environmental conditions, natural phenomena, and accidents in accordance with 10 CFR 71. Shipping casks are not licensed under 10 CFR 72. However, since the shipping casks are used to transport spent fuel to and from the PFSF and are part of the canister transfer process in the Canister Transfer Building, this section provides a brief summary of the shipping casks and associated components.

The shipping casks are shipped to the PFSF and shipped offsite at a later date complete with impact limiters, a shipping cradle, and tie downs. The shipping casks are shipped from the railroad mainline to the PFSF either by rail on a railroad line or by highway. Shipment by highway requires the shipping casks be transferred from the rail car to a heavy haul tractor/trailer at an intermodal transfer point. During the rail to trailer transfer, the cask and shipping components remain an integral unit under 10 CFR 71 packaging requirements. At the PFSF, the shipping cask is unloaded from the rail car or heavy haul tractor/trailer and moved to a canister transfer cell where the shipping cask is opened and the canister is removed. After the canister is unloaded, the shipping cask is resealed and sent back to the power plants for reloading of another sealed canister of spent fuel.

The shipping components addressed in this section are:

- HI-STAR shipping cask system
- Shipping cask repair and maintenance area
- Skull Valley Road / Intermodal transfer point
- Low Corridor rail line

The shipping casks and associated components are described below. Figures are provided to illustrate the systems and their function.

4.5.1 HI-STAR Shipping Cask System

The HI-STAR system is a shipping system used to ship spent fuel from the originating power plants to the PFSF. The HI-STAR (Holtec International Storage, Transport, and Repository) is a spent fuel packaging design in compliance with DOE's design procurement specifications for multi-purpose canisters and large transportation casks. The HI-STAR system consists of the same sealed metal canister as used in the HI-STORM storage system, which is confined within a metal overpack or cask with impact limiters. Holtec submitted a SAR to the NRC in accordance with 10 CFR 71 for the HI-STAR system (Reference 3) and the NRC has issued a Certificate of Compliance for the HI-STAR 100 shipping cask system (Reference 80). The HI-STAR system components are shown on Figure 4.5-1. Details of the system and design parameters are addressed in the HI-STAR shipping SAR.

4.5.2 (deleted)

4.5.3 Shipping Cask Repair and Maintenance

If shipping cask repair or maintenance activities are necessary, they will be conducted at the Operation and Maintenance Building or at a vendor designated location. No special contamination control measures are anticipated for repair or maintenance activities since the spent fuel is contained within a sealed canister and the shipping casks used for the PFSF do not enter any nuclear plant spent fuel pools and therefore, remain free of radioactive contamination.

Health physics surveys will be taken on all incoming canisters as normal receiving operations at the PFSF. In the event contamination above acceptance levels is discovered and cannot be removed, the canister will be shipped back to the originating nuclear power plant for canister decontamination and/or spent fuel repackaging.

4.5.4 Skull Valley Road / Intermodal Transfer Point

4.5.4.1 Intermodal Transfer Point

Shipments that utilize the Skull Valley Road / intermodal transfer point are moved by the use of roads from the rail mainline to the PFSF using heavy-haul tractor/trailers. The intermodal transfer point is located 1.8 miles West of Timpie, approximately 24 miles north of the PFSF. The intermodal transfer point equipment is designed to accommodate transfer of the shipping casks from the rail car to the heavy haul tractor/trailer unit for highway shipping. The intermodal transfer point consists of rail sidings off the Union Pacific Railroad mainline, a 150 ton gantry crane, and a tractor/trailer yard area. The gantry crane is a single-failure-proof crane to preclude the accidental drop of a shipping cask even though the cask is designed to withstand such drops in accordance with 10 CFR 71. The crane is housed in a weather enclosure, which provides a clean, dry environment for transfer of the shipping cask.

The intermodal transfer point is shown on Figure 4.5-3.

4.5.4.2 Shipping Cask Heavy Haul Tractor/Trailer

Heavy haul transport tractor/trailers are used to transport the shipping cask from the intermodal transfer point to the PFSF by highway. The maximum weight of a loaded shipping cask with impact limiters and shipping cradle is approximately 142 tons, which requires the use of overweight trailers. The heavy haul tractor/trailers are designed to accommodate road conditions at the intermodal transfer point, frontage road, Skull Valley Road, and PFSF. A minimum of 2 heavy haul tractor/trailer units would be used if the casks were transported by highway from the ITP to the PFSF. The units are designed to travel at low speeds and are 12 ft wide with multiple wheel sets to provide stable transport of the shipping cask. Based on vendor information from three of the largest trailer manufacturers for this type of trailer, the heavy haul trailers range from 150 ft to 180 ft in length. The trailers use up to 100 tires to distribute the weight within typical highway limits. However, use of these trailers usually requires permitting due to the overall weight and length. The trailers are articulated, that is they can pivot in several places and include steerable axles to accommodate tight radius turning. The turning radius ranges from 75 ft to 150 ft, depending on whether steerable dollies are used. The tractor/trailers will usually be stored in either the Canister Transfer Building truck bay or in the intermodal transfer point enclosure in preparation for their next assigned task. Both buildings are designed to fully enclose the tractor/trailer unit. Maintenance activities will be conducted at the Operation and Maintenance Building, except such maintenance duties that are complex enough in nature that they require off-site contracted major maintenance. It is anticipated that contract facilities within the area would be used for such items as engine overhaul, etc.

The unit is classified as not Important to Safety since safety of the spent fuel canister is maintained by the shipping cask. A typical heavy haul transport tractor/trailer unit is shown on Figure 4.5-4.

4.5.5 Low Corridor Rail Line

4.5.5.1 Rail Line

Shipments that utilize the railroad line continue on from the rail mainline to the PFSF by rail car. A rail line will be built from the Union Pacific mainline located at Low Junction to the PFSF. The rail line is designed to standard railroad load, grade, and clearance requirements per the Union Pacific Railroad and industry standards to facilitate use of Union Pacific and standard railroad equipment.

The Low Corridor rail line is shown on Figure 4.5-6

4.5.5.2 Shipping Cask Rail Car

The railcars will be heavy duty 3 axle or span bolster 4 axle (2 sets of 2-axle trucks) railcars with a capacity of approximately 150 tons similar to those used by the Department of Defense for their spent fuel shipments. The maximum weight of the shipping cask with impact limiters and shipping cradle on the railcar is approximately 142 tons as discussed in Section 4.5.4.2, which would be within the allowable load for a 150 ton railcar. The Canister Transfer Building cask load/unload bays are designed with railroad tracks to facilitate rail car shipments where the shipping casks would be unloaded or loaded.

The radius of the track for railcars is dependent on various factors such as car length. The final design is not complete on the railcar, so the turning radius of the cask car has not been determined. However, direct rail transportation to the PFSF has been designed using mostly 3 degree curves (1909 ft radius) with the tightest curve being a 10 degree curve (573 ft radius). The railcars, which typically will be in transit to pickup more spent fuel, will be stored on the railroad storage siding at the PFSF when not in use (See SAR Figure 1.2-1). If the intermodal transfer point is utilized, parking for the cars when not in use would either be provided at the intermodal transfer point or at leased space somewhere in the vicinity. Routine maintenance will be performed at the PFSF or the intermodal transfer point, depending on the case. Major overhauls and maintenance would have to be in a privately operated railroad equipment servicing shop approved for such activities and inspections.

The railcars are classified as not Important to Safety since spent fuel safety functions are maintained by the shipping cask. A typical 150-ton railcar similar to what could be used at PFSF is shown on Figure 4.5-5.

4.7 SPENT FUEL HANDLING OPERATION SYSTEMS

The spent fuel handling systems are provided to transfer the spent fuel canisters from the shipping cask to the storage cask, and eventually back to the shipping cask for transporting offsite. During transfer operations, the spent fuel remains confined within the sealed metal canister at all times. No individual spent fuel assemblies are handled at the PFSF.

The spent fuel handling systems used to handle spent fuel at the PFSF consist of the following:

- Canister Transfer Building
- Canister transfer cranes
- HI-STORM transfer equipment
- Cask transporter

The spent fuel handling systems are designed to ensure adequate safety and to withstand the effects of site environmental conditions, natural phenomena, and accidents in accordance with 10 CFR 72.122(b) and 10 CFR 72.128(a).

The handling SSCs are designed to permit testing, inspection, and maintenance in accordance with 10 CFR 72.122(f). The acceptance test and maintenance program of the storage systems is specified in HI-STORM SAR Chapter 9.

Regulation 10 CFR 72.122(l) requires that the storage systems be designed to allow ready retrieval of spent fuel for further processing or disposal. The canister based storage systems utilized at the PFSF accommodate this requirement. At the end of the storage period, the sealed canisters will be shipped offsite to the federal government.

Chapter 5 outlines the procedure for canister transfer from a storage cask into a shipping cask and offsite.

Retrieval of individual spent fuel assemblies from the canister before offsite shipping is not anticipated. As described earlier in this chapter, the canister is designed to withstand all normal, off-normal, and accident-level events. Nevertheless, retrieval of the spent fuel from the canister can be achieved if necessary. In the event the spent fuel assemblies require unloading prior to being shipped offsite, the canister will be shipped back to the originating nuclear power plant via a shipping cask (if the originating plant is still available), or to another facility licensed to perform such operations, where the individual spent fuel assemblies will be transferred into a different canister.

Each of the spent fuel handling systems is described in the following sections. Figures are provided to illustrate the major components of the systems and their function.

tornado wind will be greater in that direction. The exposed building area is approximately (270')(90') = 24,300 ft^2. The lateral force due to tornado wind is determined from NUREG-0800, Section 3.3.2 as:

Lateral force = 0.00256 V^2 x wall pressure coefficient x wall area

Where wall pressure coefficient = 0.8 (windward wall) + 0.5 (leeward wall)

Therefore, the lateral force = 0.00256 $(240)^2$ (0.8 + 0.5)(24,300 ft^2) = 4,658,135 lb. ≅ 4,658 kips. By comparison, the lateral force due to the design basis earthquake in the E-W direction is 36,500 kips. The earthquake force is taken from Calculation 05996.02-SC-5 (Reference 44) and excludes the force due to acceleration of the base mat.

Out of plane pressures on the exterior walls due to tornado loads are caused by the 240 mph wind velocity and the 1.5 psi pressure drop. The worst pressure is outward on the side walls and is equal to 0.00256 $(240)^2$ (0.7) + 1.5 psi (144 in^2/ft^2) = 319 psf. The out of plane seismic inertia load, based on a typical horizontal acceleration of 0.9 g results in an equivalent pressure for a 2 ft. thick wall of 2'(150 pcf)(0.9) = 270 psf. Although the pressure due to tornado is slightly higher than that due to seismic loads, the shear in the walls due to seismic are much greater and seismic loads will govern the design. For completeness, bending in the exterior walls due to tornado will be checked at the final design stage, and some additional reinforcing may be required in local areas. Effects of tornado missiles are addressed in the calculation for design of reinforcing steel for the CTB (Reference 47).

The Canister Transfer Building is a large and massive building consisting of exterior reinforced concrete walls 2'-0" thick, a reinforced concrete roof 1'-0" thick, and a solid reinforced concrete mat foundation 5'-0" thick. The interior partitions that make up the low level waste holding area will be constructed of concrete or concrete masonry. The equipment and office areas on the east side of the building will utilize steel framed partition walls covered with gypsum board. The total weight (static load) of the building and foundation is approximately 75,000 kips (Reference 44) or 37,500 tons.

The following provides verification that the site specific and operational criteria of the PFSF are enveloped by the Canister Transfer Building analysis and design.

A. Dead Loads

The Canister Transfer Building will be designed for the self weight of the structure and all permanently attached equipment.

B. Live Load

The Canister Transfer Building will be designed for the following live loads:

- Snow and ice loads - 45 psf per County Building Department exposure C, importance factor = 1.2 (Category IV) per ASCE-7
- Bridge crane and semi-gantry crane loads
- Normal crane handling loads and transfer operations Normal wind load - 90 mph, exposure C, importance factor = 1.15 (Category IV) per ASCE-7
- Vehicle loads (including impact loads)
  - Fully loaded cask transport vehicle
  - H20-44 truck per AASHTO
  - Heavy haul tractor/trailer
  - Rail car and prime mover
- Equipment Loads
  - Storage cask (with loaded canister and transfer cask)
  - Transfer cask (with loaded canister)
  - Shipping cask (with loaded canister and transfer cask)

Two critical load cases were considered. The first is that which produces the worst downward loading on the roof, and includes dead load, live load, and the vertical seismic load acting downward. The vertical seismic load is developed by applying as a static load the enveloped ZPA accelerations from the seismic analysis to the mass of the structure. Included in this load combination is 40% of the enveloped ZPA acceleration in each of the two horizontal (N-S and E-W) directions. This load combination governs the design of the roof, some of the walls, and portions of the base mat. The second load case was selected because it had the greatest overturning potential. It includes dead load, reduced live load, the enveloped E-W ZPA acceleration, 40% of the enveloped vertical ZPA acceleration upward, and 40% of the enveloped ZPA acceleration in the N-S direction. This load combination governs the design of portions of the base mat, crane support beams and some walls. Selected results of the analyses are presented in Figures 10 through 16. The finite element analysis, including the soil model and building model, is described in calculation SC-6 (Reference 46).

Results of the analysis were used to design the reinforcing steel for the concrete walls, slabs, beams and columns (pilasters). In general, the reinforcing required was not excessive. Highly stressed areas are in the roof slab, in the N-S walls where the roof beams intersect the wall, in the crane support beams, in the E-W shear walls, and in the corners of the base mat. The design of the reinforcing steel is described in calculation SC-7 (Reference 47).

Mat Foundation Stability Analyses and Settlement

In addition to the structural analyses described above, stability and settlement analyses of the Canister Transfer Building were also performed to confirm the adequacy of the structure and its foundation. Calculation 05996.02-G(B)-13 (Reference 48) evaluated the stability of the Canister Transfer Building and determined it is stable with respect to bearing capacity, overturning, and sliding due to static and dynamic load conditions.

Calculation 05996.02-G(C)-14 (Reference 49) evaluated the soil settlements due to static load conditions and found the resulting building settlements to be uniform and small and to have little effect on the structure. Calculation 05996.02-G(B)-11 (Reference 84) evaluated the soil settlements due to dynamic load conditions and also found the resulting building settlements to be small and to have little effect on the structure. In summary, the stability and settlement analyses of the Canister Transfer Building indicate that the building is stable and will retain its structural integrity and the performance of the structure will not be adversely affected.

As part of the stability analyses, soil bearing capacity evaluations were performed for the mat founded on soils that were conservatively assigned average undrained shear strengths and effective-stress strength parameters applicable for the soils in the upper ~25 to ~30 ft layer at the site. The strengths of the underlying soil layers are much greater than are applicable for the upper layer; therefore, assuming that they have the same strengths provides a conservative, lower-bound assessment of their stability. See Section 2.6.1.11 for a detailed discussion of the static and dynamic strengths of the soils underlying the site and Section 2.6.1.12.2 for a detailed discussion of the stability analyses of the Canister Transfer Building.

Several load cases were considered in the stability analyses, which consisted of combinations of vertical static, vertical seismic in upward and downward directions, and horizontal seismic in E-W and N-S directions. Loads developed in Calculation SC-5 (Reference 44) were used in these analyses. As in the structural analyses discussed earlier, seismic loads used were based on 100% of the enveloped ZPA acceleration in one direction, combined with 40% of the enveloped ZPA accelerations in each of the other two directions. Minimum factors of safety of 3.0 for the static load case and 1.1 for the seismic load cases are required against a bearing capacity failure of the foundation in soil.

Tables 2.6-9 and 2.6-10 present the results of the bearing capacity analyses of the

4.7.2.2 Plans and Sections

The canister transfer bridge and semi-gantry cranes are shown in Figures 4.7-5 and 4.7-6 respectively.

4.7.2.3 Function

The function of the canister transfer cranes is to assist in the canister transfer operations at the PFSF. A description of the canister transfer operations is contained in Chapter 5.

The overhead bridge crane performs the following activities:

- Remove the impact limiters and personnel barrier from the shipping cask and move them to a laydown area, and
- Upright and remove the shipping cask from the rail car or heavy haul trailer and move the cask into a canister transfer cell.

The overhead bridge crane or the semi-gantry crane performs the following activities:

- Remove the lid from the shipping cask,
- Lift the transfer cask and place on top of the shipping cask, then lift the canister into the transfer cask,
- Lift the transfer cask containing the canister off the shipping cask and onto the top of the storage cask,
- Lower the canister into the storage cask, and
- Remove the transfer cask from on top of the storage cask and place the lid on top of the storage cask.

4.7.2.4 Components

The major components of the overhead bridge crane are the bridge, trolley, main hoist, and auxiliary hoist. The major components of the semi-gantry crane are the gantry frame, trolley, main hoist, and auxiliary hoist.

4.7.2.5 Design Bases and Safety Assurance

The canister transfer cranes are classified as being Important to Safety to provide the safety assurance commensurate with shipping cask and canister lifting activities. The design bases for the canister transfer cranes is described in Chapter 3. Each crane has sufficient capacity to lift the maximum lifted load the crane is designed for during transfer operations. Based on maximum weights presented by Holtec (HI-STORM SAR Tables 3.2.1 and 3.2.2, HI-STAR shipping SAR Table 7.1.1), the maximum lifted loads are addressed in the following Sections.

Since the cranes are classified as Important to Safety, they must be capable of performing their intended functions under all loading conditions including off-normal and accident conditions.

The failure of a crane during canister transfer operations is discussed in SAR Section 8.1.1.3, which shows that the cranes will not drop their loads under off-normal conditions.

The crane operations are designed not to exceed the handling loads (live loads) assumed in the HI-STORM SAR. SAR Section 8.1.4.3 assumes an off-normal handling load is generated from a 2 fps horizontal impact. The crane design parameters limit the high speed of the trolley to less than 60 fpm (1 fps).

SAR Section 8.2.1.2 shows that the cranes maintain their structural integrity and functionality under seismic conditions. However, it is not a design requirement that the crane be operable during an earthquake nor that it be operable after an earthquake. The following is mandatory:

a) The crane bridge (gantry) and trolley are provided with suitable restraints so that they do not leave their rails during an earthquake.
b) No part of the crane shall become detached and fall during an earthquake.
c) The crane load shall not lower in an uncontrolled manner during or as the result of an earthquake.

Additionally, the crane design specification requires that the crane design include the ability to manually release the hoist, emergency, bridge, gantry, and trolley brakes to allow for controlled lowering and positioning of the load in the event of an emergency.

4.7.2.5.1 Maximum Loads Applicable to the Overhead Bridge Crane

The weight of loaded shipping cask, impact limiters, cask support cradle, and personnel barrier is approximately 142 tons (HI-STAR system).

The weight of loaded shipping cask and shipping cask lifting yoke is approximately 121 tons (HI-STAR system).

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The weight of loaded concrete storage cask is approximately 177 tons (HI-STORM system).

The overhead bridge crane capacity is 200 tons, which exceeds the heaviest load of 177 tons.

4.7.2.5.2 Maximum Loads Applicable to Both Overhead Bridge Crane and Semi-Gantry Crane

The weight of transfer cask with a loaded canister and transfer cask lifting yoke is approximately 121 tons (HI-TRAC system).

The semi-gantry crane capacity is 150 tons, which exceeds the heaviest load of 121 tons.

4.7.2.5.3 Seismic Analysis

The PFSF overhead and semi-gantry cranes have been seismically analyzed in accordance with ASME NOG-1 to ensure they will remain in place and support the load during and after a seismic event. The analyses were performed for both cranes by Anatech Corporation to qualify the crane designs for the PFSF deterministic design earthquake (0.67g horizontal, 0.69g vertical – See Section 8.2.1.1). The analysis methods, modeling, and results for the 200 ton bridge crane and 150 ton semi-gantry crane are documented in References 56 and 57 respectively. The maximum stresses were calculated for the major structural components through response spectrum analyses using the amplified response spectra at the crane rail elevation in the Canister Transfer Building. The calculated normal and shear stresses for all components were within allowables as defined in ASME NOG-1. In addition, the cranes were reviewed by Ederer for their seismic stability based on the current PFSF design basis ground motion of 0.53g horizontal and 0.53g vertical (See Section 3.2.10.1.1) and resulting response spectra curves. The response spectra curves for this design basis ground motion are shown in Calculation 05996.02-SC-5 (Reference 44) and include the effects of properties of the soil underlying the Canister Transfer Building. Although the seismic accelerations in the new design basis are lower, the soil properties resulted in increased accelerations at higher elevations in the building and therefore, modifications to both crane designs (Reference 63). However, the crane dimensions still fit within the same envelope shown on Figures 4.7-5 and 4.7-6. PFS will have Ederer formally update the seismic analysis for both cranes as part of the final detailed engineering phase of the crane design and fabrication.

For the 200 ton overhead bridge crane, the vertical peak due to the design basis ground motion increased approximately 14 percent from the deterministic design earthquake. The N-S lateral forces are governed by wheel slip and remain constant. Since the bridge girders, trolley trucks, trolley girder, and equalizing sill are designed at approximately 90 percent or more of the allowable stress (of which this margin should be maintained) the section moduli for these components has increased approximately 14 percent. The E-W horizontal peak increased 100 percent, which affects the bridge trucks. However, since the bridge trucks were designed at a lower allowable stress of 72 percent, the section modulus only required an increase of approximately 10 percent.

For the 150 ton semi-gantry crane, the vertical peak due to the design basis ground motion from the deterministic design earthquake increased approximately 14 percent on the west end of the crane and 8 percent on the east end of the crane. The N-S lateral forces are governed by wheel slip and remained constant. The E-W lateral peak increased 100 percent on the west end and decreased 17 percent on the east end. Based on these changes, the equalizing sill, end tie, and gantry truss, which were designed well below allowable stresses, were unchanged. However, the bridge girder, trolley truck, and trolley girder, which were designed with less margin from the allowable stresses, required an increase of the section moduli by approximately 12 percent. The

4.7.4 (deleted)

4.7.5 Cask Transporter

A cask transporter is used to move the loaded storage cask between the Canister Transfer Building and the storage pad.

4.7.5.1 Design Specifications

The cask transporter is a commercial grade system that has no specific code or specification criteria.

4.7.5.2 Plans and Sections

A drawing of a typical cask transporter is shown on Figure 4.7-4.

4.7.5.3 Function

The function of the cask transporter is to enable transfer of the loaded storage casks between the canister transfer facility and the concrete storage pads.

4.7.5.4 Components

The cask transporter is a large tracked vehicle designed to straddle a storage cask and lift it for transport between the Canister Transfer Building and the storage pads. The transporter lifting mechanism consists of a lift beam supported on either end by two hydraulic lift rams. The lift beam is designed with lift connections to attach to the lifting eyes in the storage cask. The transporter is controlled by a driver who is located on the back corner of the vehicle. The braking system is designed to automatically set when the vehicle operating levers are in neutral or the parking brake is set.

The transporter travels up to 2 mph, has a capacity of 200 tons, and weighs approximately 135,000 lb (Reference 21).

4.7.5.5 Design Bases and Safety Assurance

The cask transporter is classified as not Important to Safety. A failure of any cask transporter components will not result in any safety concerns since the cask would only lower 4 inches back to the ground. Drops this small are within analyzed accident conditions presented in Section 8.2.6. The transporter is designed to mechanically limit the lifting height of a cask to a maximum of 10 inches. The hydraulic lift cylinders are equipped with double locking valves and a cam locking system engages and holds the load in the event a cylinder looses holding power. Indicator lights on the operating console tell if the cams are disengaged or engaged. Markings on the lift boom and a meter on the operating console give indication of the lifted height.

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64. (deleted)

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67. NFPA 70, National Electric Code, 1996.

68. IEEE 692, Criteria for Security Systems for Nuclear Power Generating Stations, 1986.

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TABLE 4.1-1
(Sheet 1 of 7)

PFSF COMPLIANCE WITH GENERAL DESIGN CRITERIA (10 CFR 72, SUBPART F)

10 CFR 72 REQUIREMENT	REQUIREMENT SUMMARY	SAR SECTION WHERE COMPLIANCE IS DEMONSTRATED
72.122 (a) Quality standards	Structures, systems, and components Important to Safety must be designed, fabricated, erected, and tested to quality standards commensurate with the importance to safety of the function.	• Section 3.4 provides the QA classifications for SSCs Important to Safety. • Chapter 4 describes the design of SSCs Important to Safety. • Section 9.2.2 describes the Preoperational Test Plan • Section 9.4.1.1.5 describes the QA procedures req'mts. • Chapter 11 shows that the QA Program is in accordance with 10 CFR 72.140.
72.122 (b) Protection against environmental conditions and natural phenomena	Structures, systems, and components Important to Safety must be designed to accommodate the effects of and be compatible with site characteristics and environmental conditions and to withstand postulated accidents.	• Sections 3.2 and 3.2.10.2.11 provide req'mts for environmental and site design criteria for SSCs Important to Safety. • Sections 4.2 and 4.7 describe the design to mitigate environmental effects. • Chapter 8 and Sections 8.2.1.1, 8.2.1.2, and 8.2.2.2 demonstrate the capability of SSCs Important to Safety to withstand postulated accidents.
72.122 (c) Protection against fires and explosions	Structures, systems, and components Important to Safety must be designed and located so that they can continue to perform their safety functions under credible fire and explosion exposure conditions.	• Section 3.3.6 provides fire and explosion protection req'mts. • Sections 4.2.1.5.1 (I) and (J) and 4.7.3.5.1(E) describe the design that provides fire and explosion protection. • Sections 8.2.4.2 and 8.2.5.2 show the capability of SSCs Important to Safety to withstand postulated fire and explosion accidents.

TABLE 4.1-1
(Sheet 2 of 7)

PFSF COMPLIANCE WITH GENERAL DESIGN CRITERIA (10 CFR 72, SUBPART F)

10 CFR 72 REQUIREMENT	REQUIREMENT SUMMARY	SAR SECTION WHERE COMPLIANCE IS DEMONSTRATED
72.122 (d) Sharing of structures, systems, and components	Structures, systems, and components Important to Safety must not be shared between the PFSF and other facilities unless it is shown that such sharing will not impair the capability of either facility to perform its safety functions.	• Section 1.2 verifies that the PFSF does not share SSCs with other facilities.
72.122 (e) Proximity of sites	An ISFSI located near other nuclear facilities must be designed and operated to ensure that the cumulative effects of their combined operations will not constitute an unreasonable risk to the health and safety of the public.	• Section 7.6.2 verifies that no other nuclear facilities are located within 5 miles of the PFSF.
72.122 (f) Testing and maintenance of systems and components	Systems and components that are Important to Safety must be designed to permit inspection, maintenance, and testing.	• Sections 4.2, 4.7, 5.1.4.7, 4.7.2.1, and 5.1.6.5 describe the capability of SSC's to permit inspection, maintenance, and testing.

TABLE 4.1-1
(Sheet 3 of 7)

PFSF COMPLIANCE WITH GENERAL DESIGN CRITERIA (10 CFR 72, SUBPART F)

10 CFR 72 REQUIREMENT	REQUIREMENT SUMMARY	SAR SECTION WHERE COMPLIANCE IS DEMONSTRATED
72.122 (g) Emergency capability	Structures, systems, and components Important to Safety must be designed for emergencies. The design must provide accessibility to the equipment by onsite and available offsite emergency facilities and services.	• Section 4.1.2 specifies that the PFSF is designed for accessibility. • Section 9.5 summarizes the Emergency Plan for the PFSF.
72.122 (h) Confinement barriers and systems	The spent fuel cladding must be protected during storage against degradation that leads to gross ruptures or the fuel must be otherwise confined. Ventilation systems must be provided to ensure confinement of airborne particulate. Storage confinement systems must have the capability for continuous monitoring to maintain safe storage conditions.	• Section 3.3.2 provides the requirements to ensure confinement of the spent fuel. • Section 4.2.1.5.5 describes the confinement design features. • Section 8.2.7.2 accident analysis shows that there is no loss of confinement. • Section 8.2.10.1 shows that the fuel cladding is protected. • Sections 7.3.4 and 7.3.5 describe the continuous monitoring process.
72.122 (i) Instrumentation and control systems	Instrumentation and control systems must be provided to monitor systems that are Important to Safety over anticipated ranges for normal operation and off-normal operation.	• Section 3.3.3.2 provides the requirements to monitor systems Important to Safety. • Section 5.4.1 describes the instrumentation and control systems.

TABLE 4.1-1
(Sheet 4 of 7)

PFSF COMPLIANCE WITH GENERAL DESIGN CRITERIA (10 CFR 72, SUBPART F)

10 CFR 72 REQUIREMENT	REQUIREMENT SUMMARY	SAR SECTION WHERE COMPLIANCE IS DEMONSTRATED
72.122 (j) Control room or control area	A control room or control area, if appropriate, must be designed to permit occupancy and actions to be taken to monitor the PFSF safely under normal conditions, and to provide safe control of the PFSF under off-normal or accident conditions.	• Section 5.5 shows that a control room/area is not required. • Section 5.1 describes the operational systems that ensure safe conditions during cask storage and canister transfer. • Section 10.2.5 defines the operational controls and limits to be used for the PFSF.
72.122 (k) Utility or other services	Each utility service system must be designed to meet emergency conditions. The design of utility services and distribution systems that are Important to Safety must include redundant systems to maintain the ability to perform safety functions assuming a single failure.	• Section 4.1.2.3 verifies that the PFSF does not rely on utility systems to ensure the safe operation of the facility.
72.122 (l) Retrievability	Storage systems must be designed to allow ready retrieval of spent fuel for further processing or disposal.	• Section 3.3.7 provides the requirements for transferring and storing the canisters that contain the spent fuel and for shipping the canisters offsite. • Section 4.7 describes the capability for retrieving the spent fuel canisters.

TABLE 4.1-1
(Sheet 5 of 7)

PFSF COMPLIANCE WITH GENERAL DESIGN CRITERIA (10 CFR 72, SUBPART F)

10 CFR 72 REQUIREMENT	REQUIREMENT SUMMARY	SAR SECTION WHERE COMPLIANCE IS DEMONSTRATED
72.124 (a) Design for criticality safety	Spent fuel handling, packaging, transfer, and storage systems must be designed to be maintained subcritical.	• Section 3.3.4 provides the requirements to ensure subcriticality is maintained. • Section 4.2.1.5.4 describes the criticality safety design.
72.124 (b) Methods of criticality control	When practicable, the design of an ISFSI must be based on favorable geometry, permanently fixed neutron absorbing materials (poisons), or both.	• Section 3.3.4 provides the requirements for the means of subcriticality control. • Section 4.2.1.5.4 describes the components that maintain subcritical conditions.
72.124 (c) Criticality monitoring	A criticality monitoring system shall be maintained in each area where special nuclear material is handled, used, or stored which will energize clearly audible alarm signals if accidental criticality occurs.	• Section 4.2.1.5.4 describes why criticality monitoring is not applicable for dry storage systems where the spent fuel is packaged in its stored configuration.
72.126 (a) Exposure control	Radiation protection systems must be provided for all areas and operations where onsite personnel may be exposed to radiation or airborne radioactive materials.	• Section 3.3.5 provides the radiological protection design criteria. • Section 4.2.1.5.3 describes the components that provide shielding for exposure control. • Sections 7.1.1 and 7.1.2 describe the program features for ensuring that occupational exposures are ALARA.

TABLE 4.1-1
(Sheet 6 of 7)

PFSF COMPLIANCE WITH GENERAL DESIGN CRITERIA (10 CFR 72, SUBPART F)

10 CFR 72 REQUIREMENT	REQUIREMENT SUMMARY	SAR SECTION WHERE COMPLIANCE IS DEMONSTRATED
72.126 (b) Radiological alarm systems	Radiological alarm systems must be provided in accessible work areas as appropriate to warn operating personnel of radiation and airborne radioactive material concentrations above a given setpoint and of concentrations of radioactive material in effluents above control limits.	• Section 3.3.5 provides the requirements for radiological alarm systems. • Section 7.3.5 describes the radiological monitoring program.
72.126 (c) Effluent and direct radiation monitoring	As appropriate for the handling and storage system, a means to measure effluents must be provided. Areas containing radioactive materials must be provided with systems for measuring the direct radiation levels in and around these areas.	• Section 3.3.5 provides the requirements for effluent and direct radiological systems. • Sections 7.3.5 and 7.6.1 describe the radiological monitoring program.
72.126 (d) Effluent control	The ISFSI must be designed to provide means to limit to ALARA levels the release of radioactive materials in effluents during normal operations and control the release of radioactive materials under accident conditions.	• Section 7.1.2 describes why effuent control is not applicable at the PFSF. • Section 8.2 demonstrates that offsite exposures for postulated accident conditions are controlled such that the dose limits specified in 10 CFR 72.106 are met.

TABLE 4.2-4

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TABLE 4.2-5

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TABLE 4.2-6

(deleted)

TABLE 4.2-7

STATIC PAD ANALYSIS MAXIMUM RESPONSE VALUES

LOADING CONDITIONS		MAXIMUM MOMENT (k-ft/ft)	MAXIMUM SHEAR FORCE (k/ft)	MAXIMUM SOIL PRESSURE (k/ft^2)
Dead Load		0.0	0.0	0.45
Live Load	2 Casks	73.2	7.9	1.81
	4 Casks	66.8	8.7	2.22
	8 Casks	53.0	6.4	2.05
	8 Casks + Transporter	50.3	11.3	3.60

Notes:

1. Values for maximum moment and shear taken from Reference 16 (page 45).
2. Values for maximum soil pressure taken from Reference 16 (page 235 and 236) and include the weight of the storage pad.

Table 4.7-2

HI-TRAC TRANSFER CASK STEADY STATE TEMPERATURE EVALUATION

COMPONENT	TEMPERATURE (°F)	SHORT-TERM TEMPERATURE LIMITS (°F)
Ambient Air	100	N/A
Transfer Cask Outer Shell	223	700
Top Neutron Shield	175	300
Bulk Average Water Jacket	269	307
Transfer Cask Inner Surface	323	600
Canister Shell	459	775
Basket	884	950
Fuel Cladding	902	1058

TABLE 4.7-3

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FIGURE 4.2-4

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FIGURE 4.2-5 (1 of 4)

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FIGURE 4.2-5 (2 of 4)

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FIGURE 4.2-5 (3 of 4)

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FIGURE 4.2-5 (4 of 4)

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FIGURE 4.2-6

(deleted)

FIGURE 4.5-2

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FIGURE 4.7-3

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CHAPTER 5

OPERATION SYSTEMS

5.1 OPERATION DESCRIPTION

This chapter describes the operations that PFSLLC personnel perform using the HI-STORM 100 Cask System (Reference 1). The operation systems provide safe control of the PFSF canister handling and storage systems, which meets the intent of 10 CFR 72.122(j).

Operations related to the storage of spent fuel at the PFSF are performed at the originating nuclear power plant and at the PFSF. Spent fuel operations at the originating power plant are performed in accordance with the originating plant Owner's 10 CFR 50 license. Transport of the spent fuel from the plant to the PFSF is performed in accordance with the requirements of 10 CFR 71 and 49 CFR 171, 172, 173, 174, and 177. The system used to transport fuel to the PFSF is the HI-STAR 100 Shipping Cask System (Reference 3). Storage of the spent fuel at the PFSF is subject to the requirements of the PFSF license that is issued in accordance with 10 CFR 72.

The operations that are performed at the PFSF include the following:

- Receipt and inspection of incoming shipping casks with canisters containing spent fuel
- Transfer of canisters from shipping cask to storage cask
- Placement of the storage cask on storage pads
- Surveillance of storage casks
- Security of the PFSF

- Health Physics at the PFSF
- Maintenance at the PFSF
- Removal of canisters from the PFSF offsite
- Inventory documentation management

The majority of PFSF activity occurs during placement of canisters and casks within the facility and future removal of the canisters offsite. Supporting activities include monitoring storage casks and periodic maintenance of onsite equipment. The PFSLLC will provide detailed procedures for operating, inspecting, and testing the PFSF systems. These procedures will ensure that the spent fuel handling and storage operations are properly accomplished and are in accordance with the PFSF SAR and vendors' SARs.

The following description provides an overview of the operational process for the spent fuel storage facility systems.

5.1.1 Operations at Originating Nuclear Power Plant

The spent fuel operations at the originating nuclear power plant are not part of the PFSF. The description provided in this section is for information only.

Typically, an empty canister is placed inside a transfer cask and both the canister and transfer cask are placed into the spent fuel pool where the canister is loaded with spent fuel. (Some power plants may need to utilize dry transfer methods due to low crane capacity or plant restrictions). The canister exterior is prevented from direct contact with potentially contaminated spent fuel pool water by means of a water or inflatable barrier. Once the fuel is loaded, the canister lid is placed on the canister and the transfer cask is removed from the spent fuel pool. The canister lid and a redundant closure lid are welded to the canister and the canister is drained and vacuum dried.

the cradle, lifted off the transport vehicle, and moved into one of three canister transfer cells. The shipping cask is secured in place by attaching support struts between the cask and the transfer cell walls. The shipping cask lid is unbolted and removed. The canister is then accessible through the top of the shipping cask where the canister lifting attachments and hoist slings are installed on the canister lid. Temporary shielding is positioned as required to maintain worker doses as-low-as-is-reasonably-achievable (ALARA).

The transfer cask is placed onto the shipping cask by the overhead bridge crane or semi-gantry crane. In order to assure cask stability in the event of an earthquake, the crane is not disconnected from the HI-TRAC transfer cask until seismic support struts are attached to the transfer cask, as discussed in Section 4.7.4.5.1. The HI-TRAC transfer cask can remain connected to the crane throughout the canister transfer operation since this transfer cask has a canister downloader that raises and lowers the canister and the crane is not needed to hoist the canister. In this case, it is not necessary to connect the seismic support struts since continuous connection of the HI-TRAC transfer cask to the crane provides assurance that the transfer cask cannot topple in the event of an earthquake. The seismic support struts are attached between the transfer cask and building columns. Shield doors installed on the bottom of the transfer cask are opened. The hoist slings are pulled up through the transfer cask and the canister is lifted up into the transfer cask just above the shield doors. The shield doors are closed and the canister is lowered onto the doors, which support the weight of the canister. The support struts are disconnect from the transfer cask. The transfer cask is lifted from the shipping cask by the crane and placed on top of the storage cask. Support struts are again attached between the transfer cask and transfer cell walls. The canister is lifted slightly to remove its weight from the transfer cask shield doors.

The shield doors are opened and the canister is lowered into the storage cask. The transfer cask is removed from the top of the storage cask and the storage cask lid is installed. Temporary shielding is removed from the cask transfer area. During the transfer process, radiation levels are measured to assure doses to workers are ALARA.

5.1.4.3 Placement of the Storage Cask on the Storage Pad

The storage cask loading is now complete and ready for transport to a storage pad. The storage cask is moved out of the canister transfer cell by the cask transporter. The cask transporter lifts the storage cask approximately 4 inches high. The cask is then moved to the appropriate storage pad by the cask transporter. At the storage pad, the storage cask is positioned and lowered onto the storage pad. The temperature at the air outlet vents is taken after the cask is placed on the pad in accordance with Technical Specification requirements to confirm proper operation of the storage system.

5.1.4.4 Surveillance of the Storage Casks

While in storage, the proper operation of the storage casks is verified by surveillance procedures. Cask temperatures are measured by a continuous monitoring system to verify temperatures do not exceed temperature limits in the Technical Specifications. In addition, the cask air vents are inspected for blockage on a quarterly basis. An overall site observation surveillance is also performed on a periodic basis to detect any cask damage or accumulation of site debris.

Dose rates associated with individual storage casks are measured to verify that dose rates are within Technical Specification limits to ensure adequate shielding of the canister so that radiation exposure to the general public is minimized and occupational doses to personnel working in the vicinity of the storage casks are maintained as low as is reasonably achievable. Radiation doses emitted from the storage casks are measured by thermoluminescent dosimeters (TLDs) located at the restricted area (RA) and owner controlled area (OCA) fences to ensure doses are within 10 CFR 20.1301 and 10 CFR 72.104 or 40 CFR 191 limits.

5.1.4.5 Security Operations

Security personnel coordinate several security related functions that include performing continual surveillance for intruders, evaluating intrusion alarms, processing visitors to the PFSF, searching packages and vehicles, issuing badges to workers, coordinating with local law enforcement agencies, and contacting appropriate emergency response personnel. The security personnel are also responsible for identifying and assessing off-normal and emergency events during off-shift hours of PFSF operation. Details for the security personnel are discussed in the PFSF Security Plan (Reference 5).

5.1.4.6 Health Physics Operations

The health physics (HP) personnel are responsible for taking radiation dose and contamination surveys on incoming spent fuel shipments. In order to maintain the PFSF philosophy of "Start Clean/Stay Clean", HP personnel ensure that contamination levels on the canisters of incoming shipments are within the Technical Specification requirements. Canisters exceeding the limits will be returned to the originating power plant for decontamination.

During the transfer process, HP personnel monitor doses to ensure that workers are not exposed to unnecessary radiation. In the event high doses are detected, temporary shielding, in the form of lead blankets, neutron shielding, portable shield walls, etc., are used to maintain doses ALARA. HP personnel perform dose rate surveillances of the loaded storage cask to ensure Technical Specification limits are met.

In addition to surveillance activities, HP personnel monitor onsite and offsite radiation levels to ensure worker and offsite doses are in accordance with regulatory requirements. HP personnel also calibrate radiation protection instrumentation.

5.1.4.7 Maintenance Operations

Because of their passive nature, the storage casks require little maintenance over the lifetime of the PFSF. Typical maintenance tasks may involve occasional replacement and recalibration of temperature monitoring instrumentation.

Periodic maintenance is required on the overhead bridge crane, semi-gantry crane, transfer equipment, and shipping casks. Maintenance of these SSCs, which are classified as important to safety, ensure that they are safe and reliable throughout the life of the PFSF per 10 CFR 72.122(f).

Maintenance is also required on the following components not important to safety: the heavy haul tractor/trailer (if used), rail car and locomotive (if used), cask transporter, security systems, temperature and radiation monitoring systems, diesel generator, electrical systems, fire protection systems, and site infrastructure. The Operations and Maintenance (O&M) Building is provided to facilitate maintenance activities.

5.1.4.8 Transfer of Canisters from PFSF Offsite

A 10 CFR 71 licensed shipping cask will transport in the future the canisters offsite to another facility. Transfer operations will utilize the Canister Transfer Building to transfer the canisters from the storage casks to the shipping casks. Once loaded in a shipping cask, the spent fuel canister is shipped to the designated facility.

5.1.5 Flow Sheets

A flow diagram and illustration showing the sequence of operations for canister receipt, transfer, and placement into storage is shown on Figures 5.1-1 and 5.1-2 for the HI-

STORM storage system.

A flow diagram showing the sequence of operations required to remove the canisters from the PFSF and ship them offsite is shown on Figure 5.1-5.

The number of personnel and the time required for the various operations are given in Table 5.1-1 for the HI-STORM system. This table is used to develop the occupational exposures in Chapter 7.

5.1.6 Identification of Subjects for Safety Analysis

5.1.6.1 Criticality Prevention

As discussed in Section 4.2.1.5.4, criticality is controlled at the PFSF by utilizing fuel assembly geometry. Poison materials are primarily for underwater canister loading in the originating nuclear plant spent fuel pool. During storage, with the canister dry and sealed from the environment, no further criticality control measures within the storage installation are necessary.

5.1.6.2 Chemical Safety

There are no chemical hazards associated with the operation of the PFSF.

5.1.6.3 Operation Shutdown Modes

During storage, there are no operational shutdown modes associated with the HI-STORM Storage System since the system is passive and relies on natural air circulation for cooling. During canister transfer, the transfer process may be shut down at the end of the day until the next day because of the transfer duration. Operation procedures ensure that no shutdown can occur in the middle of an operational step. Operational shutdown steps following emergency or accident events are also addressed by the PFSF operational procedures. All operational shutdown modes at the PFSF are safe shutdown modes due to the design features of the facility.

5.1.6.4 Instrumentation

Due to the totally passive nature of the storage casks, there is no need for any instrumentation to perform safety functions. Temperature monitors are utilized as a means to monitor the cask temperature during storage. Area radiation monitors are used to measure radiation levels in the Canister Transfer Building during canister transfer operations and in the LLW storage room. Continuous air monitors, located in the exhaust of each canister transfer cell, monitor radioactivity concentrations in the air leaving each canister transfer cell. Portable radiation monitors are used to measure radiation levels of casks following canister transfer. PFSF personnel are equipped with personnel dosimeters whenever they are in the RA. The radiation dose will be monitored at the perimeters of the RA and OCA. The temperature and radiation monitors are classified as Not Important to Safety.

5.1.6.5 Maintenance Techniques

No special maintenance techniques are necessary that would require a safety analysis.

There is preventative maintenance performed on a regular basis on the overhead transfer crane, canister lifting equipment, cask transporter, heavy haul tractor/trailers, radiation detection and monitoring equipment, cask temperature monitoring equipment, security equipment, fire detection and suppression equipment, etc. Maintenance is performed in accordance with 10 CFR 72.122(f) and manufacturer's requirements.

5.2 SPENT FUEL CANISTER HANDLING SYSTEMS

5.2.1 Spent Fuel Canister Receipt, Handling, and Transfer

An operational description for the systems used for the receipt and transfer of spent fuel canisters is provided in the following paragraphs. Special features of these systems to ensure safe handling of the spent fuel canisters are also described.

5.2.1.1 Spent Fuel Canister Receipt

5.2.1.1.1 Functional Description

The shipping casks and impact limiters comprise the system in which the spent nuclear fuel canisters are contained when they arrive at the PFSF. The shipping cask system protects the enclosed spent fuel canister from physical damage, provides shielding, and allows sufficient cooling of the canister while enroute to the PFSF.

5.2.1.1.2 Safety Features

Safety features of the system include the impact limiters, which help protect the spent fuel shipping cask during transportation, and the design, materials, and construction of the shipping casks, which provide gamma and neutron shielding, conductive and radiant cooling, criticality control, and structural strength to protect the spent fuel canister. A tamperproof device on the cask provides indication of an unauthorized attempt to obtain access to the cask. These safety features are fully described in the HI-STAR shipping cask SAR.

5.2.1.2 Spent Fuel Canister Handling

5.2.1.2.1 Functional Description

The overhead bridge and semi-gantry cranes perform handling functions inside the Canister Transfer Building for the shipping cask and the transfer cask. The canister downloader, bolted on top of the HI-TRAC transfer cask is used to raise and lower the HI-STORM canister.

Shipping and transfer cask handling components include the shipping cask and transfer cask lifting yokes, trunnions, and seismic support struts.

The storage cask handling component consists of the storage cask lifting attachments, cask transporter, and the overhead bridge crane, if needed.

The canister handling components consist of the lifting slings and the HI-STORM canister lifting cleats.

5.2.1.2.2 Safety Features

Safety features of the overhead bridge and semi-gantry cranes include single-failure-proof designs for sustaining the load upon failure of any single component, limit switches for prevention of hook travel beyond safe operating positions, and provisions for lowering a load in the event of an overload trip. The cranes are classified as ASME NOG-1 Type I cranes. A Type I crane is defined as a crane that is designed and constructed to remain in place and support a critical load during and after a seismic event and has single-failure-proof features such that any credible failure of a single component will not result in the loss of capability to stop and/or hold the critical load. Design requirements for the cranes require testing, inspection, and maintenance activities on the cranes in accordance with 10 CFR 72.122(f) which, are performed in accordance with the QA Program described in SAR Chapter 11 to ensure that the design requirements are satisfied. Strict adherence to the design, testing, inspection, and maintenance criteria as noted above ensure adequate safety margins are provided to prevent damage to the shipping cask, canister, or storage cask during normal, off-normal, and accident conditions. The crane designs include limit switches for prevention of bridge, trolley, and hook travel beyond safe operating positions, limits on bridge, trolley, and hook travel speeds, and provisions for lowering a load in the event of an overload trip. Periodic inspection and testing will be performed to keep the cranes certified to ASME NOG-1.

Safety features of the HI-TRAC downloader, used to raise and lower the HI-STORM canister during canister transfer operations, include a single-failure-proof design for sustaining the load upon failure of any single component and/or loss of hydraulic pressure as described in the HI-STORM SAR.

Safety features of the shipping and transfer cask handling components include single-failure-proof lift capacity or equivalent safety factor as described in the HI-STAR Shipping Cask and HI-STORM Storage Cask SARs.

Use of seismic support struts ensure the shipping and transfer cask do not topple over during an earthquake. Safety features of the seismic support struts include using standard rigid support assemblies that conform to ASME III, NF requirements for Class 2 nuclear grade supports. As such, the struts are subject to QA requirements per 10 CFR 50, Appendix B; material certification, design, and NDE per ASME III NF; and welder and weld qualifications per ASME IX. Each cask utilizes 2 struts, which provide restraint in both orthogonal horizontal directions.

There are no safety features associated with the cask transporter since the storage cask is designed to withstand drops that could result from a failure associated with the transporter lift components. The transporter is designed such that the lift mechanism can only lift the storage cask within lift heights specified by the Technical Specifications. The hydraulic lift cylinders are equipped with double locking valves and a cam locking system engages and holds the load in the event a cylinder looses holding power. Indicator lights on the operating console inform the operator if the cams are disengaged or engaged. Markings on the lift boom and a meter on the operating console give indication of the lifted height.

The safety features of the canister handling components, slings and canister lifting cleats, are their redundancy and the required stress safety margins as described in the HI-STORM SAR.

5.2.1.3 Spent Fuel Canister Transfer

5.2.1.3.1 Functional Description

The transfer cask is used for transfer of the spent fuel canister between the shipping cask and the storage cask. The transfer cask protects the spent fuel canister from physical damage and provides radiation shielding.

5.2.1.3.2 Safety Features

The transfer casks provide radiation shielding and act as special lifting devices when carrying a canister loaded with spent fuel. The transfer cask lifting trunnions are designed and tested to the single-failure-proof requirements of NUREG-0612 (Reference 6) and ANSI N14.6 (Reference 7) so that canisters can be lifted by the transfer cask without the requirement to analyze a transfer cask drop. However, annual testing requirements per ANSI N14.6 of the transfer cask trunnion welds is not performed since the welds cannot be accessed for testing and NDE.

The transfer casks consist of cylindrical steel liners with a lead gamma shield and a neutron shield. Two trunnions are provided for transfer cask handling. The transfer cask has movable shield doors at the bottom to allow raising the canister into the transfer cask, lowering of the canister into the storage or shipping cask, or to support the canister weight and provide shielding while in the transfer cask. The doors slide in steel guides along each side of the transfer cask. Steel pins or bolts are used to prevent inadvertent opening of the doors. Roller bearings on the HI-TRAC transfer cask enable the cask doors to be manually operated.

The transfer casks are designed to prevent the canister from being lifted beyond the top of the cask, which would expose the canister and cause high radiation doses. On the HI-TRAC transfer cask, the canister downloader, which raises the canister, is bolted on top of the cask. The canister can only be lifted up to the downloader hoist mechanical stops and is prevented from being raised beyond the top of the HI-TRAC cask.

The lifting yokes provided with the transfer casks are used to interface with the crane.

The safety features of the HI-TRAC transfer cask are described in greater detail in the HI-STORM SAR.

5.2.2 Spent Fuel Canister Storage

Spent fuel storage consists of the HI-STORM storage system, which includes spent fuel canisters placed in the steel and concrete storage casks located on the storage pads. The storage system is a passive design and requires no support systems for operation. The storage system performs its functions under normal conditions as discussed in Chapter 4 and off-normal and accident level conditions as discussed in Chapter 8. Limits of operation associated with various normal and off-normal conditions are contained in the PFSF Technical Specifications. Surveillance requirements are also contained in the PFSF Technical Specifications.

5.2.2.1 Safety Features

Safety features include a passive dry cask design and administrative controls. The canister is enclosed in the cavity of the storage cask, which protects the canister from severe natural phenomena (such as tornado-driven missiles), provides required shielding of the canister, and flow paths for natural convection cooling. The results of analyses of hypothetical storage cask tipover events are described in Section 8.2.6, where it is concluded that the canister will remain intact inside the storage cask and canister internals will not be damaged. Safety features are discussed in greater detail in Chapter 4, Chapter 8, and the HI-STORM SAR.

5.7 REFERENCES

1. Final Safety Analysis Report for the Holtec International Storage and Transfer Operation Reinforced Module Cask System (HI-STORM 100 Cask System), Holtec Report HI-2002444, NRC Docket No. 72-1014, Revision 0, July 2000.

2. (deleted)

3. Topical Safety Analysis Report for the Holtec International Storage, Transport, and Repository Cask System (HI-STAR 100 Cask System), Holtec Report HI-951251, Docket 71-9261, Revision 9, April 2000.

4. (deleted)

5. PFSF Security Plan, Revision 0.

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

7. ANSI N14.6, Radioactive Materials - Special Lifting Devices for Shipping Containers, 1993.

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TABLE 5.1-2
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CHAPTER 6

SITE-GENERATED WASTE CONFINEMENT AND MANAGEMENT

6.1 ONSITE WASTE SOURCES

This chapter addresses the requirements for confining and managing any site-generated radioactive wastes in accordance with 10 CFR 72.128(b).

The Private Fuel Storage Facility (PFSF) is designed to a "Start Clean/Stay Clean" philosophy. The spent fuel storage canisters are sealed by welding at the originating nuclear power plants to preclude any leakage of radionuclides.

All spent fuel stored at the PFSF is contained in sealed canisters. Under all normal, off-normal, and credible accident conditions of transport, handling, and storage, the potential does not exist for breach of the canister and release of radioactive material associated with spent fuel from inside the canister.

The potential for radionuclide contamination of the outside surface of the canisters is minimized by using design concepts that preclude intrusion of spent fuel pool water into the annular gap between the transfer cask and the canister while they are submerged in the pool water at the originating nuclear power plants, as described in Chapter 7 of the HI-STAR shipping cask Safety Analysis Report (SAR) (Reference 1) and Chapter 8 of the HI-STORM storage cask SAR (Reference 3). Health physics surveys required to be performed at the originating nuclear power plants, following removal of loaded canisters from the spent fuel pools, include a smear survey to assess removable contamination levels on accessible surfaces of the canister (canister lid and approximately 3 to 6 inches on canister sides down from the lid). In the event removable contamination levels measured on accessible canister surfaces exceed the criteria specified in the canister vendor's Certificate of Compliance, the canister will not be released for shipment to the PFSF. Canisters with levels of removable contamination above the specified limit must be decontaminated prior to release for transport to the PFSF. The shipping cask externals are also surveyed and decontaminated, as necessary, before the cask leaves the originating nuclear power plants. Radioactive wastes generated during the canister and shipping cask loading operations are processed at the originating nuclear power plants.

After a shipping cask arrives at the PFSF, the shipping manifest is checked and contamination surveys of the outer surfaces of the loaded shipping cask are performed in accordance with the manifest and U.S. Department of Transportation (DOT) regulations (49 CFR 173.443 - Reference 5). Surveys are also performed on the storage cask after the canister is transferred from the shipping cask to the storage cask. Should an off-normal event occur that results in a storage cask becoming contaminated, removal of the contaminants would be conducted using decontamination methods that only result in the generation of dry active wastes. The small amount of dry active waste that may be generated would consist of anti-contamination garments, rags, and associated health physics material. This solid waste would be packaged and temporarily stored in the low-level waste (LLW) holding cell of the Canister Transfer Building until the waste is shipped offsite to a low-level radioactive waste disposal facility.

6.4 SOLID WASTES

All spent fuel stored at the PFSF is contained in sealed canisters. Under all normal, off-normal, and credible accident conditions of transport, handling, and storage, the potential does not exist for breach of the canister and release of radioactive material associated with spent fuel from inside the canister.

There is a potential for the presence of some contamination on the external surfaces of canisters as a result of submergence in spent fuel pools during spent fuel loading operations at the originating nuclear power plants, even though measures are taken to prevent contamination (see Chapter 7 of the HI-STAR shipping SAR). Following fuel loading operations at the originating nuclear power plants, a smear survey is performed to determine removable contamination levels on accessible outer canister surfaces near the top of the canister (canister lid and approximately 3 to 6 inches on canister sides down from the lid). In the event canister removable contamination levels (measured on canister top surfaces) exceed the criteria specified in the canister vendor's Certificate of Compliance, the canister will not be released for shipment to the PFSF. Canisters with levels of removable contamination above the specified limit must be decontaminated prior to release for transport to the PFSF.

Once the shipping cask arrives at the PFSF and its closure is removed, a smear survey of accessible portions of the canister is again performed. If removable surface contamination levels exceed the limits specified in the PFSF Technical Specifications, the canister is returned to the originating nuclear power plant for decontamination.

Even with these measures to assure canister external surfaces are relatively free of removable contamination, contamination surveys are performed on outer surfaces of storage casks, following loading of canisters into the storage casks in the Canister Transfer Building. Under off-normal conditions, such as a canister mishandling event, it is considered possible for removable contamination to be released from the external surfaces of a canister, possibly depositing contamination upon surfaces of the shipping, transfer, or storage casks. Any necessary decontamination of these casks will be performed using dry methods. If such decontamination is necessary, a small quantity of solid LLW may be generated, consisting of smears, disposable clothing, tape, blotter paper, rags, and related health physics material. This material will be collected, identified, packaged in suitable LLW containers (such as standard 55-gallon steel drums that comply with transportation and disposal requirements), marked in accordance with 10 CFR 20 requirements, and temporarily stored in the LLW holding cell of the Canister Transfer Building while awaiting removal to a LLW disposal facility. The LLW holding cell is regularly surveyed and inventoried, including inspection of the materials stored, to evaluate the status of materials and controls (e.g., physical condition of containers, access control, posting).

The volume of solid waste is expected to be minimal since the occurrence of contamination would be due to an off-normal event.

Any wastes that are generated are controlled, stored, and disposed in compliance with the requirements of 10 CFR 20. All solid wastes are packaged for removal to a LLW disposal facility. Packaging complies with requirements specified by 49 CFR 171-177, 10 CFR 71, and the disposal facility criteria, as applicable.

State-of-the art solid radwaste handling equipment and procedures will be used in handling any solid waste generated at the PFSF. The following is an example of the process.

6.6 REFERENCES

1. Topical Safety Analysis Report for the Holtec International Storage, Transport, and Repository Cask System, (HI-STAR 100 Cask System), Holtec Report HI-951251, Docket 71-9261, Revision 9, April 2000.

2. (deleted)

3. Final Safety Analysis Report for the Holtec International Storage and Transfer Operation Reinforced Module Cask System (HI-STORM 100 Cask System), Holtec Report HI-2002444, NRC Docket No.72-1014, Revision 0, July 2000.

4. (deleted)

5. 49 CFR 173, Shippers - General Requirements for Shipments and Packagings.

6. NRC Regulatory Guide 8.10, Operating Philosophy for Maintaining Occupational Radiation Exposures As Low As Is Reasonably Achievable, Rev. 1R, September 1975.

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CHAPTER 7

RADIATION PROTECTION

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CHAPTER 7

RADIATION PROTECTION

7.1 ENSURING THAT OCCUPATIONAL RADIATION EXPOSURES ARE AS LOW AS IS REASONABLY ACHIEVABLE (ALARA)

The objective for the Private Fuel Storage Facility (PFSF) Radiation Protection Program is to keep radiation exposures to facility workers and the general public as low as is reasonably achievable (ALARA). Section 7.1.1 describes the policy and procedures that ensure that ALARA occupational exposures are achieved. Section 7.1.2 describes the ALARA design considerations and Section 7.1.3, the ALARA operational considerations.

7.1.1 Policy Considerations

A Radiation Protection Program will be implemented at the PFSF in accordance with requirements of 10 CFR 72.126, 10 CFR 20.1101, and 10 CFR 19.12 (References 1, 2, and 3). The program will draw upon the experience and expertise of programs and personnel of the Private Fuel Storage L.L.C. (PFSLLC) member utilities.

The primary goal of the Radiation Protection Program is to minimize exposure to radiation such that the individual and collective exposure to personnel in all phases of operation and maintenance are kept ALARA. This is accomplished by integrating ALARA concepts into design, construction, and operation of the facility.

Trained personnel will develop and conduct the Radiation Protection Program and will assure that procedures are followed to meet PFSLLC and regulatory requirements.

Training programs in the basics of radiation protection and exposure control will be provided to all facility personnel whose duties require working in radiation areas.

Basic objectives of the ALARA program are:

1. Protection of personnel, including surveillance and control over internal and external radiation exposure to maintain individual exposures within permissible limits and ALARA, and to keep the annual integrated (collective) dose to facility personnel ALARA.

2. Protection of the public, including surveillance and control over all conditions and operations that may affect the health and safety of the public.

The radiation protection staff is responsible for and has the appropriate authority to maintain occupational exposures as far below the specified limits as reasonably achievable. Ongoing reviews will be performed to determine how exposures might be reduced. The program will ensure that PFSF personnel receive sufficient training and that radiation protection personnel have sufficient authority to enforce safe facility operation. Periodic training and exercises will be conducted for management, radiation workers, and other site employees in radiation protection principles and procedures, protective measures, and emergency responses. Revisions to operating and maintenance procedures and modifications to PFSF equipment and facilities will be made when the proposed revisions will substantially reduce exposures at a reasonable cost. The program will also ensure that adequate equipment and supplies for radiation protection work are provided.

The PFSLLC is committed to a strong ALARA program. The ALARA program follows the guidelines of Regulatory Guides 8.8 (Reference 4) and 8.10 (Reference 5) and the requirements of 10 CFR 20 (Reference 2). Management is committed to compliance with regulatory requirements regarding control of personnel exposures and will establish and maintain a comprehensive program at the PFSF to keep individual and collective doses ALARA. Management will assure that each staff member integrates appropriate radiation protection controls into work activities. PFSF personnel will be trained and updated on ALARA practices and dose reduction techniques to assure that each individual understands and follows procedures to maintain his/her radiation dose ALARA. Design, operation, and maintenance activities will be reviewed to ensure ALARA criteria are met.

The ALARA program will ensure that:

1. An effective ALARA program is administered at the PFSF that appropriately integrates management philosophy and NRC regulatory requirements and guidance.

2. PFSF design features, operating procedures, and maintenance practices are in accordance with ALARA program guidelines. Formal periodic reviews of the Radiation Protection Program will assure that objectives of the ALARA program are attained.

3. Pertinent information concerning radiation exposure of personnel is reflected in design and operation.

4. Appropriate experience gained during the operation of nuclear power stations relative to radiation control is factored into procedures, and revisions of procedures, to assure that the procedures continually meet the objectives of the ALARA program.

5. Necessary assistance is provided to ensure that operations, maintenance, and decommissioning activities are planned and accomplished in accordance with ALARA objectives.

6. Trends in PFSF personnel and job exposures are reviewed to permit corrective actions to be taken with respect to adverse trends.

PFSF personnel will be responsible for ensuring that activities are planned and accomplished in accordance with the objectives of the ALARA program. Staff will ensure that procedures and their revisions are implemented in accordance with the objectives of the ALARA program, and that radiation protection staff is consulted as necessary for assistance in meeting ALARA program objectives. Individual radiation doses, and collective doses associated with tasks controlled by radiation work permits, will be tracked to identify trends and support development of alternative procedures that result in lower doses.

7.1.2 Design Considerations

ALARA considerations have been incorporated into the PFSF design, in accordance with 10 CFR 72.126(a), based upon the layout of the PFSF area and the type of spent fuel storage system selected. The following summarizes the design considerations:

- The peripheral storage pads are located 150 ft (45.7 meters) from the Restricted Area (RA) fence and 2,119 ft (646 meters) from the owner controlled area (OCA) boundary at their closest locations. This provides an acceptable distance from radiation sources to offsite personnel to ensure dose rates at the OCA boundary are minimized and maintained within specified limits.

- The storage pads have been sized to allow adequate spacing between storage casks to permit workers to function efficiently during placement/removal of storage casks at the pads and during performance of maintenance (e.g. clearing blockage from the inlet ducts) and surveillances. Adequate work space helps to minimize time spent by workers in the vicinity of storage casks, limiting worker dose.

- The storage system design is based on a metal canister that is sealed by welding for spent fuel confinement, preventing release of radionuclides from inside the canister. Radioactive effluents are thus precluded by design. This meets the intent of 10 CFR 72.126(d), which requires that the ISFSI design provide means to limit the release of radioactive materials in effluents during normal operations to levels that are ALARA. There are no radioactive effluents released from the PFSF during normal operations. This passive system design also requires minimum maintenance and surveillance requirements by personnel.

- The data acquisition of the storage cask temperature monitoring system enables remote readout of temperatures representative of cask thermal performance, avoiding time spent by PFSF staff to perform daily walkdowns, or take measurements, or read instrumentation in the vicinity of the storage casks.

- Holtec International (Holtec), the vendor for the spent fuel storage system, has incorporated a number of design features to provide ALARA conditions during transportation, handling, and storage as described in its HI-STORM Safety Analysis Report (Reference 6).

- Where practical, power operated wrenches will be used to reduce the times associated with tasks involving bolt insertion and removal during shipping cask receipt and canister transfer operations. This will minimize time spent in radiation fields. Temporary shielding will be used where it is determined to be effective in reducing total dose for a task (considering doses to personnel involved in its installation and removal).

- Solid low-level waste (LLW) generated during operations is packaged and stored in LLW containers in a LLW holding cell within the Canister Transfer Building.

Regulatory Position 2 of Regulatory Guide 8.8, is incorporated into design considerations, as described below:

- Regulatory Position 2a on access control is met by use of a fence with a locked gate that surrounds the PFSF RA and prevents unauthorized access.

- Regulatory Position 2b on radiation shielding is met by the heavy shielding of the shipping, storage, and transfer casks which minimizes personnel exposures during shipping cask reception, canister transfer, canister storage, and offsite shipment operations. The designs of the storage cask air inlet and outlet ducts prevent direct radiation streaming. Each of the transfer cells within the Canister Transfer Building has thick concrete walls, with a substantial steel roller door, designed to shield personnel in adjacent transfer cells and in the main bay of the Canister Transfer Building from relatively high dose rates that could be associated with the canister transfer operations. The designs of the shipping, storage, and transfer casks assure adequate shielding for personnel inside the transfer cells to accomplish the transfer operation with dose rates ALARA.

The Canister Transfer Building itself is located approximately 425 ft (130 meters) from the nearest storage pad. It has reinforced concrete walls and roof designed to withstand tornado-driven missiles. The walls and roof provide substantial shielding of gamma and neutron radiation emitted from the sides (direct) and tops (scattered) of storage casks in the cask storage area. The Security and Health Physics Building is located approximately 948 ft (289 meters) from the nearest storage pad, and approximately 396 ft (121 meters) from the Canister Transfer Building. Shielding provided by the walls and roof of this structure reduce dose rates from the cask storage area to security and radiation protection personnel who will spend a large fraction of their working hours in this building. The Administration Building is located approximately 2,580 ft (786 m) and the Operations and Maintenance Building is located approximately 1,960 ft (597 m) from the nearest storage pad. Dose rates are sufficiently low at these distances such that the buildings do not require shielding to assure dose rates are ALARA to personnel in the buildings.

- Regulatory Position 2c on process instrumentation is met since the cask temperature monitoring system will utilize a data acquisition system to record cask temperature instrumentation readings, avoiding time spent by PFSF staff to make daily cask vent blockage surveillances and to read instrumentation in the vicinity of the storage casks.

- Regulatory Position 2d on control of airborne contaminants is not applicable because gaseous releases are precluded by the sealed canister design. No surface contamination is expected on the outer surfaces of the canister since process controls are maintained during fuel loading into the canister at the originating nuclear power plants. Assuming the outer surfaces of a canister have removable Co-60 contamination at the maximum levels permitted by the PFSF Technical Specifications, and all of this is postulated to be released into the Canister Transfer Building atmosphere, general area radionuclide concentrations in the Canister Transfer Building would not exceed 10 CFR 20 Appendix B, Table 1, allowable airborne concentrations for occupational workers.

- Regulatory Position 2e on crud control is not applicable to the PFSF because there are no systems at the PFSF that could produce crud.

- Regulatory Position 2f on decontamination is met because the internal surfaces of shipping, transfer, and storage casks have hard surfaces that lend themselves to decontamination by wiping. Surfaces of the transfer cells' walls and floors are painted with a special paint that is easily decontaminated.

- Regulatory Position 2g on radiation monitoring is met with the use of area radiation monitors in the Canister Transfer Building for monitoring general area dose rates from the casks and canisters during canister transfer operations, and with thermoluminescent dosimeters (TLDs) along the perimeters of the RA and OCA to provide information on radiation doses. Continuous air monitors will be located in the exhaust of each canister transfer cell.

- Regulatory Position 2h on resin treatment systems is not applicable to the PFSF because there will not be any radioactive systems containing resins.

- Applicable portions of Regulatory Position 2i concerning other miscellaneous ALARA items is met because PFSF features provide a favorable working environment and promote efficiency (paragraph 2i(13)). These include:

adequate lighting in the Canister Transfer Building, including in the canister transfer cells, and on the storage pads; adequate ventilation in the Canister Transfer Building; adequate working space in the Canister Transfer Building and at the storage pads; and accessibility - with platforms or scaffolding and ladders that facilitate ready access to the tops of the shipping casks and storage casks and to the transfer cask doors where operators need to perform tasks during canister transfer operations. Regulatory Position 2i(15) is met because the emergency lighting system is adequate to permit prompt egress from any high radiation areas that could possibly exist in the vicinity of the canister/casks during canister transfer operations.

7.1.3 Operational Considerations

Specific PFSF operational considerations to achieve ALARA conditions are as follows:

- Fuel loading operations take place at the originating nuclear power plants, away from the PFSF. There are no fuel assembly handling operations at the PFSF.

- No surface contamination is expected on the canisters as the result of controls applied during the fuel loading operations at the originating nuclear power plants. Workers will therefore not be exposed to surface contamination or airborne contamination during canister transfer operations.

- Canister transfer between the shipping cask and the storage cask will take place within a shielded transfer cask.

- Prior to canister transfer operations, "dry runs" will be performed to train personnel on canister transfer procedures, discuss methods to minimize exposures, and refine procedures to achieve minimum probable exposures.

- The PFSF procedures and work practices will reflect ALARA lessons learned from other ISFSIs that use dry cask storage, as applicable.

- Operations research will be performed to determine types of tools, portable shielding, and equipment that will help to minimize exposures to workers involved in canister transfer operations.

- The overhead bridge crane and the semi-gantry crane, located in the Canister Transfer Building, are both single-failure-proof and are designed to withstand the design basis ground motion, as described in Chapter 4. The overhead bridge crane, whose range of travel covers the length and width of the Canister Transfer Building, including the transfer cell area, handles the shipping casks and moves the shipping casks into and out of the transfer cells. It can also be used to lift a transfer cask, a canister, or a storage cask, as necessary. The semi-gantry crane is designed to serve the transfer cells, and is used to lift the transfer casks and canisters during the canister transfer operations. Operation of these cranes during canister transfer operations is discussed in Chapter 5. The HI-STORM canister is handled by the canister downloader, which is also a single-failure-proof lifting device. The cranes and lifting devices used during the canister transfer operation comply with single-failure criteria to avoid a cask or canister drop.

- A self-propelled cask transporter is used to move storage casks from the Canister Transfer Building to the storage pads. The cask transporter requires minimum personnel and allows for quick and accurate placement of a storage cask.

- The storage casks are spaced on the storage pads with sufficient tolerance to facilitate ease of placement operations and minimize the time spent by operators near adjacent casks.

- The storage systems do not require any systems that process liquids or gases or contain, collect, store, or transport radioactive liquids. Therefore, there are no such systems to be maintained or operated. Any solid radioactive waste collected during canister transfer operations will be temporarily staged in steel drums in a holding cell in the Canister Transfer Building while awaiting shipping offsite, as described in Section 6.4.

- As discussed in Section 7.5.2, any water collected in the Canister Transfer Building shipping cask load/unload bay drain sumps is sampled and analyzed to verify it is not contaminated prior to its release. If contaminated water is detected, it will be collected in a suitable container, solidified by the addition of a suitable compound, staged in a holding cell while awaiting shipment offsite, and shipped offsite as solid waste in accordance with Radiation Protection procedures.

Regulatory position 4 of Regulatory Guide 8.8 is met with the use of area radiation monitors in the Canister Transfer Building and TLDs around the RA and OCA boundaries. In addition, radiation protection personnel will use portable monitors during shipping cask receipt, inspection, and canister transfer operations, and the operating staff will have personal dosimetry (Section 7.5.2). Continuous air monitors will be located in the exhaust of each canister transfer cell (Section 7.3.5). The access control point will be at the Security and Health Physics Building, as described in Section 7.5.2.

Protective equipment, including anti-contamination clothing and respirators, will be available in the Security and Health Physics Building and controlled by radiation protection personnel. Airborne monitoring will be performed using portable monitors as needed. A low-radiation background counting room is included in the Security and Health Physics Building.

Regulatory Guide 8.10 is incorporated into the PFSF operational considerations as described below:

1. Facility personnel are made aware of management's commitment to keep occupational exposures ALARA.
2. Ongoing reviews are performed to determine how exposures might be lowered.
3. There is a well-supervised radiation protection capability with specific, defined responsibilities.
4. Facility workers receive sufficient training.
5. Sufficient authority to enforce safe facility operation is provided to radiation protection personnel.
6. Modification to operating and maintenance procedures and to equipment and facilities are made where they substantially reduce exposures at a reasonable cost.
7. The radiation protection staff understands the origins of radiation exposures in the facility and seeks ways to reduce exposures.
8. Adequate equipment and supplies for radiation protection work are provided.

7.2 RADIATION SOURCES

The PFSF radiological shielding evaluation is based on the vendor's cask designs and its associated radiological source term and dose evaluations. The following discussion summarizes the assessments. The vendor's source terms bound fuel that will be stored at the PFSF. For more complete information, refer to the Holtec storage system SAR (Reference 6).

7.2.1 Characterization of Sources

The vendor determined source data for several types of spent fuel having differing characteristics (i.e. burnup, cooling time).

Holtec performed multiple calculations using the SAS2H and ORIGEN-S modules of the SCALE 4.3 system to confirm that the B&W 15X15 pressurized water reactor (PWR) and the GE 7X7 boiling water reactor (BWR) assemblies, the fuel assemblies with the highest UO_2 mass, have source strengths that bound all other PWR and BWR fuel assemblies. The design basis damaged fuel is GE 6X6. Holtec used these codes to determine the gamma and neutron source data for these fuels with the following assumed characteristics:

HI-STORM PWR Reference Intact Fuel

45 GWd/MTU	40 GWd/MTU
5-yr cooled	8-yr cooled
(Zircaloy clad)	(Stainless steel clad)

HI-STORM BWR Reference Intact Fuel

45 GWd/MTU	22.5 GWd/MTU
5-yr cooled	10-yr cooled
(Zircaloy clad)	(Stainless steel clad)

HI-STORM BWR Reference Failed Fuel

30 GWd/MTU

18-yr cooled

The HI-STORM reference fuels include intact Zircaloy and stainless steel clad fuels and failed BWR fuel described in HI-STORM SAR Tables 5.2.1 through 5.2.3. The reference fuel assemblies listed in these tables, and having the characteristics noted above, produce the highest neutron and gamma sources and the highest decay heat load. Reference failed BWR fuel listed produces the highest total neutron and gamma sources from the failed fuel assemblies at Dresden 1 and Humboldt Bay. Analyses are presented in the HI-STORM SAR which demonstrate that the storage of failed fuel in the HI-STORM storage system is bounded by the BWR intact fuel analyses during normal and accident conditions. Since lower enrichments produce higher gamma and neutron source terms for fuel having the same burnup and cooling time (with the neutron source more sensitive to enrichment), Holtec assumed enrichments of 3.6 percent for the PWR intact fuel and 3.2 percent for the BWR intact fuel, which are below the average enrichments normally used to obtain the burnups analyzed. An enrichment of 2.24 percent was used to describe the damaged fuel. A single full power cycle was used in the model to achieve the desired burnups. The results of gamma and neutron source determination are described in the sections that follow.

7.2.1.1 Fuel Region Gamma Source

The fuel region gamma source includes gammas originating from fission products, actinides, and activated materials in the active fuel region.

HI-STORM gamma source terms for the active fuel region and the remainder of the fuel assembly were computed by the SAS2H and ORIGEN-S modules of the SCALE 4.3 system and are given in HI-STORM SAR Tables 5.2.5 through 5.2.9 for the reference

PWR and BWR fuel assemblies, including intact Zircaloy and stainless steel clad fuels, and damaged BWR fuel. Energies in the range of 0.45 to 3.0 MeV were used in the shielding calculations. As discussed in Section 5.2.1 of the HI-STORM SAR, gamma energies below 0.45 MeV are too weak to penetrate the storage or transfer casks and gamma energies above 3.0 MeV are too few to contribute significantly to external dose. Methodology used by Holtec to account for the gamma source from activated non-fuel components in the fuel region is described in Chapter 5 of the HI-STORM SAR.

The stainless steel clad fuel has a higher source in the 1.0 to 1.5 MeV energy range due to cobalt activation. However, the Zircaloy clad fuel has a higher source in all other energy groups. The photons/sec in the higher energy groups and the total photons/sec for the Zircaloy fuel are higher than the values for the stainless steel fuel. As noted below, the neutron source strengths for the Zircaloy clad fuels were shown to bound those for the stainless clad fuels.

7.2.1.2 Non-Fuel Region Gamma Source

The gamma sources for the non-fuel regions of a fuel assembly are almost entirely due to Co-60 in activated metal components. The non-fuel region component masses identified in HI-STORM SAR Table 5.2.1 (based on References 10, 11, and 22) were used to obtain assumed masses of steel for various components, which are larger than those of most fuel assemblies. A high concentration of cobalt in steel was conservatively assumed to arrive at an initial Co-59 impurity level. The grams of impurity were then input to ORIGEN-S to calculate Co-60 activation levels for various burnups and decay times, using methodology developed from Reference 9. The ORIGEN-S runs were based on core region flux levels for full power operation. This calculated Co-60 activation level for the active fuel region was then modified by appropriate scaling factors from Reference 9, which reflect the lower neutron fluxes at the tops and bottoms of a fuel assembly. HI-STORM SAR Table 5.2.10 provides the scaling factors that were used to calculate the Co-60 activation of different components in PWR and BWR fuel assemblies. HI-STORM SAR Tables 5.2.12 and 5.2.13 identify the Curies of Co-60 calculated in various regions of the reference fuel assemblies for Zircaloy and stainless steel clad fuels. These regions include, for PWR fuel, the lower end fitting, gas plenum springs, gas plenum spacer, incore grid spacers, and upper end fitting; and for BWR fuel, the lower end fitting, gas plenum springs, expansion springs, grid spacer springs, the upper end fitting, and the handle.

7.2.1.3 Neutron Source

Neutrons are produced in the active fuel region by spontaneous fission sources from various actinides and alpha/neutron reactions. The primary neutron source is the spontaneous fission of Cm-244. HI-STORM neutron sources for the PWR and BWR fuels, determined using the SAS2H and ORIGEN-S codes, are shown in HI-STORM SAR Tables 5.2.16 through 5.2.20. These tables present the neutron sources for HI-STORM reference fuels, including intact Zircaloy and stainless steel clad fuels and damaged BWR fuel. The neutron source strengths for the Zircaloy clad fuels are greater than the source strengths for the stainless steel clad fuels, for all neutron energy groups.

Unlike the gamma source spectrum, the neutron source spectrum does not vary significantly with fuel burnup level or cooling time. Holtec assumed enrichments of 3.6 percent for the PWR fuel and 3.2 percent for the BWR fuel, which are below the average enrichments normally used to obtain the burnups analyzed, as indicated in the OCRWM LWR Database (Reference 8). Low initial enrichments are assumed since the neutron source strength increases substantially as initial enrichment decreases for LWR fuel of a given burnup.

7.2.2 Airborne Radioactive Material Sources

Loading of spent fuel into the canisters takes place at the originating nuclear power plants where procedures are in place to prevent the spread of contamination. The canisters are dried and seal welded within the controlled environment of the originating nuclear power plant. Once the canister is dried and seal welded, there are no credible off-normal events or accidents that will cause breach of the canister and thus no credible releases of airborne radioactivity from the spent fuel assemblies.

During normal operation of the PFSF, the only potential source of airborne radioactivity is from loose surface contamination on the canister exterior, which could potentially be deposited there during fuel loading operations. However, measures are implemented at the originating nuclear power plants to prevent contaminating the canisters. For wet transfers in spent fuel pools utilizing the HI-STORM system, an inflatable seal is placed in the annulus between the canister and the HI-TRAC transfer cask and the annulus is filled with demineralized water prior to submerging the empty transfer cask/canister in the pool. The seal prevents contaminated spent fuel pool water from entering the annulus and contaminating the outer surface of the canister. The outside of the transfer cask is washed down after being lifted out of the spent fuel pool to remove loose surface contamination. For dry transfers, it is less likely for contamination of the canister to occur since the fuel loading process is done outside of the pool.

Following fuel loading operations at the originating nuclear power plants, a smear survey is performed to determine removable contamination levels on accessible outer canister surfaces near the top of the canister (canister lid and approximately 3 to 6 inches along canister sides down from the lid). In the event canister removable contamination levels (measured on canister top surfaces) exceed the criteria specified in the canister vendor's Certificate of Compliance, the canister will not be released for shipment to the PFSF. Canisters with unacceptable levels of removable contamination must be decontaminated prior to release for transport to the PFSF. Once the shipping cask arrives at the PFSF and its closure is removed, a smear survey of accessible portions of the canister is again performed. If removable surface contamination levels on the top of the canister exceed the limits specified in the PFSF Technical Specifications (22,000 dpm/100 cm^2 beta/gamma and 2,200 dpm/100 cm^2 alpha), the canister is returned to the originating nuclear power plant for decontamination.

Section 8.1.5 evaluates doses resulting from an off-nomal event involving the postulated release of Co-60 contamination assumed to cover the entire exterior surface of a canister at a concentration of 1.0 E-4 $\mu Ci/cm^2$ (approximately 22,000 dpm/100 cm^2). The evaluation concludes that the consequences of such a release to an individual at a distance of 492 ft (150 meters) would be 0.03 mrem committed effective dose equivalent, with lower doses at the OCA boundary, a distance of 1,640 ft (500 meters) from the Canister Transfer Building at its nearest point.

Section 8.2.7 evaluates canister leakage under hypothetical accident conditions, conservatively assuming cladding rupture of all the fuel rods within the canister, and release of conservative fractions of fission and activation products from the fuel and canister. The doses calculated to result from this hypothetical accident are below the 5 rem total effective dose equivalent regulatory limit specified in 10 CFR 72.106 (b) at the OCA boundary.

7.3 RADIATION PROTECTION DESIGN FEATURES

7.3.1 Installation Design Features

A description of the PFSF layout and design is provided in Section 4.1. The PFSF layout and design are in accordance with the facility and equipment design features identified in Position 2 of Regulatory Guide 8.8, as described in Section 7.1.2.

The PFSF has the following design features that ensure that exposures are ALARA:

- The site is located far from population centers. The distance to the nearest town is over 10 miles. The military town of Dugway, with a population of approximately 1,700, is located about 12 miles south of the PFSF. Terra, a small residential community of about 120 people, is located 10 miles east-southeast of the PFSF. There are about 36 residents within a 5-mile radius of the PFSF.

- The only sources of radiation at the PFSF are the sealed canisters containing spent fuel assemblies. These canisters will always be shielded by shipping, storage or by transfer casks during canister transfer operations.

- Low-level radioactive waste will be packaged and staged in LLW containers in the LLW holding cell (discussed in Chapter 6) while awaiting shipment to a LLW disposal site. Because of the low activity inventory associated with any LLW, dose rates on the outer surfaces of the LLW containers are expected to be negligible.

- Measures are taken at the originating nuclear power plants to prevent loose surface contamination levels on the exterior of the canisters, as discussed in

Section 7.2.2. Controls assure that canisters are not transported to the PFSF unless contamination levels are within specified limits.

- The canisters will be sealed by welding, eliminating the potential for release of radioactive gases or particles.

- The canisters will not be opened, nor will spent fuel assemblies be unloaded at the PFSF.

- The fuel will be stored dry inside the canisters, so that no radioactive liquid is available for release.

- The shipping, transfer, and storage casks are heavily shielded to minimize external dose rates.

- The PFSF site layout provides substantial distance between the cask storage area and the OCA boundary, minimizing radiation exposures to individuals outside the OCA and assuring offsite dose rates are well below the 10 CFR 72.104 criteria. The closest distance from a storage pad to the OCA boundary is 2,119 ft (646 meters).

- The Administration Building is located approximately 2,580 ft (786 m) and the Operations and Maintenance Building is located approximately 1,960 ft (597 m) from the nearest storage pad. These distances provide separation of radioactive material handling and storage functions from other functions on the site. The Security and Health Physics Building, located near the storage area to maintain security and radiological access control, is provided with radiation shielding, as is the Canister Transfer Building.

- The location of the Canister Transfer Building inside the RA minimizes the route between the handling facility and storage pads, provides for minimal other traffic on the route, and maintains substantial distance from the OCA boundary.

- There are no radioactive liquid wastes associated with the PFSF.

As shown in Section 7.3.3.5, the design of the PFSF assures that dose rates at the OCA fence are sufficiently low that individuals at the fence will not exceed 25 mrem per year whole body dose, in compliance with the requirements of 10 CFR 72.104.

The PFSF building ventilation systems are not designed for any special radiological considerations since there is no credible scenario for which a significant radioactive release could occur. Shielding of the canisters is provided by the storage casks and by the shipping and transfer casks during canister receipt, transfer and, offsite shipping operations. Shielding is provided in the design of the Canister Transfer and the Security and Health Physics Buildings for additional radiation dose protection.

The general area inside the RA fence is a restricted area, as defined by 10 CFR 20, and will be controlled in accordance with applicable requirements of 10 CFR 20, with personnel dosimetry required. Certain areas within the RA will be designated as Radiation Areas, and specific locations within the RA have the potential to be High Radiation Areas, and will be posted and controlled in accordance with applicable requirements of 10 CFR 20. The cask load/unload bay, crane bay, cask transporter bay, and canister transfer cells inside the Canister Transfer Building will be designated as Radiation Areas whenever loaded canisters are present in these areas, since the potential exists for dose rates to exceed 5 mrem/hr in these areas. Upon removal of the impact limiters from the shipping casks in the cask load/unload bay of the Canister

Transfer Building, the potential exists for dose rates in the vicinity of the top and/or bottom of the casks to exceed 100 mrem/hr in localized areas, and these localized areas will be posted as High Radiation Areas, with necessary controls applied. The external walls of the Canister Transfer Building, adjacent to the east, south, and west sides of the cask load/unload bay, are 2 ft thick concrete, with steel roller bay doors in the truck/rail entrance/exits at the east and west ends of the bay. Due to distances from the shipping casks when their impact limiters are removed, dose rates outside the Canister Transfer Building will be well below 100 mrem/hr. The concrete walls of the cask load/unload bay, and steel roller bay doors, will reduce dose rates outside the building to levels as low as is reasonably achievable.

It is anticipated that the canister transfer cells within the Canister Transfer Building (Figure 4.7-1) will be posted as High Radiation Areas during canister transfer operations, since the dose rates in the cells could potentially exceed 100 mrem/hr in localized areas 30 cm from cask surfaces. Due to distances from cask surfaces into the crane bay, cask transporter bay, and areas external to the Canister Transfer Building, dose rates will be well below 100 mrem/hr without credit for the shield walls that surround the canister transfer cells. The north wall of cell no. 1 (an external wall), and the west cell walls of all three cells (adjacent to the cask transporter bay), will be 2 ft thick concrete. The walls between the cells, the south wall of cell no. 3, and the east walls of all three cells, will be 1 ft thick concrete. The sliding doors will be steel with a polyethylene (or similar) shield, as necessary, to minimize neutron doses. The walls and doors provide radiation shielding that will limit the dose rates outside of the canister transfer cells during transfer operations to as low as is reasonably achievable. The walls and sliding doors of the canister transfer cells shall be seismically designed to withstand earthquake induced loads and remain in place following the PFSF design basis ground motion.

The east wall of the crane bay is 2 ft thick concrete, and it is expected that dose rates in the rooms and offices east of this wall will be less than 5 mrem/hr, even when shipping cask movements and canister transfer operations are in progress, and will not require posting as Radiation Areas. Dose rates in the vicinity of low level waste storage containers are expected to be insignificant, due to the relatively low quantities of radioactivity that will be stored in this area resulting from incidental cleanup of any contamination. Nevertheless, the 2 ft thick concrete north and east walls, and 1 ft thick concrete south and west walls of the Low Level Waste Room will assure that dose rates outside this room are as low as is reasonably achievable. No credit is taken for shielding by other walls of the Canister Transfer Building.

7.3.2 Access Control

The PFSF is designed to provide access control in accordance with 10 CFR 72. Access control to the RA is provided for both personnel radiological protection and facility physical protection. The physical protection program is covered in the Security Plan, which is classified and submitted as part of the License Application under separate cover.

The access control boundaries for the controlled and restricted areas are established along the site fence lines (see Figure 1.1-2, the PFSF Site Plan). The RA is that space which is controlled for purposes of protecting individuals from exposure to radiation or radioactive materials and for providing facility physical security. The boundary for the RA is the security fence where the dose rate is less than 2 mrem/hr, in accordance with 10 CFR 20.1301. The controlled area is the area inside the site boundary (delineated by the OCA fence). The dose rate beyond the OCA fence is less than 25 mrem/yr, in accordance with 10 CFR 72.104.

Access to the RA is controlled through a single access point in the Security and Health Physics Building (see Figure 1.2-1, the PFSF General Arrangement). Personal dosimetry is issued and controlled in this building to individuals entering the RA. Provisions exist in this building for donning and removing personal protective equipment, such as anti-contamination clothing and/or respirators, which could be necessary in the event of contamination in the Canister Transfer Building as a result of off-normal or accident conditions. Provisions for personnel decontamination are also contained in the Security and Health Physics Building. The RA also includes the cask storage area and Canister Transfer Building. In accordance with the PFSF Radiation Protection Program (Section 7.5), radiation protection personnel will monitor radiation levels in the RA and establish access requirements as needed.

7.3.3 Shielding

The storage systems are designed to maintain radiation exposures ALARA. The HI-STORM storage cask design objectives specified in Section 2.3.5.2 of the HI-STORM SAR are maximum contact dose rates of 40 mrem/hr on the side, 10 mrem/hr at the top, and 60 mrem/hr at the air vents.

Special design provisions are not required to shield low-level radioactive waste materials that could potentially be generated due to the low activity inventory that would be associated with these materials. As discussed in Section 6.4, low-level solid waste would be expected to consist of smears, disposable clothing, tape, blotter paper, rags and related health physics material. This material will be processed and temporarily stored in the LLW holding cell of the Canister Transfer Building, while awaiting removal to a licensed LLW disposal facility. The material will be packaged and stored in sealed LLW containers. The LLW containers provide necessary shielding, and dose rates on the outside surfaces of the drums are expected to be negligible. In the unlikely event materials are stored in the holding cell with significant activity levels, temporarily located shielding may be used to maintain dose rates in the area ALARA, as determined by radiation protection personnel.

7.3.3.1 Shielding Configurations

Chapter 5 of the HI-STORM SAR identifies the shielding materials and geometries of the transfer and storage casks and describes the codes used to model shielding and assess cask dose rates.

The thick steel canister lid provides radiation protection for workers engaged in the canister transfer operations, as well as substantial shielding in the top axial direction during storage. Additional shielding in the top axial direction is provided by the lid on the storage cask. Shielding located axially below the spent fuel assemblies consists of the steel canister bottom plate and a thick section of concrete/steel beneath the canister. Radiation shielding in the radial direction during storage is provided primarily by the steel canister shell, the steel storage cask liners, and the approximately 2 1/4 ft thick concrete walls of the storage cask. The designs of the inlet and outlet ducts for the storage cask prevent direct radiation streaming from the canister to the cask exterior. Shielding materials and geometries are described in detail in the HI-STORM SAR.

The transfer cask is designed to reduce the dose rates from a canister loaded with fuel to ALARA, enabling personnel to perform the canister transfer operation without exposure to excessive dose rates. With a canister in the transfer cask, shielding at the top is provided primarily by the canister lid. Radial shielding is provided by the transfer cask, composed of steel shells with gamma and neutron shielding materials. The transfer cask lower shield doors consist of a sandwich of steel, lead, and neutron shield materials for Holtec's HI-TRAC transfer cask.

In the HI-STORM shielding model, the canister internals were explicitly modeled and fuel assembly materials were homogenized. The end fittings and plenum regions were modeled as homogeneous regions of steel. This homogenization was determined to result in a noticeable decrease in computer run time without any loss of accuracy. Several conservative approximations were made in the model, such as not including shielding provided by the fuel spacers, as described in Chapter 5 of the HI-STORM SAR.

7.3.3.2 Shielding Evaluation

The shielding analyses, documented in Chapter 5 of the HI-STORM SAR, model generic reference PWR and BWR fuel (Section 7.2.1). Dose rates projected from the HI-STORM storage cask will envelope those dose rates that will actually be produced at the PFSF and provide conservative dose rates for purposes of assessing onsite and offsite radiation exposures. The results of these shielding analyses are summarized in the following sections.

The storage cask vendor considered different types of canisters in the shielding analyses: PWR canisters that contain 24 PWR spent fuel assemblies and BWR canisters that contain 68 assemblies.

The HI-STORM shielding analyses were performed using the MCNP code. Discussions of the modeling code and methodologies are included in Chapter 5 of the HI-STORM SAR.

7.3.3.3 Dose Rates for a Single Storage Cask

Gamma and neutron dose rates for a single storage cask were determined at the following locations for the HI-STORM storage cask, assuming PWR and BWR reference fuels having various burnups and cooling times:

- on contact with the side of the storage cask,
- one meter from the side of the storage cask,
- on contact with the top of the storage cask,
- one meter from the top of the storage cask,
- at the air inlet duct openings, and
- at the air outlet duct openings.

Resulting dose rates for a single storage cask are presented in Table 7.3-1 for the PWR and BWR reference fuel cases analyzed and documented in the vendor SAR. Figure 7.3-1 identifies the locations of the points relative to the storage casks at which dose rates were calculated.

7.3.3.4 Dose Rates for a Transfer Cask

Gamma and neutron dose rates for a transfer cask were also determined at various locations for the HI-TRAC transfer cask. Table 7.3-3 presents calculated dose rates for the HI-TRAC transfer cask, assuming the cask contains an MPC-24 canister of PWR reference fuel with 45 GWd/MTU burnup, 9-year cooling time. Figure 7.3-2 identifies the locations of the points relative to the transfer cask at which dose rates were calculated.

7.3.3.5 Dose Rates at Distances from the PFSF Array of Storage Casks

Gamma and neutron dose rates at various distances from a single storage cask were calculated, assuming fuel with conservative burnup and cooling time representative of high radiation source fuel expected to be stored at the PFSF, instead of reference fuel. The results of these single storage cask calculations were then used in support of the dose rate vs. distance analyses for the fully loaded PFSF array of 4,000 casks.

The basis for these calculations is that all 4,000 casks contain 40 GWd/MTU burnup and 10-year cooled PWR spent fuel, with a low initial enrichment assumed for this burnup. The assumption of 40 GWd/MTU burnup and 10-year cooled PWR fuel is intended to provide a conservative representation of dose rates associated with average fuel in the PFSF array of 4,000 casks at the restricted area (RA) fence and owner controlled area (OCA) boundary. It is assumed that the design inventory of 4,000 storage casks stored on the storage pads has these characteristics for the purpose of calculating dose rates for comparison with the applicable limits of 10 CFR 20.1301 (dose rate less than 2 mrem/hr for unrestricted areas) and 10 CFR 72.104 (annual dose to an individual at the OCA boundary of less than 25 mrem).

A more realistic cooling time of 10 years (as compared to 5-year cooled reference fuel) is used since it is not reasonable to assume that 4,000 loaded storage casks are stored at the PFSF with an average cooling time of 5 years. This is based on the following: (1) the majority of the nuclear power plant spent fuel currently available to be stored at the PFSF is over 10 years old; (2) the minimum cooling time requirement for transporting 40 GWd/MTU PWR fuel is 12 years for the Holtec HI-STAR shipping cask system (Reference 24); and (3) the anticipated maximum storage cask loading rate at the PFSF is one cask per operating day or about 200 casks per year, which at this rate would take 20 years for the PFSF to be filled. Therefore, a 10-year cooling time is considered to be conservative for the 4,000-cask PFSF array since the actual average cooling time is expected to be much greater than 10 years. 40 GWd/MTU is considered to represent a conservative burnup for the majority of fuel stored at the PFSF.

DOE's Energy Information Administration's Service Report entitled "Spent Nuclear Fuel Discharges from U.S. Reactors - 1994" (Reference 19), provides information regarding characteristics of spent fuel in the U.S. This report was reviewed to evaluate average burnups and cooling time associated with the spent fuel inventory at the end of 1994. At this time, the spent fuel inventory from PWRs was approximately 19,000 MTU, and the inventory from BWRs approximately 11,000 MTU, for a total inventory of approximately 30,000 MTU (Table 5 of Reference 19). This spent fuel inventory represents 75% of the capacity of the PFSF. While it is recognized that provisions already exist for storage of some of this spent fuel and the PFSF will not furnish storage for this entire inventory, data associated with this spent fuel is considered representative of fuel that the PFSF could be expected to receive. The weighted average burnup (weighted by MTU) for the BWR spent fuel inventory in the U.S. was calculated from Table 6 of Reference 19 to be approximately 23.8 GWd/MTU, and the weighted average burnup for the PWR spent fuel inventory in the U.S. was calculated from Table 7 of Reference 19 to be approximately 32.4 GWd/MTU (Reference 20).

Weighted average cooling times were also calculated from the data presented in Tables 6 and 7 of Reference 19, conservatively assuming that the PFSF receives 2,000 MTU of spent fuel each year, beginning in the year 2002, until all 30,000 MTU have been received (in year 2016). It was assumed that the older spent fuel, whether BWR or PWR, is received first. Based on these assumptions, the weighted average cooling time for spent fuel assumed to be received at the PFSF was calculated to be 23.0 years (Reference 20).

Because of the large inventory of spent fuel taken into account (approximately 30,000 MTU), this is considered to be a reasonable representation of typical fuel that will be received at the PFSF. Based on this evaluation of the spent fuel inventory in existence in the U.S. at the end of 1994, it is determined that use of the 40 GWd/MTU burnup and 10-year cooled PWR fuel assumed in the shielding analyses to evaluate dose rates at the RA fence and OCA boundary from the array of 4,000 casks is conservative.

Holtec computed dose rates at the surface of a HI-STORM storage cask and at various distances from the cask, assuming fuel with 40 GWd/MTU burnup and 10-year cooling time, using the MCNP code. The HI-STORM SAR shows that a HI-STORM storage cask containing a PWR canister (MPC-24) has higher contact dose rates on the top and at the outlet duct opening than a HI-STORM storage cask containing a BWR canister (MPC-68), for fuel of identical burnup and cooling times. The dose rate at the midplane for an overpack containing a PWR canister is essentially the same as that for a storage cask containing a BWR canister. Therefore, it was determined that the dose rates from a HI-STORM storage cask containing a PWR canister will bound dose rates from a storage cask containing a BWR canister, and dose rates at distances from the PFSF array were assessed conservatively assuming all storage casks are loaded with PWR canisters. The primary radiation source terms accounted for in Holtec's analysis were: gamma and neutron sources from the decay of fission products and the gamma source from the decay of Co-60 in the fuel assembly end-fittings. Secondary radiation source terms accounted for were secondary neutrons from fast fission in the fuel and secondary gammas from prompt neutron interaction in the canister and overpack. The canister and overpack were modeled in full three-dimensional detail using the MCNP code, in the same manner that the storage cask was modeled with reference fuel, as described in the HI-STORM SAR.

The single storage cask dose rate versus distance data for HI-STORM casks containing 40 GWd/MTU, 10-year cooled fuel are shown in Table 7.3-5 for the following four components: gammas and neutrons from the cask side and top. These data were used, along with the layout of the cask array at the PFSF (see PFSF Site Plan, Figure 1.1-2), to determine dose rates at various distances, including the RA fence and the OCA boundary from the PFSF array of 4,000 casks. The following paragraphs summarize the methodology used and results of dose rate projections from the PFSF array, assuming the PFSF is filled with HI-STORM storage casks containing 40 GWd/MTU, 10-year cooled fuel.

Holtec used the dose rate vs. distance data from a single HI-STORM storage cask, shown in Table 7.3-5, to project dose rates at various distances from the PFSF array, assumed to be filled with 4,000 HI-STORM storage casks containing 40 GWd/MTU, 10-year cooled fuel (Reference 13). The dose rate contributions from the tops and sides of the casks were separately analyzed using the MCNP code. The total dose rate from the tops of casks is a summation of the gamma and neutron top doses from all 4,000 casks, where the actual distance from each cask to the dose receptor is accounted for.

The total dose from the sides of the casks is a summation of side doses from all 4,000 casks where the distances within the facility and self-shielding of one row of casks by another row are accounted for. The fraction of radiation blocked by a cask directly in front of another cask was calculated by MCNP and used in the determination of total side dose rates. Self-shielding effects are different along the north/south faces than along the east/west faces because of the different geometries, as seen in Figure 1.2-1.

It was impractical to model the entire facility in MCNP, therefore, numerous smaller calculations were performed for configurations of several casks and combined in a conservative fashion to accurately estimate dose rates from the sides of the casks at various distances from the PFSF array. Modeling of configurations of casks determined: the number of casks in a single row along the east/west and north/south faces that effectively constitute an infinite line at various distances from the dose receptor; the fractional increases in dose rates when a second row of casks is added directly behind the first row along the east/west and north/south faces at various distances; and the fractional increases when two more rows of casks are added behind the first two rows (adding a second column of storage pads, with two rows of casks per pad) along the east/west faces at various distances.

The results of the dose rate vs. distance analysis for the PFSF array full of HI-STORM storage casks are given in Table 7.3-7 (Reference 13). Total dose rates at the RA fence (150 ft from the nearest storage pads) at the north side of the array are 1.68 mrem/hr. The RA fence south of the array is 265 ft from the nearest storage pads, so will have lower dose rates. Total dose rates at the RA fence on the east and west sides of the array (also 150 ft from the nearest storage pads) are 1.37 mrem/hr. It is considered that dose rates calculated by this analysis are very conservative, since PWR fuel having 40 GWd/MTU burnup and 10-year cooling time represents relatively "hot" fuel, which will produce substantially higher array dose rates than PFSF average fuel. Spent PWR fuel having 35 GWd/MTU burnup and 20-year cooling time is considered to be representative of typical fuel expected to be received at the PFSF, as explained in Section 7.4. Applying scaling factors to calculate dose rates assuming all 4,000 HI-STORM casks contain this typical fuel, the highest dose rate at the RA fence for this typical fuel is 0.57 mrem/hr (Reference 23).These dose rates are less than the 2 mrem/hour criteria for unrestricted areas specified in 10 CFR 20.1301 and are therefore acceptable. Assuming all 4,000 casks contain the relatively hot PWR fuel having 40 GWd/MTU burnup and 10-year cooling time, the total dose rates at the OCA boundary were calculated to be 2.80 E-3 mrem/hr at a point on the boundary 1,969 ft (600 meters) north of the RA fence, and 1.69 E-3 mrem/hr at a point on the boundary 600 meters west of the RA fence (Reference 13). Dose rates will be lower at points along the south and east sides of the OCA boundary, since these points are further from the storage casks than the north and west OCA boundaries. Conservatively assuming a hypothetical individual spends 2,000-hours per year at the north OCA boundary results in a maximum annual dose of 5.60 mrem. The maximum annual dose at the OCA boundary assuming typical fuel expected to be received at the PFSF (scaling the dose rates to be representative of PWR fuel having 35 GWd/MTU burnup, 20-year cooling time) is 1.90 mrem (Reference 23). These dose rates are less than the 25 mrem criteria specified in 10 CFR 72.104 for maximum permissible annual whole body dose to any real individual located beyond the controlled area boundary and are therefore acceptable.

Dose at Nearest Residence

The approximate distance to the nearest residence is 2 miles east-southeast of the PFSF. At distances greater than several thousand feet, the accuracy of computer code calculational techniques becomes questionable. The error bands in statistical codes like MCNP become large and for deterministic codes like Skyshine, the conditions may be beyond the range of the codes data. However, dose rates were estimated that could occur at long distances from the PFSF, assuming the PFSF array of 4,000 HI-STORM storage casks loaded with 40 GWd/MTU, 10-year cooled PWR fuel, and conservatively taking no credit for any intervening shielding from berms, natural terrain or buildings at the PFSF. Holtec estimated the dose rate at 2.0 miles from the PFSF by extrapolating the maximum dose rate at the OCA boundary (2.80 E-3 mrem/hr) out to a distance of 2.0 miles using a power curve (Reference 13). The result was 3.89 E-6 mrem/hr, which would result in an annual dose of 0.034 mrem at a distance of 2.0 miles from the OCA boundary, assuming a person continually present (8,760 hrs/yr) at this location.

7.3.4 Ventilation

10 CFR 72.122(h)(3) requires that ventilation systems and off-gas systems be provided where necessary to ensure the confinement of airborne radioactive particulate materials during normal or off-normal conditions. However, there are no special ventilation systems installed in the PFSF facilities. There are no credible scenarios that would require installation of ventilation systems to protect against off-gas or particulate filtration.

7.3.5 Area Radiation and Airborne Radioactivity Monitoring Instrumentation

10 CFR 72.122(h)(4) requires the capability for continuous monitoring of the storage system to enable the licensee to determine when corrective action needs to be taken to maintain safe storage conditions. This is not applicable to the PFSF because the canisters are sealed by welding and with the canisters in storage casks and the casks on the storage pads, there are no credible events that could result in releases of radioactive material from within the canisters or unacceptable increases in direct radiation levels. Area radiation and airborne radioactivity monitors are therefore not needed at the storage pads. However, TLDs will be used to record dose rates in the RA and along the OCA boundary fence. TLDs provide a passive means for continuous monitoring of radiation levels and provide a basis for assessing the potential impact on the environment.

TLDs will be located along the RA and OCA boundary fence such that each side of the boundary has one TLD at each corner, one on the N-S or E-W centerlines of the storage cask array, and one equidistant between each corner and the N-S or E-W centerlines. This provides a total of 16 TLD locations for each boundary. These TLDs will be used to record dose rates along the RA and OCA boundary fence and will provide documentation that radiation levels at these boundaries are within regulatory limits. TLDs will also be placed on the outside of several buildings as follows: NW corner of the Administration Building, NW corner of the Operations and Maintenance Building, NW corner of the Canister Transfer Building, and at three locations along the West wall of the Security and Health Physics Building. Additionally, TLDs will be located at strategic locations inside the Canister Transfer Building and the Security and Health Physics Building where personnel will normally be working. These TLDs will serve as a backup for monitoring personnel radiation exposure and maintaining this exposure ALARA. For redundancy, each TLD location mentioned above will house a set of two TLDs. The TLDs will be retrieved and processed quarterly. The TLDs will primarily detect gamma radiation and have a lower limit of sensitivity of approximately 0.02 mrem.

Local radiation monitors with audible alarms will be installed in the Canister Transfer Building. These will provide warning to personnel involved in the canister transfer operation of abnormal radiation levels that could possibly occur during transfer operations. Because of the measures taken at the originating nuclear power plants to minimize loose surface contamination levels on the exterior of the canisters during fuel loading operations, as discussed in Section 7.2.2, and PFSF Technical Specification limits on surface contamination concentrations, whose bases are included in Chapter 10, it is unlikely that canister transfer operations would generate significant levels of airborne contaminants. Airborne radioactivity concentrations will be detected by continuous air monitors located in the exhaust of each canister transfer cell. The continuous air monitors will include local alarms to warn operating personnel in the unlikely event of an airborne release, remote alarm in the Security and Health Physics Building alarm station to ensure coverage at all times, and charting capability to provide data necessary to quantify any release. The radiological alarm systems will be designed with provisions for calibration and operability testing. There are no liquid or gaseous effluent releases from the PFSF. This satisfies the requirements of 10 CFR 72.126(b) and (c).

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7.4 ESTIMATED ONSITE COLLECTIVE DOSE ASSESSMENT

The shipping, transfer and storage casks are designed to limit dose rates to ALARA levels for operators, inspectors, maintenance, and radiation protection personnel when the canisters are being transferred from the shipping to the storage casks, when the storage casks are being moved to the storage pads, and while the storage casks are being stored on the pads.

Table 7.4-1 shows the estimated occupational exposures to PFSF personnel during receipt of the HI-STAR shipping cask, transfer of the canister from the shipping cask to the HI-STORM storage cask using the HI-TRAC transfer cask, movement of the storage cask to the pad, and emplacement on the pad. The estimated occupational exposures were calculated in Reference 20. The operational sequence for these operations is also described in Chapter 5.

Dose rate values include both gamma and neutron flux components, and are based on PWR fuel with 35 GWd/MTU burnup and 20-year cooling time. Fuel with these characteristics is considered to be representative of typical fuel that will be contained in canisters handled at the PFSF, and dose estimates based on fuel with these characteristics are considered to be realistic and reflect expected personnel exposures. For this reason, the values of burnup and cooling time used in Section 7.3.3.5 to assess dose rates at boundaries from the array of 4,000 casks, and shown to be conservative in that section, were not applied to estimate worker integrated doses. Evaluation of weighted average burnups and cooling times of the nations' PWR and BWR spent fuel inventory in existence at the end of 1994, as discussed in Section 7.3.3.5, indicates an overall weighted average burnup (weighted by metric tons uranium) of approximately 32.4 GWd/MTU for PWR fuel and approximately 23.8 GWd/MTU for BWR fuel, with a weighted average cooling time for both types of fuel of approximately 23.0 years (assuming 30,000 MTU of spent fuel is received during the first 15 years of PFSF operation). Based on this evaluation, the 35 GWd/MTU burnup and 20-year cooling time characteristics for spent fuel assumed in the onsite dose assessment are considered to be representative of typical fuel expected to be received at the PFSF.

From Table 7.4-1, the total dose from receipt of a loaded shipping cask, transfer of the canister into a storage cask, movement of the storage cask to the pad, and performance of initial surveillances is estimated to be about 247 person-mrem for the HI-STORM system. Assuming a storage cask loading rate of 200 casks per year, the total annual dose to operations and Radiation Protection personnel involved in these operations is estimated to be approximately 49 person-rem, assuming all storage casks are HI-STORM casks. Occupational doses to individuals will be administratively controlled to ensure that they are maintained below 10 CFR 20.1201 limits and ALARA.

Temporarily positioned shielding will be used during transfer operations to reduce dose rates from streaming paths or relatively high radiation areas where its use will result in a net reduction in worker exposures. The effects of temporarily positioned shielding, calculated in Reference 20, are considered in the Table 7.4-1 dose estimates for canister transfer operations.

Occupational exposures are also estimated to security personnel and PFSF personnel that conduct inspections, surveillances, and maintain the storage systems. These estimates are based on the assumption that the PFSF is at its 4,000 storage cask capacity. It is estimated that security personnel that conduct security inspections will accrue approximately 0.62 person-rem annually, based on one 1 hour inspection per shift (3 shifts per day, 365 days per year) along the RA fence, using the 0.57 mrem/hr dose rate at the fence discussed in Section 7.3.3.5. It is considered that dose rates inside the Security and Health Physics Building are negligible due to shielding provided by the building structure. One visual inspection per quarter is required to be performed for each storage cask to check for the buildup of debris at the inlet ducts and to inspect the cask exterior. Assuming one person spends 1.0 minute inspecting each cask, in an average dose field of 12.4 mrem/hr during the inspection, this surveillance will result in approximately 0.83 person-rem per quarter to PFSF personnel conducting the inspections, for a total of 3.3 person-rem annually. The 12.4 mrem/hr average dose field estimate near a cask inside the cask array is based on the Reference 21 calculation, which assumes that storage casks contain "typical" PFSF fuel, represented by PWR fuel with 35 GWd/MTU burnup and 20 year cooling time. Conservatively assuming that 5 percent of the 4,000 casks require clearing of debris from the inlet ducts once a year at 10 minutes each (Reference 21), in a dose field of 12.4 mrem/hr, an additional annual dose of 0.41 person-rem is estimated. Monitoring of temperatures representative of the thermal performance of the casks will be performed remotely with a data acquisition system and will not result in significant exposure. Based on the above, the total dose to personnel involved in security inspections, surveillance, and storage cask maintenance operations is estimated to be 4.3 person-rem annually, assuming all storage casks are HI-STORM casks.

PFS considers that the occupational exposures calculated and reported above are conservative (i.e., actual doses to individual workers at the PFSF will be a fraction of those calculated). Additionally, doses to workers will be closely monitored throughout operations involving loaded canisters at the PFSF. Based on actual doses received from the first few canister transfer operations, measures will be implemented to maintain occupational exposure ALARA. These may include additional shielding, optimizing handling operations to maximize distance to the source, and reducing time in the radiation field. PFS is committed not only to maintaining occupational exposures below federal guidelines but to maintaining exposures ALARA as well.

A combination of building location and shielding will minimize the dose to staff personnel working in the PFSF facilities. The west sides of the Canister Transfer Building and Security and Health Physics Building are approximately 425 ft (130 meters) and 948 ft (289 meters), respectively, from the nearest storage pad (see Figure 1.2-1). The building structures will provide shielding to reduce doses to workers in the buildings from the cask storage area to levels that are ALARA. The Operations and Maintenance Building and Administration Building will be located near the entrance gate to the OCA (see Figure 1.1-2). The Administration Building is further from the storage pads (2,580 ft) than the nearest distances to the OCA boundary (2,119 ft), and the Operations and Maintenance Building is nearly as far away (1,960 ft). Dose rates at these buildings will be less than 25 mrem/yr (at a 2,000 hr/yr occupancy rate) without consideration for shielding provided by the building structures.

7.5 RADIATION PROTECTION PROGRAM

7.5.1 Organization

The PFSF Radiation Protection Manager reports to the General Manager and to the Board of Managers (Figure 9.1-3) and is responsible for administering the radiation protection program and for the radiation safety of the facility. Minimum qualification requirements are set forth in Section 9.1.3.1. The radiation protection and ALARA programs are discussed in Section 7.1.

Responsibilities of the PFSF Radiation Protection Manager include the following:

- Administer the Radiation Protection program policies and procedures
- Review and approve radiation protection procedures
- Coordinate radiation protection group activities with operations and maintenance personnel
- Ensure adequate staffing, facilities, and equipment are available to perform the functions assigned to radiation protection personnel
- Establish goals for the Radiation Protection program
- Initiate and implement an exposure control program that factors dosimetry results into operational planning
- Issue or rescind "stop work" orders as appropriate
- Ensure that locations, operations, and/or conditions that have the potential for causing significant exposures to radiation are identified and controlled
- Review and approve training programs related to work in radiological areas or involving radioactive material
- Administer shipments of solid radioactive waste offsite for disposal
- Review root causes and corrective actions for incidents and deficiencies associated with Radiation Protection

- Ensure an effective ALARA program is maintained, in accordance with the guidance provided in Regulatory Guides 8.8 and 8.10
- Supervise the collection, analysis and evaluation of data obtained from radiological surveys and monitoring activities
- Participate in the event of an emergency, as required

Radiation protection technicians report to the Radiation Protection Manager. Responsibilities of the radiation protection technicians include the following:

- Conduct radiation, contamination, and airborne surveys and prepare complete and accurate records
- Prepare Radiation Work Permits to control access to and activities in radiologically controlled areas
- Identify and post radiation, contamination, hot particle, airborne and radioactive material areas in accordance with 10 CFR 20 requirments
- Monitor PFSF operations to assure good radiological work practices
- Implement ALARA program requirements
- Maintain and calibrate portable monitoring instruments
- Issue "stop work" orders whenever activities have the potentail to jeopardize the health and safety of workers, visitors, or the general public
- Verify proper packaging of any radioactive material
- Participate in the event of an emergency, as required

7.5.2 Equipment, Instrumentation, and Facilities

A sufficient inventory and variety of operable and calibrated portable and fixed radiological instrumentation will be maintained to allow for effective measurement and control of radiation exposure and radioactive material and to provide back-up capability for inoperable equipment. Equipment will be appropriate to enable the assessment of sources of gamma, neutron, beta, and alpha radiation, including the capability to measure the range of dose rates and radioactivity concentrations expected. Radiation protection procedures will govern instrument calibration, instrument inventory and control, and instrument operation.

Portable survey and personnel monitoring instrumentation will include, but not be limited to, the following:

- Low-level contamination meters
- Beta/gamma portable survey meters
- Alarming beta/gamma personnel friskers
- Portable air samplers

Radiological instrument storage, calibration and maintenance facilities will also be located in the Security and Health Physics Building, along with a low-radiation background counting room containing laboratory equipment for measuring radioactivity.

Area radiation monitors are utilized in the Canister Transfer Building since the operations performed in this building (shipping cask receipt, inspection, and canister transfer operations) pose the greatest risk to the operating staff for radiation exposure. These monitors have audible alarms to warn operating personnel of abnormal radiation levels. Area radiation monitors are not utilized outside the Canister Transfer Building since these areas have relatively low area radiation levels and there are no operations performed in these areas which could result in a rapid change in radiation level and pose a risk for over-exposure of personnel.

The RA is approximately 99 acres and is surrounded by a chain link security fence and an outer chain link nuisance fence with an isolation zone and intrusion detection system between the two fences. Access to the RA is controlled through a single access point in the Security and Health Physics Building (see Figure 1.2-1, the PFSF General Arrangement). Personal dosimetry is issued and controlled in this building to individuals entering the RA. External radiation dose monitoring will be accomplished through the use of thermoluminescent dosimeters (TLDs) and self reading dosimeters (SRDs) or digital alarming dosimeters (DADs). All operating personnel inside an active canister transfer cell will be required to utilize alarming dosimeters during the canister transfer process to warn of excessively high direct radiation to maintain exposures ALARA, thereby providing additional assurance that occupation exposures will not exceed the limits of 10 CFR Part 20. The official record of external dose to beta and gamma radiations will normally be obtained from the TLDs, with SRDs or DADs used as a means for tracking dose between TLD processing periods and as a backup to TLDs. Self-reading dosimeters will be administered in accordance with the guidance in Regulatory Guide 8.4 (Reference 15).

Provisions exist in the Security and Health Physics Building for donning and removing personal protective equipment, such as anti-contamination clothing and/or respirators, which could be necessary in the event of contamination in the Canister Transfer Building due to off-normal or accident conditions. The respiratory protection program will be established in accordance with 10 CFR 20 and consistent with the guidance of NUREG-0041 (Reference 16). Radiation protection procedures will include the conduct of bioassays including criteria for the performance of bioassay, dose tracking and methods for data analysis and interpretation. The bioassay program will be based on NRC Regulatory Guide 8.26 (Reference 17) and Regulatory Guide 8.9 (Reference 18).

Provisions for personnel decontamination are contained in the Security and Health Physics Building. Contamination of equipment or personnel is not expected to occur under normal conditions of operation. In accordance with the PFSLLC's policy of preventing generation of liquid radioactive waste, any necessary decontamination of equipment and personnel will be conducted using methods that produce only solid radioactive waste. Decontamination methods would typically include wiping the contaminated item with rags or paper wipes. Drain sumps are provided in the cask load/unload bay of the Canister Transfer Building which catch and collect water that drips from shipping casks (e.g. from melting snow) onto the floor. Water collected in the cask load/unload bay drain sumps is sampled and analyzed to verify it is not contaminated prior to its release. In the event contaminated water is detected, it will be collected in a suitable container, solidified by the addition of an agent such as cement or "Aquaset" so that it qualifies as solid waste, staged in the LLW holding cell while awaiting shipment offsite, and transported to a LLW disposal facility, in accordance with Radiation Protection procedures.

No process or effluent monitors are necessary because of the design of the PFSF storage system, in which spent fuel assemblies are stored in welded canisters. During routine storage operations at the PFSF, the only radiological instrumentation in use in the storage area will be the TLDs, as described in Section 7.3.5. Routine radiological surveys will use instruments that are controlled by the Radiation Protection Program and governed by existing procedures. Calibration procedures for radiological instrumentation will be established and applied to instruments used at the PFSF.

7.5.3 Procedures

Radiation protection requirements for all radiological work at the PFSF will be governed by radiation protection procedures. Radiation protection practices for cask loading and unloading operations, canister transfer, canister storage, and monitoring will also be based on these procedures, as well as on anticipated conditions when the task is to be performed. These procedures include, but are not limited to, the following:

- Procedure for performing badging functions for access authorization to the RA.
- Procedure for issuing personnel dosimetry, and monitoring, recording, and tracking individual exposures.
- Procedure for performing radiological safety training and refresher training.
- Procedure for performing ALARA reviews of plant procedures and monitoring of operations.
- Procedure for determining radiation doses on a periodic basis at RA and OCA boundaries using TLDs.
- Procedure for issuing, revising, and terminating radiation work permits and standing radiation work permits.
- Procedure for roping off, barricading, and posting radiation control zones.
- Procedure for decontaminating personnel, equipment, and areas.
- Procedure for performing radiation surveys.
- Procedure for smear swab sampling, counting, and calculation.
- Procedure for calibrating detection, monitoring, and dosimetry instruments.
- Procedure for quantifying airborne radioactivity.

- Procedure for maintaining records of the radiation protection program, including audits and other reviews of program content and implementation; radiation surveys; instrument calibrations; individual monitoring results; and records required for decommissioning.

Implementation of the Radiation Protection Program procedures ensures that occupational doses are below the limits required by 10 CFR 20.1201 and are ALARA both in the Canister Transfer Building as well as other parts of the facility. Area radiation montitors in the Canister Transfer Building have audible alarms and warn operating personnel of abnormal radiation levels. While area radiation monitors are not installed in the RA, measures are in place to ensure that personnel in the RA do not exceed dose limits. As discussed in Section 7.5.2, access to the RA is controlled through a single access point in the Security and Health Physics Building where personal dosimetry is issued to individuals entering the RA. Periodic radiation surveys will be conducted of areas inside the RA and maps will be generated showing the radiation levels in all areas. Radiation work permits (RWPs) will be completed by qualified radiation protection personnel prior to any entry and will identify normal and unusual radiation readings. Workers will be required to read, understand and sign that they are aware of the conditions or unknowns. Personnel will be trained to use the appropriate radiation detection instruments or will be required to have a qualified radiation protection technician with them at all time while in the areas. Training will include responses to unusual readings and off-scale conditions. The Radiation Protection program will provide for the immediate reading of any individual's TLD if an unusual reading or off-scale condition occurs.

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7.6 ESTIMATED OFFSITE COLLECTIVE DOSE ASSESSMENT

Figure 1.1-2 shows the PFSF OCA fence, which serves as the site boundary. Areas at and beyond the OCA fence are considered to be offsite. A maximum dose rate of 9.50 E-4 mrem/hr was calculated (Section 7.3.3.5) at the OCA boundary fence 1,969 ft (600 meters) from the RA fence at its closest points of approach. This dose rate is comprised of direct and scattered gamma and neutron radiation emanating from 4,000 storage casks and is based on the assumption that all 4,000 casks contain typical fuel expected to be received at the PFSF with 35 GWd/MTU burnup and 20-year cooling time. Operations inside the Canister Transfer Building would not contribute significantly to dose rates at the OCA fence as a result of shielding provided by the Canister Transfer Building walls and 500 meter minimum distance from the Canister Transfer Building to the OCA fence. Based on the dose rate calculated from the storage casks (9.50 E-4 mrem/hr for the HI-STORM storage casks), a hypothetical individual conservatively assumed to spend 2,000 hours a year at the OCA fence would receive a dose of 1.9 mrem, which is below the 25 mrem annual dose limit of 10 CFR 72.104.

The nearest residence is located approximately 2 miles east-southeast of the PFSF. As discussed in Section 7.3.3.5, a total dose rate of 3.89 E-6 mrem/hr (HI-STORM casks containing relatively hot fuel represented by PWR fuel having 40 GWd/MTU burnup and 10-year cooling time) is estimated at about 2 miles from the fully loaded ISFSI array, taking no credit for intervening shielding from berms, natural terrain, or buildings at the PFSF. Assuming full-time occupancy (8,760 hrs/yr), this equates to an annual dose of about 0.034 mrem. This dose is far less than the 25 mrem to any real individual outside the controlled area criteria of 10 CFR 72.104.

7.6.1 Effluent and Environmental Monitoring Program

10 CFR 72.126(c) requires the means to measure effluents. Since there are no radioactive liquid or gaseous waste effluents released from the PFSF during transfer and storage operations, this criterion is not applicable to the PFSF.

The storage system is a passive design with the spent fuel stored dry within welded canisters. No handling of individual fuel assemblies is planned at the PFSF. Therefore, a radioactive effluent monitoring system is not needed and routine monitoring for effluents is not performed.

Solid low level radioactive wastes will be temporarily stored in the LLW holding cell while awaiting shipment to a LLW disposal facility, as discussed in Section 6.4. The LLW holding cell will be regularly surveyed and inventoried, including inspection of the materials stored to evaluate the status of materials and controls (e.g. physical condition of containers, access control, posting). Radiation protection procedures govern the packaging, storage, surveying, inventorying, and monitoring of solid LLW.

The PFSF spent fuel storage operations will emit radiation that will be monitored in the environment with TLDs that will be located along the perimeter of the RA and along the OCA boundary fence.

7.6.2 Analysis of Multiple Contributions

Evaluation of incremental collective doses resulting from other nearby nuclear facilities in addition to the ISFSI is required per 10 CFR 72.122(e). This is not applicable to the PFSF since there are no other nuclear facilities located within a 5-mile radius of the PFSF. The closest nuclear facility is the Envirocare low-level radioactive and mixed waste disposal facility, which is about 25 miles northwest of the PFSF.

7.6.3 Estimated Dose Equivalents From Effluents

The canisters are high integrity vessels sealed by welding. The confinement design of the HI-STORM canister is discussed in Section 4.2.1.5.5, and Chapter 7 of the HI-STORM SAR. Holtec performed an evaluation of the potential for leakage from a HI-STORM canister under normal conditions of storage in Reference 25. The object of Holtec's study was to determine whether it is credible for a HI-STORM canister to develop a leak while it is stored in a storage overpack for a period of up to forty years.

As noted in Reference 25, Holtec has designed the HI-STORM canister to ASME Section III, Class 1, which is the highest category available and the same category to which critical nuclear components, such as the reactor vessel, are engineered. To provide for additional margin in the ability of the canister to maintain absolute leak tightness, the wall thicknesses of the canister enclosure vessel were set to be much greater than those required by the ASME Code. For example, while the Code would have called for an approximately 2.25 inch thick top cover, the actual cover thickness used in the HI-STORM canister varies from 9.5 inch (PWR canisters) to 10 inches (BWR canisters). Likewise, the shell is over 100% thicker than required by the ASME Code. Further, as required by Class I of the ASME Nuclear Code, the material of the canister enclosure vessel is subjected to volumetric examinations to check for internal flaws and the welds are subjected to multiple surface non-destructive examinations to ensure that any welding flaw buried within the weld mass will be minimal and so small that it will remain unconditionally stable under normal storage conditions.

Holtec's analysis (Reference 25) was based on classical fracture mechanics and determined that, even if the largest possible material non-homogeneity is postulated to exist in the canister enclosure vessel, it is not possible for a leak path to develop from the inside of the vessel to the outside. The minimum factor-of-safety against flaw propagation implying through boundary leakage was calculated to be 4.25. This factor of safety translates into a virtually unbreachable boundary in normal storage because the material strength (yield strength for example) would have to be 1/20 of its standard ASME Code required strength for the canister, to reduce the fracture toughness to a value that reduces the safety factor to 1.0. Steel mills do not produce austenitic stainless steel materials to any user, nuclear or commercial, which have such reduced yield strengths. Reference 25 indicates that in procuring the material for the HI-STORM canisters, Holtec employs the highest standards of quality assurance, which have been reviewed and approved by the NRC. The manufacturer of the canisters is required to hold the ASME Code stamp for Class 1 nuclear components.

In view of the design margins engineered into the canister enclosure vessel, the stringent quality control measures implemented in its manufacturing and use of proven stainless steel alloy materials, Holtec concluded (Reference 25) that a through-wall leak from a HI-STORM canister is not a credible event. Since canisters will not leak under normal conditions of storage, there will be no doses attributable to effluents from canisters. The PFSF does not generate any gaseous or liquid effluents during normal operating conditions and there will be no doses attributable to effluents in the areas surrounding the PFSF.

7.6.4 Liquid Release

There are no radioactive liquid effluents generated at the PFSF. As discussed in Section 7.5.2, any water collected in the Canister Transfer Building shipping cask load/unload bay drain sumps from potential moisture gathered on the outer surfaces of shipping casks during transport is sampled and analyzed to verify it is not radioactive prior to its release. In the event contaminated water is detected, it will be collected in a suitable container, solidified so that it qualifies as solid waste, staged in the LLW holding cell while awaiting shipment offsite, and transported to a LLW disposal facility, in accordance with Radiation Protection procedures.

7.7 REFERENCES

1. 10 CFR 72, Licensing Requirements for the Independent Storage of Spent Nuclear Fuel and High Level Radioactive Waste.

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

3. 10 CFR 19, Notices, Instructions and Reports to Workers: Inspection and Investigations.

4. Regulatory Guide 8.8, Information Relevant to Ensuring That Occupational Radiation Exposures at Nuclear Power Stations Will Be As Low As Is Reasonably Achievable, U.S. NRC, Revision 3, June 1978.

5. Regulatory Guide 8.10, Operating Philosophy for Maintaining Occupational Radiation Exposures As Low As Is Reasonably Achievable, U.S. NRC, Revision 1-R, May 1977.

6. Final Safety Analysis Report for the Holtec International Storage and Transfer Operation Reinforced Module Cask System (HI-STORM 100 Cask System), Holtec Report HI-2002444, NRC Docket No. 72-1014, Revision 0, July 2000.

7. (deleted)

8. DOE/RW-0184-R1, Characteristics of Potential Repository Wastes, Office of Civilian Radioactive Waste Management, U.S. Department of Energy, July 1992.

9. PNL-6906, Spent Fuel Assembly Hardware: Characterization and 10 CFR 61 Classification for Waste Disposal, Pacific Northwest Laboratory, June 1989.

10. ORNL/TM-9591/V1&R1, Physical and Decay Characteristics of Commercial LWR Spent Fuel, Oak Ridge National Laboratory, January 1996.

11. ORNL/TM-6051, Revised Uranium-Plutonium Cycle PWR and BWR Models for the ORIGEN Computer Code, Oak Ridge National Laboratory, September 1978.

12. (deleted)

13. Holtec Report HI-971645, Revision 1, Radiation Shielding Analysis for the Private Fuel Storage Facility, Holtec International, March 2000.

14. (deleted)

15. Regulatory Guide 8.4, Direct-Reading and Indirect-Reading Pocket Dosimeters, U.S. NRC, February 1973.

16. NUREG-0041, Manual of Respiratory Protection Against Airborne Radioactivity Materials, October 1976.

17. Regulatory Guide 8.26, Application of Bioassay for Fission and Activation Products, U.S. NRC, September 1980.

18. Regulatory Guide 8.9, Acceptable Concepts, Models, Equations and Assumptions for a Bioassay Program, U.S. NRC, September 1973.

19. U.S. Department of Energy, Energy Information Administration's Service Report entitled "Spent Nuclear Fuel Discharges from U.S. Reactors - 1994", published in February 1996.

20. PFSF Calculation No. 05996.02-UR-6, Calculational Basis for PFSF SAR Tables 7.4-1 and 7.4-2, Estimated Personnel Exposures for Canister Transfer Operations, Revision 2, Stone & Webster.

21. PFSF Calculation No. 05996.02-UR-5, Dose Rate Estimates from Storage Cask Inlet Duct Clearing Operations, Revision 1, Stone & Webster.

22. DOE/RW-0184, Characteristics of Spent Fuel, High Level Waste, and Other Radioactive Wastes Which May Require Long-Term Isolation, U.S. Department of Energy, December 1987.

23. PFSF Calculation No. 05996.02-UR(D)-12, Dose Rates From the 4000 Storage Cask PFSF Array Representative of PFSF Typical Spent Fuel, Assumed to be PWR Fuel Having 35 GWd/MTU Burnup and 20 Year Cooling Time.

24. Topical Safety Analysis Report for the Holtec International Storage, Transport, and Repository Cask System (HI-STAR 100 Cask System), Holtec Report HI-951251, Docket 71-9261, Revision 9, April 2000.

25. Holtec Report HI-2002424, Revision 0, A Deterministic Evaluation of Potential for Leakage from a HI-STAR/HI-STORM Multi-Purpose Canister, May 2000.

TABLE 7.3-1

MAXIMUM DOSE RATES ON CONTACT AND AT ONE METER FROM A HI-STORM STORAGE CASK (mrem/hr)

Detector Location (see Figure 7.3-1)	PWR 45 GWd/MTU 5-year cool	BWR 45 GWd/MTU 5-year cool
1) Side, contact	γ = 31.70 n = 1.88 Tot = 33.58	γ = 31.96 n = 2.96 Tot = 34.92
2) Side, 1 meter	γ = 16.64 n = 0.78 Tot = 17.42	γ = 16.23 n = 1.07 Tot = 17.30
3) Top, contact	γ = 1.31 n = 3.60 Tot = 4.91	γ = 1.02 n = 3.29 Tot = 4.31
4) Top, 1 meter	γ = 0.60 n = 1.10 Tot = 1.70	γ = 0.53 n = 0.82 Tot = 1.35
5) Top vent, contact	γ = 7.21 n = 1.38 Tot = 8.59	γ = 6.11 n = 1.12 Tot = 7.23
6) Bottom vent, contact	γ = 10.75 n = 2.76 Tot = 13.51	γ = 10.96 n = 3.56 Tot = 14.52

TABLE 7.3-2

(deleted)

TABLE 7.3-3

MAXIMUM DOSE RATES ASSOCIATED WITH A 125-TON HI-TRAC TRANSFER CASK (mrem/hr)

Detector Location (see Figure 7.3-2)	PWR 45 GWd/MTU 9-year cool
1) Side, contact	γ = 65.66 n = 50.00 Tot = 115.66
2) Side, 1 meter	γ = 23.75 n = 18.03 Tot = 41.78
3) Top, contact	γ = 164.51 n = 158.74 Tot = 323.25
4) Top, side contact	γ = 24.48 n = 138.83 Tot = 163.31
5) Bottom, center, contact	γ = 197.98 n = 87.90 Tot = 285.88
6) Bottom, side, contact	γ = 79.25 n = 83.52 Tot = 162.77

TABLE 7.3-4

(deleted)

TABLE 7.3-5

DOSE RATES VERSUS DISTANCE FOR A SINGLE HI-STORM STORAGE CASK *

(mrem/hr)

Distance from Cask Side	Cask Side Gamma Dose Rate	Cask Side Neutron Dose Rate	Cask Top Gamma Dose Rate	Cask Top Neutron Dose Rate	Total Dose Rate
150 ft (46 m – security fence)	2.10 E-2	1.65 E-3	2.39 E-5	1.88 E-4	2.29 E-2
328 ft (100 m)	3.46 E-3	2.95 E-4	7.32 E-6	4.63 E-5	3.81 E-3
656 ft (200 m)	5.33 E-4	4.97 E-5	1.53 E-6	7.67 E-6	5.92 E-4
984 ft (300 m)	1.40 E-4	1.44 E-5	4.16 E-7	1.77 E-6	1.57 E-4
1476 ft (450 m)	2.76 E-5	2.90 E-6	7.41 E-8	2.68 E-7	3.08 E-5
1969 ft (600 m - OCA fence)	6.69 E-6	9.28 E-7	1.79 E-8	4.85 E-8	7.68 E-6

* Cask assumed to contain 40 GWd/MTU, 10-year cooled PWR fuel.

TABLE 7.3-6

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TABLE 7.3-7

DOSE RATES AT LOCATIONS OF INTEREST FROM THE PFSF ARRAY OF 4,000 ASSUMED HI-STORM STORAGE CASKS *

(mrem/hr)

Distance and Direction to Detector from Nearest Storage Pad	Dose Rate from Sides of Casks	Dose Rate from Tops of Casks	Total Dose Rate
150 ft north (security fence)**	1.64 E0	4.48 E-2	1.68
150 ft east or west (security fence)	1.32 E0	4.43 E-2	1.37
2,119 ft north (OCA boundary)***	2.75 E-3	4.91 E-5	2.80 E-3
2,119 ft west (OCA boundary)***	1.65 E-3	4.34 E-5	1.69 E-3

* Casks assumed to contain 40 GWd/MTU, 10-year cooled PWR fuel.

** The security (Restricted Area) fence is 150 ft from the nearest storage pad in the north, east, and west directions. It is further (265 ft) from storage pads in the south direction. Therefore, the dose rate at the south security fence will be less than that at the north security fence.

*** The distance from the nearest pads to the north and west Owner Controlled Area (OCA) boundary fence is 2,119 ft Distances to the OCA boundary fence are further from the storage pads in the south (≈2,300 ft) and east (≈2,260 ft) directions, and dose rates would be lower at these sections of the OCA boundary fence.

TABLE 7.3-8

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TABLE 7.4-1
(Page 1 of 4)
ESTIMATED PERSONNEL EXPOSURES FOR HI-STORM CANISTER TRANSFER OPERATIONS

Operation	No. of Personnel[1]	Task Duration[2] (hours)	Time in Dose Area[3] (hours)	Dose Rate in Area[4] (mrem/hr)	Dose[5] (person-mrem)
1. Receive and inspect shipment, and measure dose rates.	2 Ops 1 HP	0.5	0.5 0.5	4 4	4 2
2. Move shipment into Canister Transfer Building.	3 Ops 1 HP	0.5	0.5 0.5	0 0	0 0
3. Remove personnel barrier, measure dose rates, and perform contamination survey.	2 Ops 1 HP	1.6	1.0 1.6	6 8	12 12.8
4. Remove impact limiters and tiedowns.	2 Ops 1 HP	1.5	1.5 1.5	4 1	12 1.5
5. Attach lifting yoke to crane and HI-STAR shipping cask. Upright HI-STAR cask and move to transfer cell.	2 Ops 1 HP	1.0	0.5 1.0	4 1	4 1
6. Sample enclosed cask gas and vent.	1 Op 1 HP	0.5	0.5 0.5	2 1	1 0.5
7. Remove HI-STAR closure plate (lid) bolts.	2 Ops 1 HP	1.0	1.0 1.0	2 1	4 1
8. Remove HI-STAR closure plate (lid).	2 Ops 1 HP	0.2	0.2 0.2	2 1	0.8 0.2
9. Prep HI-STAR to mate with HI-TRAC transfer cask.	2 Ops 1 HP	0.2	0.2 0.2	2 1	0.8 0.2
10. Install canister lift cleats and attach slings.	2 Ops 1 HP	1.0	1.0 1.0	40 / 21 1	80 / 42 1

TABLE 7.4-1 (Page 2 of 4)

Operation	No. of Personnel [1]	Task Duration[2] (hours)	Time in Dose Area[3] (hours)	Dose Rate in Area[4] (mrem/hr)	Dose[5] (person-mrem)
11. Attach lifting yoke to crane and HI-TRAC.	2 Ops	0.5	0.5	2	2
	1 HP		0.5	1	0.5
12. Mount HI-TRAC on top of HI-STAR.	2 Ops	0.5	0.5	2	2
	1 HP		0.5	1	0.5
13. Open HI-TRAC transfer cask doors.	2 Ops	0.2	0.2	7	2.8
	1 HP		0.2	1	0.2
14. Attach slings to canister downloader hoist and raise canister.	2 Ops	0.5	0.5	14	14
	1 HP		0.5	1	0.5
15. Close HI-TRAC doors and install pins.	2 Ops	0.2	0.2	7	2.8
	1 HP		0.2	1	0.2
16. Lower canister onto HI-TRAC doors.	2 Ops	0.2	0.2	1	0.4
	1 HP		0.2	1	0.2
17. Prep HI-STORM storage cask to mate with HI-TRAC transfer cask (including installation of HI-STORM shielding inserts).	2 Ops	0.2	0.2	4	1.6
	1 HP		0.2	1	0.2
18. Move HI-TRAC from HI-STAR to HI-STORM.	2 Ops	0.7	0.7	1	1.4
	1 HP		0.7	1	0.7
19. Raise canister and open HI-TRAC doors.	2 Ops	0.5	0.2	7	2.8
	1 HP		0.5	1	0.5
20. Lower canister into HI-STORM storage cask.	2 Ops	0.5	0.5	1	1
	1 HP		0.5	1	0.5
21. Disconnect lifting slings.	2 Ops	0.2	0.2	14	5.6
	1 HP		0.2	1	0.2

TABLE 7.4-1 (Page 3 of 4)

Operation	No. of Personnel [1]	Task Duration[2] (hours)	Time in Dose Area[3] (hours)	Dose Rate in Area[4] (mrem/hr)	Dose[5] (person-mrem)
22. Close transfer cask doors.	2 Ops	0.2	0.2	7	2.8
	1 HP		0.2	1	0.2
23. Remove HI-TRAC from HI-STORM.	2 Ops	0.5	0.5	1	1
	1 HP		0.5	1	0.5
24. Remove canister lift cleats and HI-STORM shield inserts.	2 Ops	0.5	0.5	40 / 21	40 / 21
	1 HP		0.5	1	0.5
25. Install HI-STORM lid and lid bolts.	2 Ops	1.0	1.0	1	2
	1 HP		1.0	1	1
26. Perform dose survey and install HI-STORM lifting eyes.	2 Ops	0.5	0.5	1	1
	1 HP		0.5	1	0.5
27. Drive cask transporter in transfer cell.	1 Op	0.3	0.3	1	0.3
	1 HP		0.3	1	0.3
28. Connect HI-STORM to cask transporter.	2 Ops	0.5	0.5	1	1
	1 HP		0.5	1	0.5
29. Raise HI-STORM storage cask.	2 Ops	0.2	0.2	1	0.4
	1 HP		0.2	1	0.2
30. Transport HI-STORM cask to storage pad.	2 Ops	2.0	2.0	1	4
	1 HP		2.0	1	2
31. Position and lower HI-STORM cask on pad.	2 Ops	0.5	0.5	9	9
	1 HP		0.5	7	3.5
32. Disconnect HI-STORM cask from transporter and remove cask lifting eyes.	2 Ops	1.0	1.0	9	18
	1 HP		1.0	7	7

TABLE 7.4-1 (Page 4 of 4)

Operation	No. of Personnel[1]	Task Duration[2] (hours)	Time in Dose Area[3] (hours)	Dose Rate in Area[4] (mrem/hr)	Dose[5] (person-mrem)
33. Connect cask temperature instrumentation and vent duct screens.	2 Ops 1 HP	0.5	0.5 0.5	9 7	9 3.5
34. Perform cask operability tests.	1 Op 1 HP	48	1.0 1.0	9 7	9 7
TOTAL					303.6 / 246.6 [5]

1. Number of personnel typically includes 2 to 4 operators and 1 HP technician.
2. Task duration includes total estimated time required to perform task.
3. Time in dose area includes only that time personnel are in a significant dose field.
4. The values in this column represent estimated average dose rates in the area where personnel will be working to perform the associated task. For operations where it is considered that temporary shielding will be effective in keeping dose rates ALARA, two values are presented (e.g., 50 / 5). The first value (50 mrem/hr) is the projected dose rate assuming no credit for temporary shielding. The second value (5 mrem/hr) takes credit for radiation attenuation by the use of temporary shielding.
5. Doses are calculated for times in dose fields without temporary shielding and with temporary shielding, such as 50 / 5, where the first value (50 mrem) is calculated based on the time spent in an area with the dose rate in the preceding column without temporary shielding, and the second value (5 mrem) is calculated based on the dose rate in the preceding column that takes credit for temporary shielding.

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