CHAPTER 5

OPERATION SYSTEMS

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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) and the TranStor Storage Cask System (Reference 2). 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 systems used to transport fuel to the PFSF are the HI-STAR 100 Shipping Cask System (Reference 3) and TranStor Shipping Cask System (Reference 4). 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 canister is then backfilled with inert helium gas and the drain and fill ports are welded closed, thereby sealing the canister. The outer surfaces of the transfer cask are then checked for surface contamination and decontaminated, if necessary.

The sealed canister is transferred from the transfer cask to a shipping cask, and the shipping cask closure bolted in place. The shipping cask is then loaded onto the shipment vehicle, fitted with impact limiters to protect the shipping cask during transportation, and shipped to the PFSF.

5.1.2 Operations Between the Originating Nuclear Power Plant and the PFSF

The shipping cask, containing the sealed spent fuel canister, is shipped by rail to the PFSF. During transportation, the shipping cask provides a complete confinement barrier for the canister that is capable of withstanding any accident that could occur. The shipping cask is fitted with impact limiting devices for additional protection during transit. The offsite transportation system is licensed in accordance with the requirements of 10 CFR 71, "Packaging and Transportation of Radioactive Material", and complies with the requirements of 49 CFR 171, “General Information, Regulations, and Definitions”, 49 CFR 172, “Hazardous Materials Tables and Hazardous Materials Communications Regulations”, 49 CFR 173, “Shippers - General Requirements for Shipments and Packages”, 49 CFR 174, “Carriage by Rail”, and 49 CFR 177, “Carriage by Public Highway”.

5.1.3 Operations Between the Railroad Mainline and the PFSF

The PFSF is located approximately 24 miles south of the rail mainline and currently does not have rail access. The shipping casks are shipped to the PFSF by rail to Timpie and then by one of two alternatives. The first alternative utilizes intermodal transfer at Timpie, where the loaded shipping casks are transferred from the rail car to a heavy haul tractor/trailer for transport via highway the remaining 24 miles to the PFSF. At the intermodal transfer point is a rail siding, single-failure-proof gantry crane, and weather enclosure over the crane. The second alternative utilizes a new rail spur to enable the shipping casks to remain on the rail car and be transported by rail directly to the PFSF.

5.1.4 Operations at the PFSF

5.1.4.1 Receipt and Inspection of Incoming Shipping Cask and Canister

During spent fuel shipment, the canister is contained within the shipping cask, which is mounted horizontally on a rail car or heavy haul trailer. Impact limiters are mounted on either end of the shipping cask and a personnel barrier cover is located over the shipping cask between the impact limiters.

When the shipping cask arrives at the PFSF, the shipping cask, impact limiters, and shipping cradle are visually inspected for damage prior to entry into the Restricted Area (RA). After initial receipt approval the shipment is moved into the security vehicle trap for inspection by security personnel to ensure no unauthorized devices enter the RA. When security clearance is complete, the shipment proceeds into the RA and into the Canister Transfer Building where the personnel barrier is removed and the shipping cask is surveyed for dose rates and contamination levels

5.1.4.2 Transfer of Canister from Shipping Cask to Storage Cask

Transfer of the canister containing spent fuel from the shipping cask to the storage cask takes place within the Canister Transfer Building. After the receipt inspection, the overhead bridge crane is used to remove the impact limiters. The shipping cask lifting yoke is attached to the crane and hooked to the shipping cask, which is uprighted on 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. If the transfer operation is with the TranStor system, support struts are attached between the TranStor transfer cask and the transfer cell walls, because the crane must later be detached from the TranStor transfer cask to lift the canister. The HI-STORM system uses the canister downloader, which is attached to the top of the HI-TRAC transfer cask, to lift and lower the canister and does not require the support struts since the crane is never disconnected from the transfer cask. 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 for the TranStor system only. The transfer cask is lifted from the shipping cask by the crane and placed on top of the concrete storage cask. Support struts are again attached between the transfer cask and transfer cell walls for TranStor system only. 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 concrete 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 periodic basis in accordance with the Technical Specifications. An overall site observation surveillance is also performed on a periodic basis to detect any cask damage or accumulation of site debris. Surveillance requirements are discussed in Chapter 10.

Radiation doses emitted from the storage casks are measured by thermoluminecent 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 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 visitor persons, 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 concrete storage casks to the shipping cask. Once loaded in the 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 and on Figures 5.1-3 and 5-1-4 for the TranStor storage system.

A flow diagram showing the sequence of operations required to remove the storage casks from the PFSF and ship them offsite is shown on Figure in 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 and Table 5.1-2 for the TranStor system. These tables are 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 or TranStor Storage Systems since the systems are passive and rely 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. Portable radiation monitors are used to measure radiation levels of casks following canister transfer. PFSF personnel are equipped with personnel dosimeters wherever 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 perform 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 and TranStor shipping SARs

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, the transfer cask, and the TranStor canister. 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 and trunnions.

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, HI-STORM canister lifting cleats, and TranStor canister hoist rings.

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.

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-STORM and TranStor SARs.

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

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. Hydraulic cylinders are used to open the TranStor transfer cask doors.

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. On the TranStor transfer cask, the top cover of the transfer cask is designed to stop the canister and prevent the crane from inadvertently lifting the canister up and out of the transfer cask while being raised.

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

The safety features of the transfer casks are described in greater detail in the HI-STORM and TranStor SARs.

5.2.2 Spent Fuel Canister Storage

Spent fuel storage consists of the HI-STORM and the TranStor storage systems, which includes spent fuel canisters placed in the concrete storage casks located on the storage pads. The storage systems are a passive design and require no support systems for operation. The storage systems perform their functions under normal conditions as discussed in SAR Chapter 4 and off-normal and accident level conditions as discussed in SAR Chapter 8. Limits of operation associated with various normal and off-normal conditions are contained in SAR Chapter 10. Surveillance requirements are also contained in SAR Chapter 10.

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 concrete 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 and TranStor SARs.

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5.3 OTHER OPERATING SYSTEMS

The storage casks are passive and require no other operating systems for safe storage of the spent fuel once they are placed into storage. All the PFSF operating systems are described in Sections 5.1 and 5.2.

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5.4 OPERATION SUPPORT SYSTEMS

5.4.1 Instrumentation and Control Systems

Regulation 10 CFR 72.122(i) requires that instrumentation and control systems be provided to monitor systems that are classified as Important to Safety. The operation of the PFSF is passive and self-contained and therefore does not require control systems to ensure the safe operation of the system. However, temperatures of the storage casks are monitored to provide a means for assessing thermal performance of the storage casks. The temperature monitors are equipped with data recorders and alarms located in the Security and Health Physics Building. The temperature monitors are not required for safety and therefore are not subject to important to safety criteria.

Radiation monitoring is provided to ensure doses remain ALARA and is discussed in Section 7.3.4. Radiation monitoring is not required to support systems that are classified as Important to Safety.

In the event of an earthquake, the PFSF will contact the National Earthquake Information Center, Golden, Colorado, to acquire seismic data for the PFSF.

No other instrumentation or control systems are necessary or are utilized. Therefore, the requirements of 10 CFR 72.122(i) are satisfied.

5.4.2 System and Component Spares

Spare temperature monitoring devices are maintained at the site. However, these devices are not required to maintain safe conditions at the PFSF. No other instrumentation spares are required.

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5.5 CONTROL ROOM AND CONTROL AREAS

Regulation 10 CFR 72.122(j) requires the control room or control area to be designed to ensure that the PFSF is safety operated, monitored, and controlled for off-normal or accident conditions. This requirement is not applicable to the PFSF because the spent fuel storage system is a passive system and requires no control room to ensure safe operation at the PFSF.

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5.6 ANALYTICAL SAMPLING

No sampling is required for the safe operation of the PFSF or to ensure that operations are within prescribed limits. Sampling of the gas inside the shipping cask is performed prior to venting and opening the cask in the Canister Transfer Building. Evaluation of the gas sample determines if the gas can be released to the atmosphere or if it must be filtered and the appropriate radiological protection needed when removing the shipping cask closure. Since the sampling is not required for nuclear safety of the facility, it is not subject to Important to Safety criteria.

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5.7 REFERENCES

1. Topical Safety Analysis Report for the Holtec International Storage and Transfer Operation Reinforced Module Cask System (HI-STORM 100 Cask System), Holtec Report HI-951312, Docket 72-1014, Revision 1, January 1997.

2. Safety Analysis Report for the TranStor Storage Cask System, SNC-96-72SAR, Sierra Nuclear Corporation, Docket 72-1023, Revision B, March 1997.

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 4, September 1996.

4. Safety Analysis Report for the TranStor Shipping Cask System, SNC-95-71SAR, Sierra Nuclear Corporation, Docket Number 71-9268, Revision 1, September 1996.

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 5.1-1
(Sheet 1 of 2)

ANTICIPATED TIME AND PERSONNEL REQUIREMENTS FOR HI-STORM CANISTER TRANSFER OPERATIONS

OPERATION	NO. OF PERSONNEL[1]	TASK DURATION (HOURS)
1. Receive and inspect shipment. Measure dose rates.	3	0.5
2. Move shipment into Canister Transfer Building.	4	0.5
3. Remove personnel barrier, measure cask dose rates, and perform contamination survey.	3	1.6
4. Remove impact limiters and tiedowns.	3	1.5
5. Attach lifting yoke to crane and HI-STAR shipping cask. Upright HI-STAR cask and move to transfer cell. Connect support struts.	3	1.0
6. Sample enclosed cask gas and vent.	2	0.5
7. Remove HI-STAR closure plate bolts.	3	1.0
8. Remove HI-STAR closure plate (lid).	3	0.2
9. Prep HI-STAR to mate with HI-TRAC transfer cask.	3	0.2
10. Install canister lift cleats and attach slings.	3	1.0
11. Attach lifting yoke to crane and HI-TRAC.	3	0.5
12. Mount HI-TRAC on top of HI-STAR.	3	0.5
13. Open HI-TRAC transfer cask doors.	3	0.2
14. Attach slings to canister downloader hoist and raise canister.	3	0.5
15. Close HI-TRAC doors and install pins.	3	0.2
16. Lower canister onto HI-TRAC doors.	3	0.2
17. Prep HI-STORM storage cask to mate with HI-TRAC transfer cask.	3	0.2
18. Move HI-TRAC from HI-STAR to HI-STORM.	3	0.7
19. Raise canister and open HI-TRAC doors.	3	0.5
20. Lower canister into HI-STORM storage cask.	3	0.5

TABLE 5.1-1
(Sheet 2 of 2)

ANTICIPATED TIME AND PERSONNEL REQUIREMENTS FOR HI-STORM CANISTER TRANSFER OPERATIONS

OPERATION	NO. OF PERSONNEL[1]	TASK DURATION (HOURS)
21. Disconnect lifting slings.	3	0.2
22. Close transfer cask doors.	3	0.2
23. Remove HI-TRAC from HI-STORM	3	0.5
24. Remove canister lift cleats.	3	0.5
25. Install HI-STORM lid and lid bolts.	3	1.0
26. Perform dose survey and install HI-STORM lifting eyes.	3	0.5
27. Drive cask transporter in transfer cell.	2	0.3
28. Connect HI-STORM to cask transporter.	3	0.5
29. Raise HI-STORM storage cask.	3	0.2
30. Transport HI-STORM cask to storage pad.	3	2.0
31. Position and lower HI-STORM cask on pad.	3	0.5
32. Disconnect HI-STORM cask from transporter and remove cask lifting eyes.	3	1.0
33. Connect cask temperature instrumentation.	3	0.5
34. Perform cask operability tests.	2	48
Total Hours	-	19.9[2]

Notes

1. Number of personnel typically includes 2 to 3 operators and 1 HP technician.
2. Total does not reflect 48 hour duration in Step 34, which is time required for cask temperature to reach equilibrium. Personnel time required to monitor temperatures during the equilibrium phase is minimal.

TABLE 5.1-2
(Sheet 1 of 2)

ANTICIPATED TIME AND PERSONNEL REQUIREMENTS FOR TRANSTOR CANISTER TRANSFER OPERATIONS

OPERATION	NO. OF PERSONNEL[1]	TASK DURATION (HOURS)
1. Receive and inspect shipment. Measure dose rates.	3	0.5
2. Move shipment into Canister Transfer Building.	4	0.5
3. Remove personnel barrier, measure cask dose rates, and perform contamination survey.	3	1.6
4. Remove impact limiters and tiedowns and install cask rotation trunnions.	3	1.5
5. Attach lifting yoke to crane and TranStor shipping cask. Upright shipping cask and move to transfer cell. Connect support struts.	3	1.0
6 Sample enclosed cask gas and vent.	2	0.5
7. Remove shipping cask closure lid bolts.	3	1.0
8. Remove shipping cask closure lid.	3	0.2
9. Prep shipping cask to mate with TranStor transfer cask.	3	0.2
10. Install canister lift eyes and attach slings.	3	1.0
11. Attach lifting yoke to crane and transfer cask.	3	0.5
12. Mount transfer cask on top of shipping cask, connect support struts, and disengage crane.	3	0.7
13. Open transfer cask doors.	3	0.2
14. Attach slings to crane and raise canister.	3	0.5
15. Close transfer cask doors and install pins.	3	0.2
16. Lower canister onto transfer cask doors and disconnect canister slings from crane hook.	3	0.2
17. Attach lifting yoke to crane hook and engage transfer cask. Disconnect support struts.	3	0.5
18. Move transfer cask from shipping cask to storage cask. Attach support struts to transfer cask and disengage crane.	3	1.0

TABLE 5.1-2
(Sheet 2 of 2)

ANTICIPATED TIME AND PERSONNEL REQUIREMENTS FOR TRANSTOR CANISTER TRANSFER OPERATIONS

OPERATION	NO. OF PERSONNEL[1]	TASK DURATION (HOURS)
19. Engage crane to canister, raise canister, and open transfer cask doors.	2	0.5
20. Lower canister into TranStor storage cask.	3	0.5
21. Disconnect lifting slings.	3	0.2
22. Close transfer cask doors.	2	0.2
23. Attach lifting yoke to crane and engage to transfer cask. Remove transfer cask from storage cask	3	0.8
24. Remove canister lifting eyes.	3	0.5
25. Install storage cask lid and lid bolts.	3	1.0
26. Perform dose survey and install storage cask lifting eyes.	3	0.5
27. Drive cask transporter in transfer cell.	2	0.3
28. Connect storage cask to cask transporter.	3	0.5
29. Raise storage cask.	3	0.2
30. Transport storage cask to storage pad.	3	2.0
31. Position and lower storage cask on pad.	3	0.5
32. Disconnect storage cask from transporter and remove storage cask lifting eyes.	3	1.0
33. Connect cask temperature instrumentation.	3	0.5
34. Perform cask operability tests.	2	48
Total Hours	-	21[2]

Notes

1. Number of personnel typically includes 2 to 3 operators and 1 HP technician.
2. Total does not reflect 48 hour duration in Step 34, which is time required for cask temperature to reach equilibrium. Personnel time required to monitor temperatures during the equilibrium phase is minimal.

SHIPMENT RECEIPT AND INSPECTION

1. Visually inspect the shipping cask, impact limiters, and cradle for any physical damage. Measure shipment dose rates. Verify security seal is in place.
2. Move the shipping cask, which is loaded on a heavy haul trailer or rail car into the Canister Transfer Building.
3. Remove the personnel barrier, measure shipping cask (HI-STAR) surface dose rates, and perform contamination surveys.

TRANSFER PREPARATION

4. Remove security seal, impact limiters, and shipment tiedowns.
5. Attach the HI-STAR lifting yoke to the overhead bridge crane. Engage the lifting yoke with the cask trunnions. Upright the HI-STAR shipping cask on the shipping cradle in the vertical position, raise cask from the transport vehicle, and move the cask into a canister transfer cell. Attach support struts between the shipping cask and transfer cell walls. Remove the removable shear ring segments from the HI-STAR shipping cask.
6. Remove cask vent port cover plate and attach backfill tool and sample bottle. Sample gas in the annulus and evaluate. Vent the cask annulus to atmosphere by removing the vent port seal plug if results from sample are acceptable. Vent the cask annulus through backfill tool HEPA filter if results from sample are not acceptable. Remove sample equipment.
7. Remove the HI-STAR cask closure plate (cask lid) bolts.
8. Remove the HI-STAR closure plate.
9. Install the transfer collar on HI-STAR shipping cask.
10. Install canister lift cleats on top of the canister. Attach lifting slings to the canister lift cleats.

CANISTER TRANSFER

11. Attach the transfer cask lifting yoke to the overhead bridge or semi-gantry crane and engage lifting yoke to the transfer cask (HI-TRAC) trunnions.
12. Mount the HI-TRAC transfer cask on top of the HI-STAR shipping cask.
13. Remove the HI-TRAC transfer cask shield door locking pins and open the shield doors. Install the radiation shielding (trim plates).
14. Attach hoist slings to canister downloader (canister lifting hoist) hook and lift the canister into the HI-TRAC transfer cask by extending the downloader.
15. Remove trim plates, close HI-TRAC shield doors, and install locking pins.
16. Lower the canister onto the HI-TRAC shield doors.
17. Install vent duct shield inserts in the HI-STORM upper vents and install alignment pins in the HI-STORM lifting eye holes.
18. Move the HI-TRAC transfer cask from on top of the HI-STAR shipping cask and mount on top of the HI-STORM storage cask.
19. Extend the canister downloader to the full position to raise the canister off of the transfer lid shield doors, remove the shield door locking pins and open the shield doors.
20. Install the radiation shielding (trim plates). Retract the canister downloader to lower the canister into the HI-STORM storage cask.
21. Install the radiation shielding (trim plates). Retract the canister downloader to lower the canister into the HI-STORM storage cask.
22. Remove the trim plates, close the shield doors, and install the door locking pins.
23. Remove the HI-TRAC transfer cask from the HI-STORM storage cask and place the HI-TRAC cask back into its storage area.

CASK STORAGE

24. Remove the canister lift cleats and lifting slings.
25. Remove the HI-STORM vent duct shield inserts and alignment pins. Install upper vent screens, HI-STORM lid, and lid bolts.
26. Perform a health physics survey for radiation doses and install the HI-STORM lifting eyes.
27. Open the transfer cell door and drive the cask transporter into the cell straddling the HI-STORM storage cask.
28. Attach the cask transporter lift hoist to the HI-STORM lifting eyes.
29. Raise the HI-STORM storage cask approximately 4 inches.
30. Transport the HI-STORM storage cask from the Canister Transfer Building to the appropriate storage pad with the cask transporter.
31. Position the HI-STORM storage cask in its designated storage location and lower the cask to the pad.
32. Disconnect the cask transporter lift hoist from the HI-STORM storage cask and remove the lifting eyes.
33. Connect the storage cask temperature monitoring instrumentation.
34. Perform natural convection cooling operability testing.

(Reference 1, Section 8.5)
(Note: The exact operational sequence is controlled by PFSF procedures.)

Figure 5.1-1

HI-STORM CANISTER TRANSFER OPERATIONAL SEQUENCE

PRIVATE FUEL STORAGE FACILITY
SAFETY ANALYSIS REPORT

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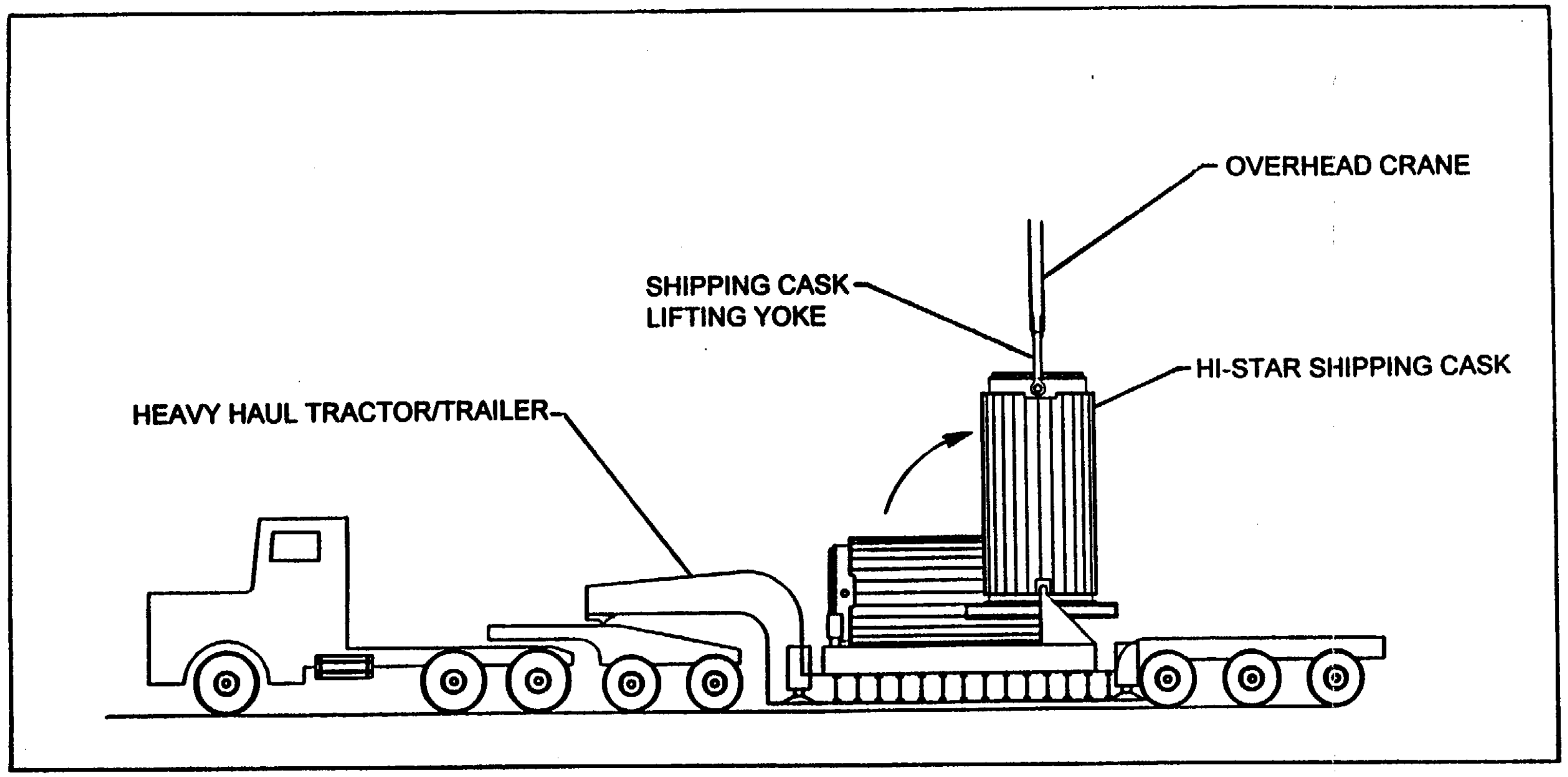
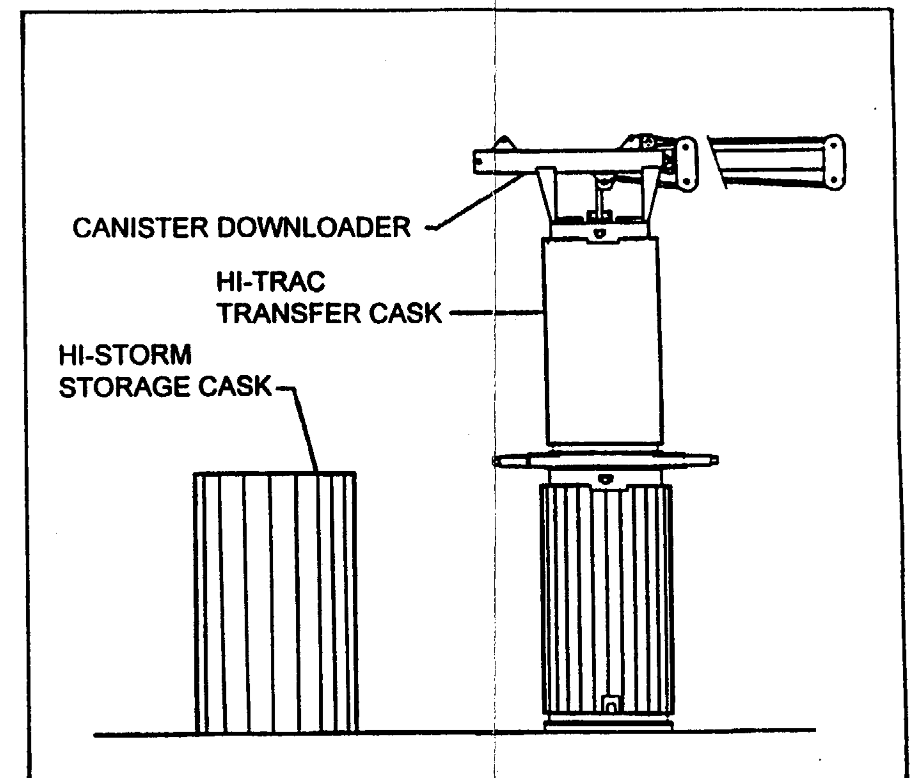
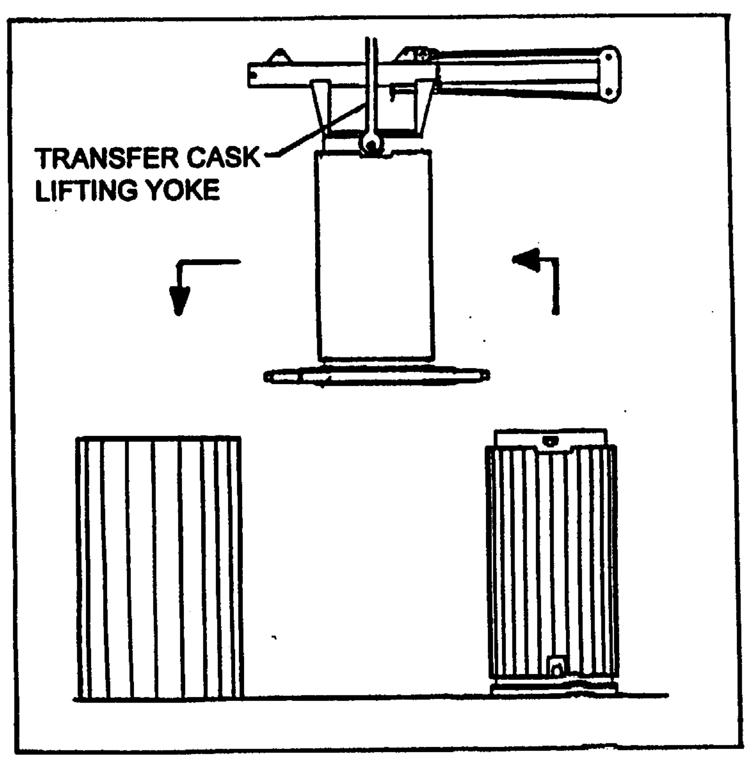


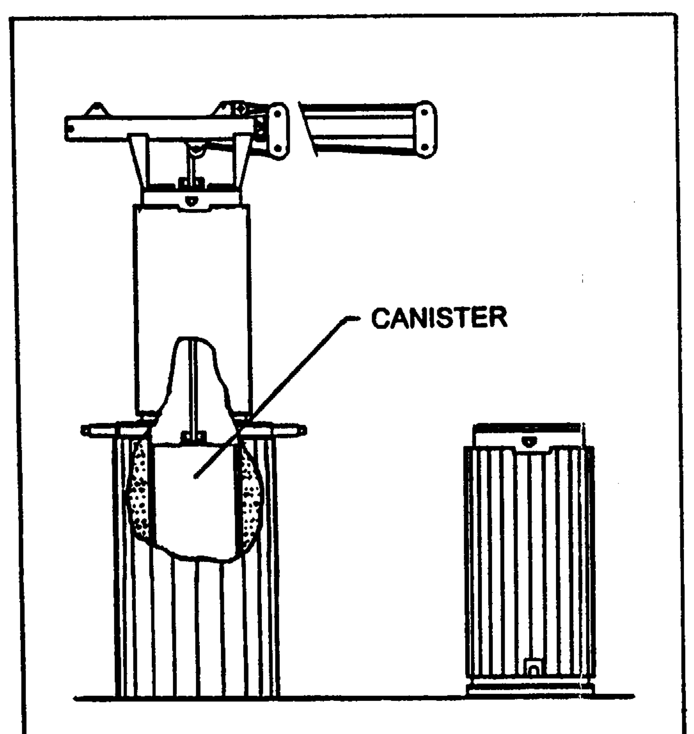
AFTER RECEIPT INSPECTION AND PLACEMENT INTO CANISTER TRANSFER BUILDING, UPRIGHT SHIPPING CASK, LIFT OFF TRAILER, AND MOVE TO TRANSFER CELL



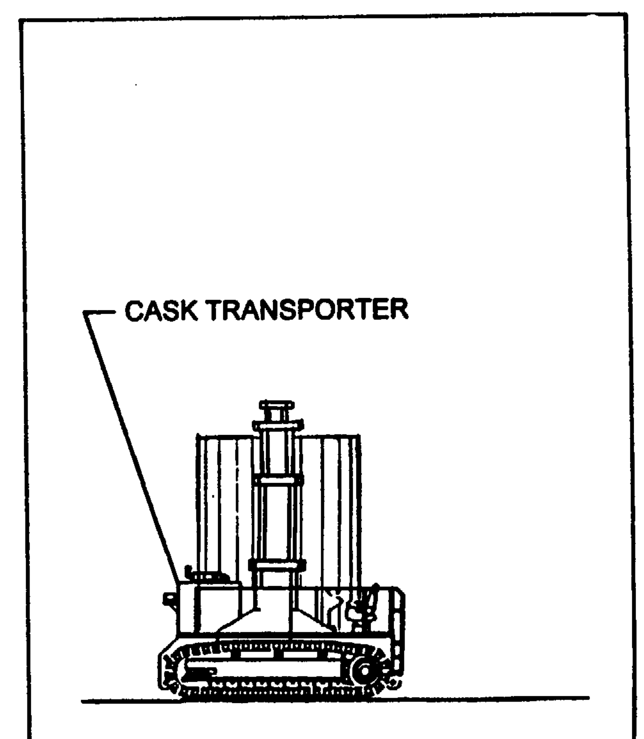
PLACE TRANSFER CASK ON SHIPPING CASK, OPEN SHIELD DOORS, AND RAISE CANISTER INTO TRANSFER CASK BY EXTENDING DOWNLOADER



MOVE TRANSFER CASK FROM TOP OF SHIPPING CASK TO TOP OF STORAGE CASK VIA CRANE



OPEN SHIELD DOORS AND LOWER CANISTER INTO STORAGE CASK BY RETRACTING DOWNLOADER. REMOVE TRANSFER CASK AND INSTALL STORAGE CASK LID



TRANSPORT STORAGE CASK TO STORAGE PAD VIA CASK TRANSPORTER

Figure 5.1-2

HI-STORM CANISTER TRANSFER OPERATION

PRIVATE FUEL STORAGE FACILITY
SAFETY ANALYSIS REPORT

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SHIPMENT RECEIPT AND INSPECTION

1. Visually inspect the TranStor shipping cask, impact limiters, and cradle for any physical damage. Measure dose rates. Verify security seal is in place.
2. Move the shipping cask, which is loaded on a heavy haul trailer or rail car into the Canister Transfer Building.
3. Remove the personnel barrier, measure TranStor shipping cask dose rates, and perform contamination surveys.

TRANSFER PREPARATION

4. Remove security seal, impact limiters, and shipment tiedowns. Bolt rotation trunnions to the TranStor shipping cask.
5. Attach the shipping cask lifting yoke to the overhead bridge crane. Engage the lifting yoke with the cask trunnions. Lift and rotate the shipping cask to the vertical position on the shipping cradle, raise the cask from the transport vehicle, and move the cask into a canister transfer cell. Attach support struts between the shipping cask and transfer cell walls.
6. Remove cask vent port cover plate and attach a pressure test manifold to the vent port to measure any pressure changes in the cask from shipment. Sample gas in the cavity and evaluate. Vent the cask cavity through the manifold to atmosphere if results from sample are acceptable. Vent the cask cavity through the manifold HEPA filter if results from sample are not acceptable. Disconnect pressure test manifold.
7. Remove the shipping cask closure bolts and install eyebolts.
8. Remove the closure lid.
9. Install cask adapter ring and alignment pins.
10. Install canister lifting eyes on top of the canister. Attach lifting slings to the canister lifting eyes.

CANISTER TRANSFER

11. Attach the transfer cask lifting yoke to the overhead bridge or semi-gantry crane and engage lifting yoke to transfer cask trunnions.
12. Mount the transfer cask on top of the TranStor shipping cask. Attach support struts between the transfer cask and transfer cell wall. Disengage the transfer cask lifting yoke from the crane and set aside.
13. Remove the door pins and open the transfer cask shield doors.
14. Engage the crane hook to the slings connected to the canister lifting eyes and raise the canister into the transfer cask.
15. Close the transfer cask shield doors and install the doors pins.
16. Lower the canister onto the transfer cask shield doors. Disconnect the slings from the crane hook.
17. Attach the transfer cask lifting yoke to the crane and engage the lifting yoke to transfer cask trunnions. Disconnect the wall-mounted support struts from the transfer cask.
18. Lift the transfer cask from on top of the shipping cask and place it on top of the TranStor storage cask. Attach the support struts between the transfer cask and the transfer cell walls. Disengage the transfer cask lifting yoke from the crane and set aside.
19. Engage the crane hook to the slings connected to the canister lifting eyes, raise the canister slightly, remove door pins, and open the transfer cask shield doors.
20. Lower the canister from the transfer cask into the storage cask.
21. Disconnect the slings from the crane hook.
22. Close the transfer cask shield doors and install the door pins.
23. Attach the transfer cask lifting yoke to the crane and engage the lifting yoke to transfer cask trunnions. Disconnect wall-mounted support struts from the transfer cask. Move the transfer cask from the storage cask and place the transfer cask back into storage.

CASK STORAGE

24. Place the shielding ring over the gap between the storage cask inner surface and canister outer surface. Remove the canister lifting eyes and lifting slings.
25. Install the storage cask cover plate and cover plate bolts.
26. Perform a health physics survey for radiation doses.
27. Open the transfer cell door and drive the cask transporter into the cell straddling the TranStor storage cask.
28. Attach the cask transporter lift hoist to the storage cask lifting attachments.
29. Raise the TranStor storage cask approximately 4 inches.
30. Transport the storage cask from the Canister Transfer Building to the appropriate storage pad with the cask transporter.
31. Position the storage cask in its designated storage location and lower the cask to the pad.
32. Disconnect the cask transporter lift hoist from the storage cask.
33. Connect the storage cask temperature monitoring instrumentation.
34. Perform natural convection cooling operability testing.

(Reference 4, Section 7.3 and Reference 2, Section 8.1)
(Note: The exact operational sequence is controlled by PFSF procedures.)

Figure 5.1-3
TRANSTOR CANISTER TRANSFER OPERATIONAL SEQUENCE

PRIVATE FUEL STORAGE FACILITY
SAFETY ANALYSIS REPORT

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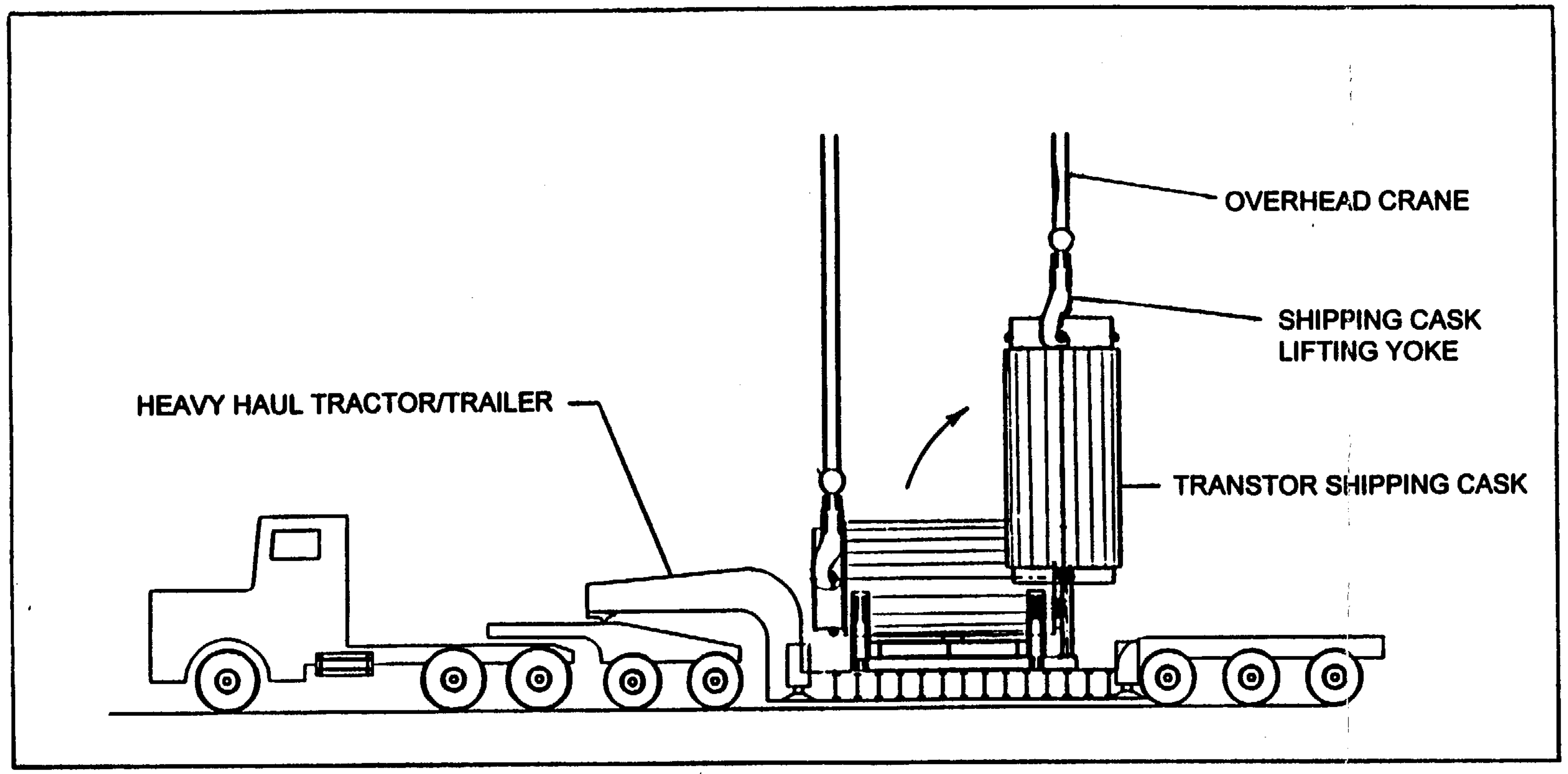
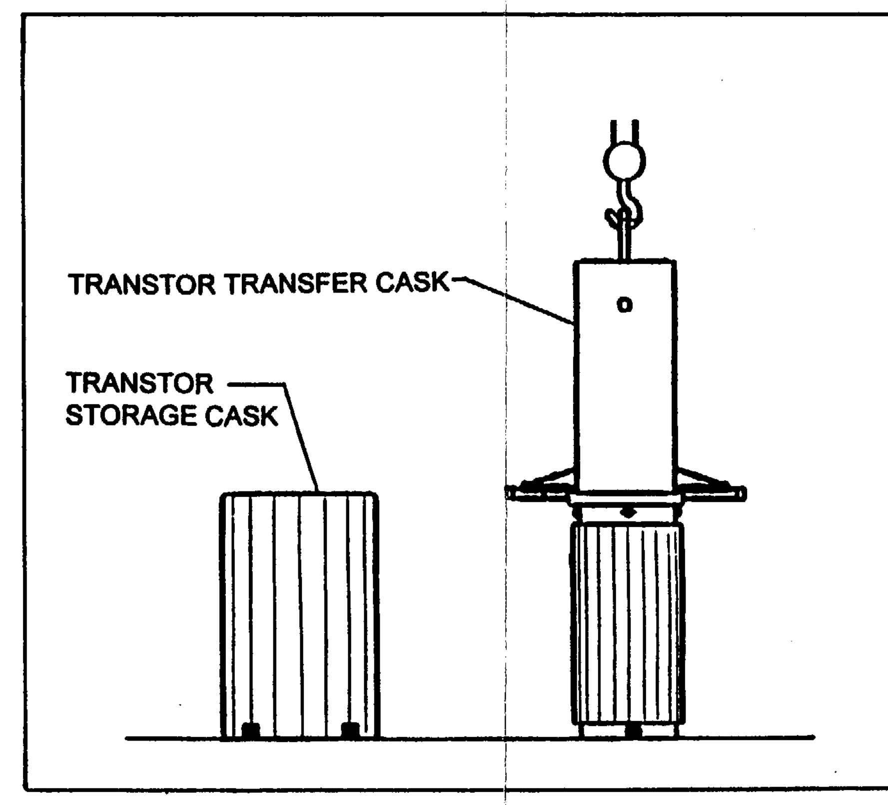
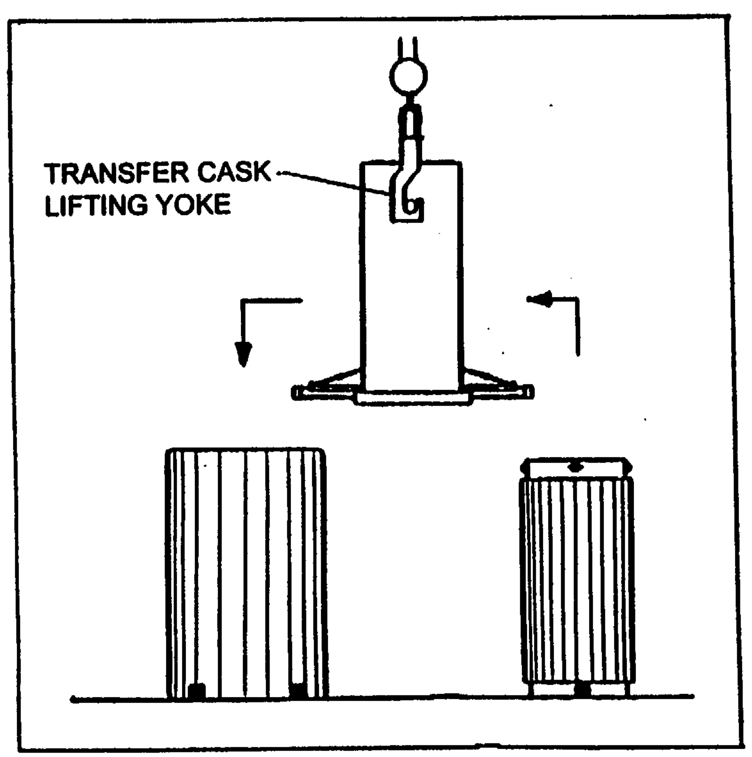


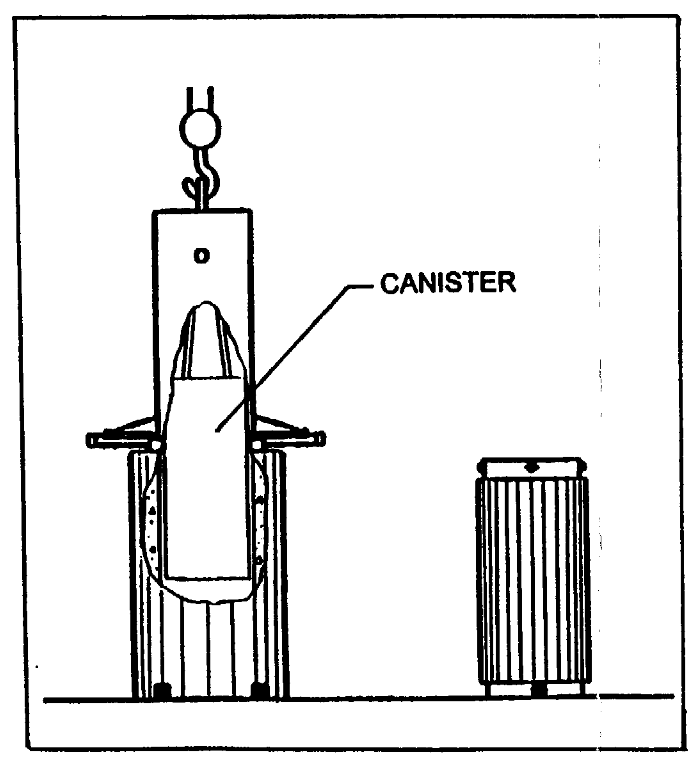
AFTER RECEIPT INSPECTION AND PLACEMENT INTO CANISTER TRANSFER BUILDING, UPRIGHT SHIPPING CASK, LIFT OFF TRAILER, AND MOVE TO TRANSFER CELL



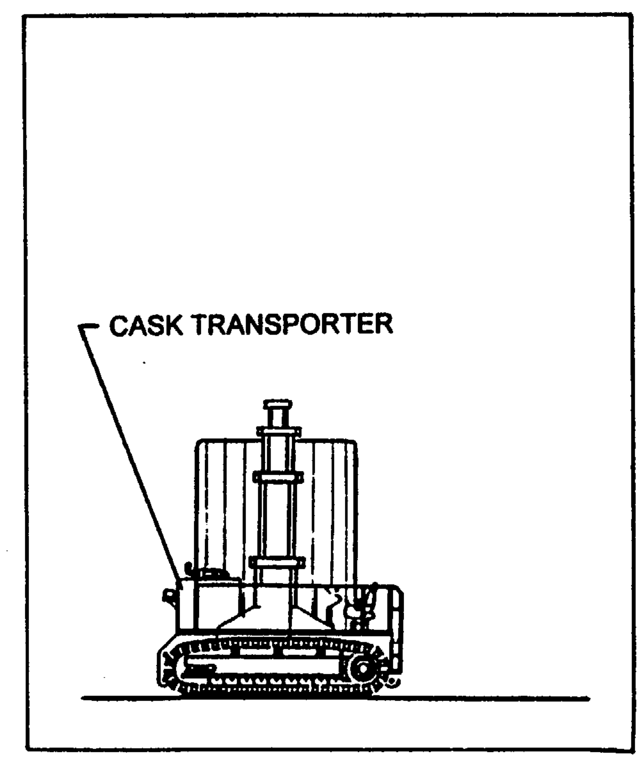
PLACE TRANSFER CASK ON SHIPPING CASK, OPEN HYDRAULIC DOORS, AND RAISE CANISTER INTO TRANSFER CASK VIA CRANE



MOVE TRANSFER CASK FROM TOP OF SHIPPING CASK TO TOP OF STORAGE CASK VIA CRANE



OPEN HYDRAULIC DOORS AND LOWER CANISTER INTO STORAGE CASK. REMOVE TRANSFER CASK AND INSTALL STORAGE CASK LID



TRANSPORT STORAGE CASK TO STORAGE PAD VIA CASK TRANSPORTER

Figure 5.1-4

TRANSTOR CANISTER TRANSFER OPERATION

PRIVATE FUEL STORAGE FACILITY
SAFETY ANALYSIS REPORT

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STORAGE CASK INSPECTION

1. Visually inspect the storage cask on the storage pad for any physical damage. Perform radiation survey. Disconnect temperature monitoring instrumentation.
2. Use the cask transporter to move the storage cask into the Canister Transfer Building.
3. Remove the storage cask cover and visually inspect the top of the canister for any degradation.

CANISTER TRANSFER

4. Install canister lifting devices and lifting slings. Lift and move the transfer cask on top of the storage cask. Open the transfer cask doors.
5. Lift the canister into the transfer cask. Close the transfer cask doors.
6. Move the transfer cask from on top of the concrete storage cask to the top of the shipping cask with the crane.
7. Open the transfer cask doors and lower the canister from the transfer cask into the shipping cask.
8. Remove the transfer cask from the top of the shipping cask and place into storage.

SHIP OFFSITE

9. Install the shipping cask closure lid, fill with helium, and perform leak tests.
10. Check radiation levels of the shipping cask.
11. Place the cask on the heavy haul trailer or rail car, lower the cask to its horizontal transport position, and install the impact limiters, tie downs, and personnel barrier.
12. Transport the shipping cask offsite.

(Note: The exact operational sequence is controlled by PFSF procedures.)

Figure 5.1-5

CANISTER SHIPMENT FROM THE PFSF OFFSITE OPERATIONAL SEQUENCE

CHAPTER 6

SITE-GENERATED WASTE CONFINEMENT AND MANAGEMENT

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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 and TranStor shipping cask Safety Analysis Reports (SARs) (References 1 and 2) and Chapter 8 of the HI-STORM and TranStor storage cask SARs (References 3 and 4). 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) and the interior of the transfer cask. In the event removable contamination levels, measured on accessible canister surfaces or inferred from levels measured inside the transfer cask, exceed the criteria specified in Chapter 10, 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.

6.2 OFFGAS TREATMENT AND VENTILATION

There are no gaseous releases from the storage systems utilized at the PFSF. After the canisters are loaded with spent fuel at the originating nuclear power plants, the canisters are vacuum dried, backfilled with helium, welded closed, and tested to verify leak tightness. Potentially contaminated gases that are purged from the canisters during the closure process are handled by the gaseous radioactive waste system at the originating nuclear power plant shipping the fuel. The canisters are ASME Boiler and Pressure Vessel Code Section III vessels designed to remain leak-tight for long-term storage at the PFSF. 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 are no special ventilation systems installed in the PFSF facilities. There are no credible scenarios that would require installation of special ventilation systems to protect against offgas or particulate filtration.

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6.3 LIQUID WASTE TREATMENT AND RETENTION

Radioactive liquid wastes are not generated at the PFSF. After the canisters are loaded with spent fuel, the canisters are vacuum dried, backfilled with helium, welded closed and tested to verify leak tightness at the originating nuclear power plants. Therefore, there is no potential for leakage of contaminated liquids from the canister internals.

At the originating nuclear power plants, outer surfaces of the shipping casks are surveyed and decontaminated as necessary so that removable contamination concentrations are below the DOT criteria (49 CFR 173.443). Upon receipt of shipping casks at the PFSF, the casks are surveyed to determine radiation and contamination levels. Removable contamination identified on the cask outer surfaces is wiped off with rags or paper wipes that can be disposed of as solid activated waste, preventing the generation of radioactive liquid wastes. This is in accordance with the Private Fuel Storage L.L.C.'s (PFSLLC) policy of preventing generation of liquid radioactive waste.

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.

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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 and TranStor shipping SARs). 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 addition, smears are taken on internal surfaces of the transfer cask, following transfer of the canister from the transfer cask into the shipping cask, and removable contamination levels on the transfer cask internal surfaces are considered to be representative of removable contamination levels on the outer surfaces of the canister. In the event canister removable contamination levels (measured on accessible canister surfaces or inferred from levels measured inside the transfer cask) exceed the criteria specified in Chapter 10, 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 Section 10.2.2.1, 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.

6.5 RADIOLOGICAL IMPACT OF NORMAL OPERATIONS - SUMMARY

Radiological impacts at the PFSF are minimized through the health physics program to maintain the "Start Clean / Stay Clean" philosophy and to maintain ALARA principles presented in Regulatory Guide 8.10 (Reference 6). 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 a canister and release of radioactive material associated with spent fuel from inside the canister. No releases of radioactive material to the environment are expected during normal facility operations and there are no radioactive gaseous or liquid effluents from the PFSF. Solid radioactive wastes are stored in suitable containers in the LLW holding cell of the Canister Transfer Building while awaiting shipment offsite to a LLW disposal facility. These wastes have negligible impact on the environment. The off-normal condition involving postulated release of removable surface contamination from a canister exterior from an event involving canister impact such as canister mishandling is evaluated in Section 8.1.5. This evaluation conservatively assumes that the entire outer surface of a canister is covered with Co-60 contamination at the maximum concentration permitted by Chapter 10, and that 100 percent of this radioactivity is removed from the canister and becomes airborne in respirable size particles. Doses to an individual at the owner controlled area (OCA) boundary from this worst case scenario are shown to be below 0.1 mrem. As such, the radiological impacts to the environment from normal operations at the PFSF (including off-normal conditions) are negligible.

Radiological impacts of the PFSF are summarized as follows:

- <u>Shipping and Canister Transfer Operations</u> - Under normal operating conditions, during receipt of the shipping cask, canister transfer operations, and movement of the loaded storage cask to the storage pad, no releases of radioactivity are expected to occur. As discussed above, the potential exists for small amounts of removable contamination deposited on the canister external surfaces from handling in the originating nuclear power plant spent fuel pool to be transferred from the canister onto the shipping, transfer, or storage casks. This contamination would be detected by smear surveys and removed using dry paper wipes or rags, producing small quantities of solid radioactive waste. There would be no radiological impact to locations outside of the restricted area or to members of the public as a result of this contamination. Constraints and equipment used to ensure as low as is reasonably achievable (ALARA) conditions while processing the contamination include storage in LLW containers, and removal of the LLW from the PFSF.

- Spent Fuel Storage - During spent fuel storage, no releases of any type of radioactive material occur. Therefore, there are no radiological waste impacts from the storage of spent fuel.

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 4, September 1996.

2. Safety Analysis Report for the TranStor Shipping Cask System, SNC-95-71SAR, Sierra Nuclear Corporation, Docket 71-9268, Revision 1, September 1996.

3. Topical Safety Analysis Report for the Holtec International Storage and Transfer Operation Reinforced Module Cask System (HI-STORM 100 Cask System), Holtec Report HI-951312, Docket 72-1014, Revision 1, January 1997.

4. Safety Analysis Report for the TranStor Storage Cask System, SNC-96-72SAR, Sierra Nuclear Corporation, Docket 72-1023, Revision B, March 1997.

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) and Sierra Nuclear Corporation (SNC), the vendors for the spent fuel storage systems, have incorporated a number of design features to provide ALARA conditions during transportation, handling, and storage as described in their HI-STORM and TranStor Safety Analysis Reports (SARs - References 6 and 7).

- 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

Section 10.2.2.1, 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. Airborne monitoring will be performed using portable monitors during canister handling operations.

- 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 a design earthquake (DE), 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). 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 vendors' cask designs and their associated radiological source term and dose evaluations. The following discussion summarizes their assessments. The vendors' source terms bound fuel that will be stored at the PFSF. For more complete information, refer to the Holtec and SNC storage system SARs (References 6 and 7, respectively).

7.2.1 Characterization of Sources

Both of the storage system vendors 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 8X8R 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	47.5 GWd/MTU
5-yr cooled	6-yr cooled

HI-STORM BWR Reference Intact Fuel

45 GWd/MTU

5-yr cooled

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 3.7 percent for the PWR intact fuel and 3.4 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.

SNC performed numerous shielding calculations with various combinations of burnup, cooling time, assembly uranium loading, and initial enrichment that produce heat generation rates approximately equal to (or above) the design limit of 26 kW/canister. The calculated dose rates were determined to be below the shielding design limits (Section 7.3.3) for all cases. SNC determined that, from a shielding perspective alone, the TranStor system could accommodate 60 GWd/MTU, 5-yr cooled fuel with lower bound initial enrichment and still have dose rates below the design limits. Such fuel could not actually be loaded into the TranStor system since its heat generation rate would exceed the 26 kW design limit. Based on this, SNC determined that the assembly heat generation limits are the limiting constraint which determines the minimum permissible cooling time for fuel to be loaded into the TranStor system, rather than shielding considerations. The shielding analyses presented in this section verify that the TranStor storage cask has sufficient shielding to ensure acceptable dose rates around the cask for any fuel that meets the heat generation requirements (1.083 kW per PWR fuel assembly and 0.426 kW per BWR fuel assembly).

SNC obtained source data from the Office of Civilian Radioactive Waste Management light water reactor Radiological Computer Database (OCRWM LWR Database - Reference 8) for generic PWR and BWR spent fuel having the following assumed characteristics that result in a fully loaded TranStor canister decay heat generation of approximately 26 kW:

TranStor PWR Reference Intact Fuel

40 GWd/MTU	45 GWd/MTU	50 GWd/MTU	60 GWd/MTU
5-yr cooled	6-yr cooled	8-yr cooled	13-yr cooled

TranStor BWR Reference Intact Fuel

35 GWd/MTU	40 GWd/MTU	45 GWd/MTU	50 GWd/MTU
5-yr cooled	6-yr cooled	7-yr cooled	8-yr cooled

The OCRWM LWR Database includes neutron and gamma source terms for PWR and BWR spent fuel as a function of burnup level, cooling time, and initial enrichment. The gamma and neutron source data compiled in the OCRWM LWR Database is based upon ORIGEN-2 calculations. The shielding analyses are conservatively based upon maximum assembly uranium loadings of 0.469 MTU for PWR fuel and 0.197 MTU for BWR fuel. SNC obtained source data for PWR and BWR spent fuel having the above characteristics from the OCRWM LWR Database, assuming low enrichments for the specified burnups. As described in the TranStor SAR, the TranStor system is designed to store stainless steel clad fuel as well as Zircaloy clad fuel. The TranStor canisters may also contain failed fuel. The fuel inventory for failed fuel is equal to or lower than that of intact fuel with the maximum fuel loading. Therefore, the source terms calculated for maximum loading intact fuel, described above, will bound the failed fuel cases.

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. Both vendors modeled the active fuel region as a homogeneous zone, assuming fuel, fuel assembly components, and canister internal materials uniformly distributed.

HI-STORM

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 0.7 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 0.7 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.

HI-STORM SAR Table 5.2.22 for the PWR canister (MPC-24), and Table 5.2.23 for the BWR canister (MPC-68) provide a comparison of design basis gamma sources for Zircaloy and stainless steel clad fuels. The Co-60 activities from the stainless steel cladding and BWR water channels were included in this comparison. The result is that the stainless steel fuel has a higher source in the 1.0 to 1.5 MeV energy range due to cobalt activation. However, the Zircaloy fuel has a higher souce 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. Holtec concluded that the total dose rates from Zircaloy clad fuel will bound those from stainless steel clad fuel and determined that it was not necessary to explicitly analyze the stainless steel clad fuels.

TranStor

TranStor fuel region gamma source descriptions are based on output of the OCRWM LWR Database, shown in TranStor SAR Tables 5.2-1 and 5.2-2 for reference PWR and BWR fuels, respectively. The source strengths presented in the OCRWM LWR Database are broken down by gamma energy levels. The 1.25 MeV gamma energy line source strength includes gammas from fission product decay as well as the additional gamma source (Co-60) from activated stainless steel structural members and cladding of control components, as discussed below.

Methodology used by SNC to account for the gamma source from activated non-fuel components in the fuel region is described in Section 5.2.1 of the TranStor SAR. The OCRWM LWR Database used by SNC includes projection of the gamma source from activated materials. The activated materials gamma source terms are based upon typical PWR and BWR core region non-fuel material inventories (e.g. cladding, grid spacers). The OCRWM LWR Database gives the cobalt mass (per MTU) and the Co-60 activity level as a function of burnup, cooling time, and initial enrichment. This enables determination of the fraction of cobalt that becomes activated in the active fuel region. In addition to the activated materials output by the OCRWM LWR Database, SNC also calculated the gamma source from stainless steel clad control components for PWR assemblies. The WE 17X17 assembly burnable poison rod components were determined to contain the greatest cobalt inventory within the active fuel region and were considered for the various burnups and cooling times noted above. SNC applied the same activation fraction to the cobalt in this control assembly that was calculated by the OCRWM LWR Database for other cobalt containing materials in the active fuel region, and superimposed the total cask Co-60 gamma source from the control components onto the 1.25 MeV gamma energy line.

SNC assessed the gamma source strengths associated with storage of stainless steel clad BWR and PWR fuel assemblies. As noted in Section 5.5.1 of the TranStor SAR, the entire stainless steel clad BWR spent fuel inventory has a burnup level under 25 GWd/MTU and will have a cooling time of over 10 years at the time of storage. As a result, the gamma source for the stainless steel clad BWR fuel is completely bounded by that shown in Table 5.2-2 of the TranStor SAR for 35 GWd/MTU, 5-year cooled Zircaloy clad BWR fuel. The neutron source strength is also bounded. However, SNC's assessment determined that the gamma source strength of PWR stainless steel clad fuel is not bounded by Zircaloy clad fuel, as discussed in Section 5.5.1 of the TranStor SAR and in the following paragraph.

The Co-60 inventory in the cladding was calculated for PWR stainless steel clad fuel with the same characteristics as reference Zircaloy clad PWR fuel: 40 GWd/MTU burnup (the maximum burnup level for stainless steel clad PWR fuel), 5-year cooling time and a lower bound enrichment of 3.02 percent. Twenty-four stainless steel clad fuel assemblies with these characteristics will generate approximately 26 kW decay heat (the TranStor canister heat generation limit). SNC applied the same methodology discussed above for the activated non-fuel components in the fuel region to calculate activation levels. Except for the cladding material, this fuel has the same neutron and gamma source strength as reference Zircaloy clad fuel. The difference is that there is an additional 1.25 MeV gamma source from Co-60 as a result of activated stainless steel cladding. Based on the stainless steel mass, the maximum cobalt content in stainless steel, and the core region Co-60 activation level (all given in the OCRWM LWR Database), the total core region Co-60 was calculated and converted into a 1.25 MeV gamma source strength. SNC determined that the additional source strength from the stainless steel cladding is 21 percent greater than the 1.25 MeV gamma source of the reference Zircaloy clad fuel, and the total gamma source strength for this energy line for stainless steel clad fuel will be a factor of 2.21 times that of Zircaloy clad fuel. Section 5.5.1 of the TranStor SAR indicates that a conservative estimate of the gamma dose rates around casks containing 40 GWd/MTU, 5-year cooled stainless steel clad fuel can be obtained by simply multiplying the gamma dose rates of 40 GWd/MTU, 5-year cooled Zircaloy clad reference fuel by a factor of 2.21.

7.2.1.2 Non-Fuel Region Gamma Source

HI-STORM

Holtec and SNC used similar methods to project the gamma source from the non-fuel regions of fuel assemblies. The gamma sources for the non-fuel regions of a fuel assembly are almost entirely due to Co-60 in activated metal components. For HI-STORM, the non-fuel region component masses identified in HI-STORM SAR Tables 5.2.1 through 5.2.3 (based on the OCRWM LWR Database and References 9 and 10) 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 11. The ORIGEN-S runs were based on core region neutron 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 11, 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 through 5.2.15 identify the Curies of Co-60 calculated in various regions of the reference fuel assemblies, for both Zircaloy and stainless steel clad fuels. These regions include, for PWR fuel, the lower end piece, gas plenum springs, gas plenum spacer, and upper end piece; and for BWR fuel, the lower end piece, gas plenum springs, expansion springs, grid spacer springs, the upper end piece, and the handle.

In the HI-STORM analyses, the gamma source from PWR activated control components was not explicitly added to the design basis gamma source, as it was considered that a conservatively large hardware and uranium mass combined with conservative Co-60 activity scaling factors produced a gamma source that bounds the source contribution of the control components. The B&W 15X15 assembly with the worst case burnup and cooling time combination is considered to provide a bounding gamma radiation source.

<u>TranStor</u>

While the OCRWM LWR Database does not output the gamma sources from non-fuel regions, SNC used information from this database to calculate the gamma source contributions from the bottom nozzle region, the gas plenum region, and the top nozzle region. The plenum region gamma source is from activated plenum springs. The top and bottom nozzle sources come from activated stainless steel and Inconel components in the assembly nozzles. The TranStor shielding analyses consider gamma dose rate contributions from each of these three non-fuel gamma source regions.

The OCRWM LWR Database identifies the various metals present in each non-fuel region of all major PWR and BWR assembly types, along with maximum cobalt concentrations for each metal type. The TranStor Co-60 gamma sources are based on the assembly type with the largest cobalt inventory for each non-fuel region. The presence of control components was considered and the total initial cobalt (Co-59) inventory for each non-fuel region determined. The active fuel region activation factors discussed above, which are a function of burnup, cooling time, and initial enrichment, were multiplied by adjustment factors calculated for each non-fuel region to yield Co-60 activation factors that apply to each non-fuel region. The cobalt inventory in each non-fuel region was multiplied by the corresponding Co-60 activation factor to yield a Co-60 gamma source activity for that region. These Co-60 activities were converted into 1.173 MeV and 1.333 MeV gamma source strengths, since each Co-60 decay produces two gammas having these energies. The TranStor non-fuel region gamma source strengths are shown in TranStor SAR Tables 5.2-3 (PWR) and 5.2-4 (BWR) for each of the three non-fuel regions.

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. The neutron sources for PWR and BWR fuels for TranStor, taken from the OCRWM LWR Database, are shown in the TranStor SAR Tables 5.2-5 and 5.2.6.

Unlike the gamma source spectrum, the neutron source spectrum does not vary significantly with fuel burnup level or cooling time. As noted above, SNC used the low initial enrichments permitted by the OCRWM LWR Database with each assumed burnup. Holtec assumed enrichments of 3.7 percent for the PWR fuel and 3.4 percent for the BWR fuel, which are below the average enrichments normally used to obtain the burnups analyzed. 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 (borated for PWR fuel pools) 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. For the TranStor fuel loading operation, a shield ring is placed in the annulus between the canister and transfer cask, which reduces the area of the annulus, and demineralized water or filtered fuel pool water (borated for PWR fuel pools) is continuously injected into the transfer cask/canister annulus. This water flows out of the area at the top of the annulus where the shield ring is installed, preventing the introduction of contaminated fuel pool water into the annulus when the transfer cask/canister is submerged in the spent fuel pool. As an alternative to this method, an inflatable seal can also be used in the TranStor fuel loading operation. 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 addition, smears are taken on internal surfaces of the transfer cask, following transfer of the canister from the transfer cask into the shipping cask, and removable contamination levels on the transfer cask internal surfaces are considered to be representative of removable contamination levels on the outer surfaces of the canister. In the event canister removable contamination levels (measured on canister top surfaces or inferred from levels measured inside the transfer cask) exceed the criteria specified in Section 10.2.2.1, 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 Section 10.2.2.1 (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 a hypothetical breach of a canister, 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 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.

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 35 mrem/hr on the side, 10 mrem/hr at the top, and 50 mrem/hr at the air vents. The TranStor design dose limits specified in Section 2.3.5.2 of the TranStor SAR are 15 mrem/hr 1 meter from the side of a TranStor storage cask (30 mrem/hr for stainless steel clad fuel) and 200 mrem/hr 1 meter above the center of the cask cover lid.

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 both the HI-STORM and the TranStor SARs 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(s) 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 liner(s), and the 2 to 2 1/2 ft thick concrete walls of the storage casks. The designs of the inlet and outlet ducts for both storage casks prevent direct radiation streaming from the canister to the cask exterior. Shielding materials and geometries are described in detail in the HI-STORM and TranStor SARs.

The transfer casks are 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(s). 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 thick steel plates for SNC's TranStor transfer cask, and 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.

In the TranStor shielding models, the fuel basket interior is subdivided into four axial regions: the bottom nozzle region, the active fuel region, the gas plenum region, and the top nozzle region. For purposes of modeling, the canister internal basket structure is mixed with the fuel assembly radiation source material into a single homogeneous material that fills each of the four axial regions. In the active fuel region, the homogeneous material includes spent fuel material (modeled as UO_2), the Zircaloy cladding and guide tube material (modeled as pure zirconium), the carbon steel fuel support sleeve material (modeled as pure iron), and the stainless steel control component cladding material (for PWR fuel). Several other basket internal components, such as the structural support tubes, which do not necessarily surround the radiation sources, were conservatively neglected in the homogeneous material density calculations. The materials from the neutron poison sheets are included in the fuel region homogeneous material for neutron calculations only.

7.3.3.2 Shielding Evaluation

The shielding analyses, documented in Chapter 5 of the HI-STORM and TranStor SARs, model generic reference PWR and BWR fuel (Section 7.2.1). Dose rates projected from the HI-STORM and TranStor storage casks 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 vendors consider different types of canisters in their shielding analyses: PWR canisters that contain 24 PWR spent fuel assemblies for both vendors and BWR canisters that contain 61 assemblies for the TranStor system and 68 assemblies for HI-STORM.

The HI-STORM shielding analyses were performed using the MCNP code. TranStor used a number of computer codes plus manual calculations for their shielding analyses including: ANISN-PC for storage cask side gamma and neutron dose rate, MCNP for the cask top neutron dose rate, and QADS for the cask top gamma dose rate. Source ratioing techniques are also used for scaling previously licensed VSC-24 dose rates where the geometries are the same. Discussions of the modeling codes and methodologies are included in Chapter 5 of the HI-STORM and the TranStor SARs.

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 and TranStor storage casks, 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 Tables 7.3-1 (HI-STORM) and 7.3-2 (TranStor) for the PWR and BWR reference fuel cases analyzed and documented in the vendor SARs. 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 and TranStor transfer casks. 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 fuels with 45 GWd/MTU burnup, 5-year cooling time, and 47.5 GWd/MTU burnup, 6-year cooling time. It was determined that dose rates from PWR reference fuel bound those produced by an MPC-68 canister of BWR reference fuel having the same burnup and cooling time. Table 7.3-4 presents calculated dose rates for the TranStor transfer cask, assuming PWR and BWR reference fuels with various burnups and cooling times. Figure 7.3-2 identifies the locations of the points relative to the transfer casks at which dose rates were calculated. The dose locations for point 4 differ slightly for the two transfer casks, with the HI-TRAC point located at the side of the cask above the neutron shield and the TranStor transfer cask point located at the top of the cask directly above the annulus between the canister and the inside of the transfer cask.

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

Each of the storage cask vendors calculated gamma and neutron dose rates at various distances from a single storage cask, 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 spent fuel basis for these calculations is that all 4,000 casks contain 40 GWd/MTU burnup and 10-year cooled PWR fuel, with a low initial enrichment assumed for this burnup. 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 vendors' minimum cooling time requirement for transporting 40 GWd/MTU PWR fuel is 10 years for the Holtec HI-STAR shipping cask system and 7 years for SNC's TranStor shipping cask system; 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.

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 duct openings 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. A surface source file was generated containing information regarding neutron and gamma tracks of the radiation leaving the surface of a single storage cask. This file was then used in the computation of dose rates at various distances from the cask, and in modeling the cask array.

SNC determined dose rates at various distances from a single TranStor storage cask by scaling dose rates from previous analyses performed with the SKYSHINE II code, using the gamma and neutron ratios of the fuel source strengths. Since SNC determined that a canister loaded with PWR reference fuel produces higher dose rates on the storage cask surface than a canister containing BWR reference fuel, the PWR case is bounding and was used for dose rate vs. distance analyses. The gamma and neutron source strength ratios were determined for 40 GWd/MTU burnup PWR fuel with 5-year cooling time (one of the TranStor reference fuels) vs. PWR fuel with 40 GWD/MTU burnup and 10-year cooling time, based on the OCRWM LWR Database. Gamma source strengths for the 10-year cooled fuel are less than half of those for the 5-year cooled fuel for all gamma energy lines between 0.8 and 2.75 MeV. Therefore, the single cask gamma dose rates at various distances previously calculated by SNC (Reference 12) for the 40 GWd/MTU, 5-year cooled PWR reference fuel were all divided by two to yield gamma dose rates vs. distance for the 40 GWd/MTU, 10-year cooled fuel. This approach is conservative since ratios of gamma source strengths for most energy lines of the fuel with 10 vs. 5 year cooling times are well below 0.5.

Information from the OCRWM LWR Database shows that the total neutron source strength for 40 GWd/MTU, 10-year cooled PWR fuel is 0.83 times the total neutron source strength for 40 GWd/MTU, 5-year cooled fuel. Since the neutron source spectrum does not vary significantly with fuel cooling time, the single neutron dose rate vs. distance values previously calculated by SNC (Reference 12) for the 40 GWd/MTU, 5-year cooled PWR reference fuel were multiplied by 0.83 to yield the single cask neutron dose rate vs. distance data for 40 GWd/MTU, 10-year cooled fuel.

The single storage cask dose rate versus distance data for HI-STORM and TranStor casks containing 40 GWd/MTU, 10-year cooled fuel are shown in Tables 7.3-5 and 7.3-6 for the following four components: gammas and neutrons from the cask side and top. This data was 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 by the vendors and results of dose rate projections from the PFSF array, assuming the PFSF is filled with either HI-STORM or TranStor storage casks containing 40 GWd/MTU, 10-year cooled fuel.

HI-STORM

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. Total dose rates at the RA fence (150 ft from the nearest storage pads) at the north side of the array are 1.19 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 0.98 mrem/hr. These dose rates are less than the 2 mrem/hour criteria for unrestricted areas specified in 10 CFR 20.1301 and are therefore acceptable. The total dose rates at the OCA boundary were calculated to be 1.94 E-3 mrem/hr at a point on the boundary 1,969 ft (600 meters) north of the RA fence, and 1.17 E-3 mrem/hr at a point on the boundary 600 meters west of the RA fence. 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 3.88 mrem. This is 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 is therefore acceptable.

<u>TranStor</u>

SNC calculated dose rates at various distances from the PFSF array, making several simplifying assumptions that result in conservative projected dose rates, as discussed in Reference 14. For cask top dose rate contributions, the casks in the array are subdivided into groups of two rows running perpendicular to the dose receptor points. The top dose rates at distances are a function of all 4,000 casks at the PFSF, since radiation leaving the tops of the casks and reflected back down to a dose receptor point on the ground would not be shielded by other casks. Dose rates vs. distance from scattered gammas and neutrons leaving the top of a single TranStor storage cask containing 40 GWd/MTU, 10-year cooled fuel are given in Table 7.3-6. The top contribution from each group was calculated by multiplying the top dose rate from the nearest (center) cask by the total number of actual casks in each group. Dose rates from the tops of all groups of casks were summed to obtain the total dose rate at the dose receptor for all the groups of rows in the PFSF array. This method conservatively neglects the lower dose rate contributions that would be produced from casks located away from the center that are at greater distances from the dose receptor than the center cask.

The casks are spaced at a 15-ft pitch, resulting in about 3.7 ft between adjacent TranStor casks. Because of the close proximity of the casks and the fact that casks are positioned in evenly spaced rows and columns and not staggered, the analysis assumed that cask side dose rate contributions from all casks except those on the edge of the ISFSI are completely blocked by other casks. Streaming could be significant where substantial space exists between storage casks, such as the 150-ft spaces between storage pad quadrants, and the 30-ft spacing between columns of storage pads. To conservatively account for streaming in these spaces between casks, SNC assumed that the spaces between columns and the space between quadrants in the front row of casks are filled in with additional casks spaced at 15-ft pitch. Thus, there are assumed to be two additional casks in each 30-ft space between columns and 10 additional casks in the 150-ft space between quadrants. This "hole plugging" assumption results in an additional 56 casks assumed to be in the front row along the north edge of the PFSF, for a total of 106 casks assumed in the front row for the dose rate vs. distance analysis. The dose rates at various distances from the side of a single TranStor storage cask containing 40 GWd/MTU, 10-year cooled fuel are given in Table 7.3-6. For a dose receptor point assumed to be centered in front of the north side of the PFSF storage area, SNC determined the distance to the nearest (center) cask, and the dose rate at this distance. This dose rate was multiplied by 106 to estimate the total side dose from the assumed 106 casks along the north edge. This approach is conservative since it assumes that all the casks in the front row are at the same distance from the dose receptor point, whereas in actuality casks near the east and west ends of the front row are much further from the dose receptor point than the nearest casks in the center of the row.

The results of the dose rate vs. distance analysis for the PFSF array full of TranStor storage casks are given in Table 7.3-8. Total dose rates at the RA fence at the north side of the array (highest doses) are calculated to be 0.45 mrem/hr. This is less than the 2 mrem/hour criteria for unrestricted areas specified in 10 CFR 20.1301 and is therefore acceptable. The total dose rate at the OCA boundary 600 meters north of the RA fence was calculated to be 1.21 E-3 mrem/hour. It was determined that the dose rates at the north OCA boundary were higher than those along the other sides, even though the west OCA boundary is the same distance from the storage pads (646 meters) as the north boundary. Conservatively assuming a hypothetical individual spends 2,000 hours per year at the portion of the OCA boundary fence with the highest calculated dose rate from the storage cask array results in an annual dose of 2.42 mrem. This is 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 is 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, both Holtec and SNC estimated dose rates 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 (1.94 E-3 mrem/hr) out to a distance of 2.0 miles using a power curve. The result was 2.7 E-6 mrem/hr, which would result in an annual dose of 0.024 mrem at a distance of 2.0 miles from the OCA boundary, assuming a person continually present (8,760 hrs/yr) at this location.

SNC made an estimate of the dose rate at 10,000 ft from the PFSF using the following approach: Based on data from Table 7.3-8, the dose per year at 2,000 ft is about 11 mrem/yr, assuming an occupancy factor of 8,760 hrs/yr. Table 7.3-8 also indicates that the dose rate decreases by at least a factor of five for every 1,000 ft of distance from the PFSF, for distances greater than 1,000 ft. Therefore an estimate for the 10,000 ft annual accrued dose is 11 mrem/yr divided by 5 to the eighth power, or 3 E-5 mrem/yr. Although this approximation has large uncertainty because of the long distance involved, SNC considered that the maximum dose rate at 10,000 ft would be far less than 0.1 mrem/yr.

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.

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 limits on surface contamination concentrations, as discussed in Chapter 10, it is not necessary to install fixed airborne activity monitors in the Canister Transfer Building to detect minor contamination releases. However, airborne monitoring will be performed using portable monitors during canister handling operations. 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. Table 7.4-2 shows the estimated occupational exposures to PFSF personnel for these operations involving the TranStor shipping, transfer, and storage systems. 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. The operational sequence for these operations is also described in Chapter 5.

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 180 person-mrem for both HI-STORM and TranStor systems. 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 36 person-rem. 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 are considered in the Table 7.4-1 and 7.4-2 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 1.3 person-rem annually, based on one inspection per shift (3 shifts per day, 365 days per year) along the RA fence, using the highest 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 15 mrem/hr during the inspection, this surveillance will result in approximately 1.0 person-rem per quarter to PFSF personnel conducting the inspections, for a total of 4.0 person-rem annually. 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, in a dose field of 15 mrem/hr, an additional annual dose of 0.5 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 5.8 person-rem annually.

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.

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7.5 RADIATION PROTECTION PROGRAM

7.5.1 Organization

The PFSF Radiation Protection Manager reports to the General Manager (Figure 9.1) 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.

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). 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.

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. Maximum dose rates of 1.94 E-3 mrem/hr (HI-STORM) and 1.21 E-3 mrem/hr (TranStor) were 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. These dose rates are comprised of direct and scattered gamma and neutron radiation emanating from 4,000 storage casks and are conservatively based on the assumption that all 4,000 casks contain fuel with 40 GWd/MTU burnup and 10-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 maximum dose rate calculated from the storage casks (1.94 E-3 mrem/hr), a hypothetical individual conservatively assumed to spend 2,000 hours a year at the OCA fence would receive a maximum dose of 3.88 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 2.7 E-6 mrem/hr 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 0.024 mrem, which 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 and breach of a canister is not a credible event. Since there will be no liquid or gaseous effluents released from the PFSF, 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.

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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. Topical Safety Analysis Report for the Holtec International Storage and Transfer Operation Reinforced Module Cask System (HI-STORM 100 Cask System), Holtec Report HI-951312, Docket 72-1014, Revision 1, January 1997.

7. Safety Analysis Report for the TranStor Storage Cask System, SNC-96-72SAR, Sierra Nuclear Corporation, Docket 72-1023, Revision B, March 1997.

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. Sierra Nuclear Corporation Design Calculation No. PGE01.10.02.01-07, Revision 0, TranStor PWR System Dose versus Distance For a Single Cask, February 1996.

13. Holtec Report HI-971645, Radiation Shielding Analysis for the Private Fuel Storage Facility, Holtec International, March 1997.

14. Sierra Nuclear Corporation Design Calculation No. PFS01.10.02.03, Revision 0, Private Fuel Storage SKYSHINE-II ISFSI Dose Rate Calculation (4000 Casks), April 10,1997.

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.

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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	PWR 47.5 GWd/MTU 6-year cool	BWR 45 GWd/MTU 5-year cool
1) Side, contact	γ = 27.60 n = 1.31 Tot = 28.91	γ = 20.38 n = 1.56 Tot = 21.94	γ = 28.05 n = 1.27 Tot = 29.32
2) Side, 1 meter	γ = 13.66 n = 0.55 Tot = 14.21	Not computed	γ = 13.69 n = 0.51 Tot = 14.20
3) Top, contact	γ = 3.69 n = 3.01 Tot = 6.70	γ = 3.29 n = 3.59 Tot = 6.88	γ = 2.67 n = 1.85 Tot = 4.52
4) Top, 1 meter	γ = 1.02 n = 0.84 Tot = 1.86	Not computed	γ = 0.73 n = 0.51 Tot = 1.24
5) Top vent, contact	γ = 27.66 n = 4.16 Tot = 31.82	γ = 24.39 n = 4.96 Tot = 29.35	γ = 23.50 n = 2.55 Tot = 26.05
6) Bottom vent, contact	γ = 45.94 n = 4.24 Tot = 50.18	γ = 40.11 n = 5.06 Tot = 45.17	γ = 36.41 n = 4.08 Tot = 40.49

TABLE 7.3-2

MAXIMUM DOSE RATES ON CONTACT AND AT ONE METER

FROM A TRANSTOR STORAGE CASK (mrem/hr)

(for 26 kW/canister fuel)

Detector Location (see Figure 7.3-1)	PWR 40 GWd/MTU 5-year cool	PWR 45 GWd/MTU 6-year cool	PWR 50 GWd/MTU 8-year cool	PWR 60 GWd/MTU 13-year cool	BWR 35 GWd/MTU 5-year cool	BWR 40 GWd/MTU 6-year cool	BWR 45 GWd/MTU 7-year cool	BWR 50 GWd/MTU 8-year cool
1) Side, contact	γ = 18.2 n = 0.7 Tot = 18.9	γ = 17.3 n = 0.9 Tot = 18.2	γ = 14.4 n = 1.1 Tot = 15.5	γ = 12.7 n = 1.4 Tot = 14.1	γ = 17.1 n = 0.7 Tot = 17.8	γ = 17.0 n = 1.3 Tot = 18.3	γ = 15.1 n = 1.4 Tot = 16.5	γ = 14.5 n = 1.4 Tot = 15.9
2) Side, one meter	γ = 9.7 n = 0.4 Tot = 10.1	γ = 9.2 n = 0.5 Tot = 9.7	γ = 7.7 n = 0.6 Tot = 8.3	γ = 6.7 n = 0.8 Tot = 7.5	γ = 9.1 n = 0.4 Tot = 9.5	γ = 9.1 n = 0.7 Tot = 9.8	γ = 8.1 n = 0.7 Tot = 8.8	γ = 7.7 n = 0.8 Tot = 8.5
3) Top, contact	γ = 11.5 n = 67.8 Tot = 79.3	γ = 10.4 n = 87.8 Tot = 98.2	γ = 8.2 n = 106.4 Tot = 114.6	γ = 4.9 n = 141.0 Tot = 145.9	γ = 14.8 n = 77.2 Tot = 92.0	γ = 14.2 n = 135.8 Tot = 150.0	γ = 12.3 n = 141.3 Tot = 153.6	γ = 10.7 n = 146.1 Tot = 156.8
4) Top, one meter	γ = 8.5 n = 58.1 Tot = 66.6	γ = 7.7 n = 75.3 Tot = 83.0	γ = 6.1 n = 91.3 Tot = 97.4	γ = 3.6 n = 120.9 Tot = 124.5	γ = 11.0 n = 66.2 Tot = 77.2	γ = 10.5 n = 116.5 Tot = 127.0	γ = 9.1 n = 121.2 Tot = 130.3	γ = 8.0 n = 125.3 Tot = 133.3
5) Top vent, contact	γ = 1.0 n = 6.1 Tot = 7.1	γ = 1.0 n = 6.1 Tot = 7.1	γ = 1.0 n = 6.1 Tot = 7.1	γ = 1.0 n = 6.1 Tot = 7.1	γ = 1.3 n = 6.2 Tot = 7.5	γ = 1.3 n = 6.2 Tot = 7.5	γ = 1.3 n = 6.2 Tot = 7.5	γ = 1.3 n = 6.2 Tot = 7.5
6) Bottom vent, contact	γ = 11.3 n = 1.9 Tot = 13.2	γ = 11.3 n = 1.9 Tot = 13.2	γ = 11.3 n = 1.9 Tot = 13.2	γ = 11.3 n = 1.9 Tot = 13.2	γ = 12.3 n = 1.7 Tot = 14.0	γ = 12.3 n = 1.7 Tot = 14.0	γ = 12.3 n = 1.7 Tot = 14.0	γ = 12.3 n = 1.7 Tot = 14.0

TABLE 7.3-3

MAXIMUM DOSE RATES ASSOCIATED WITH A HI-TRAC TRANSFER CASK

(mrem/hr)

Detector Location (see Figure 7.3-2)	PWR 45 GWd/MTU 5-year cool	PWR 47.5 GWd/MTU 6-year cool
1) Side, contact	γ = 89.76 n = 29.97 Tot =119.73	γ = 77.63 n = 37.47 Tot = 115.10
2) Side, 1 meter	γ = 37.12 n = 11.73 Tot = 48.85	Not computed
3) Top, contact	γ = 277.63 n = 269.99 Tot = 547.62	γ = 251.65 n = 322.21 Tot = 573.86
4) Top, side contact	γ = 39.34 n = 222.57 Tot = 261.91	γ = 37.56 n = 265.61 Tot = 303.17
5) Bottom, center, contact	γ = 377.06 n = 65.38 Tot = 442.44	γ = 342.10 n = 78.03 Tot = 420.13
6) Bottom, side, contact	γ = 123.63 n = 98.35 Tot = 221.98	γ = 116.41 n = 117.36 Tot = 233.77

TABLE 7.3-4

MAXIMUM DOSE RATES ASSOCIATED WITH A TRANSTOR TRANSFER CASK (mrem/hr)

(for 26 kW/canister fuel)

Detector Location (see Figure 7.3-2)	PWR 40 GWd/MTU 5-year cool	PWR 45 GWd/MTU 6-year cool	PWR 50 GWd/MTU 8-year cool	PWR 60 GWd/MTU 13-year cool	BWR 35 GWd/MTU 5-year cool	BWR 40 GWd/MTU 6-year cool	BWR 45 GWd/MTU 7-year cool	BWR 50 GWd/MTU 8-year cool
1) Side, contact	γ = 108 n = 75 Tot = 183	γ = 93 n = 98 Tot = 191	γ = 72 n = 118 Tot = 190	γ = 46 n = 157 Tot = 203	γ = 63 n = 63 Tot = 126	γ = 59 n = 111 Tot = 170	γ = 49 n = 116 Tot = 165	γ = 44 n = 119 Tot = 163
2) Side, one meter	γ = 55 n = 24 Tot = 79	γ = 48 n = 31 Tot = 79	γ = 37 n = 37 Tot = 74	γ = 23 n = 50 Tot = 73	γ = 33 n = 22 Tot = 55	γ = 30 n = 39 Tot = 69	γ = 25 n = 41 Tot = 66	γ = 22 n = 42 Tot = 64
3) Top, contact	γ = 25 n = 87 Tot = 112	γ = 23 n = 112 Tot = 135	γ = 18 n = 136 Tot = 154	γ = 11 n = 180 Tot = 191	γ = 32 n = 96 Tot = 128	γ = 31 n = 168 Tot = 199	γ = 27 n = 175 Tot = 202	γ = 23 n = 181 Tot = 204
4) Top of annulus, contact	γ = 937 n = 121 Tot = 1058	γ = 881 n = 157 Tot = 1038	γ = 720 n = 190 Tot = 910	γ = 419 n = 252 Tot = 671	γ = 1222 n = 134 Tot = 1356	γ = 1191 n = 235 Tot = 1426	γ = 1041 n = 244 Tot = 1285	γ = 910 n = 253 Tot = 1163
5) Bottom center, contact	γ = 85 n = 260 Tot = 345	γ = 76 n = 336 Tot = 412	γ = 60 n = 407 Tot = 467	γ = 37 n = 539 Tot = 576	γ = 95 n = 291 Tot = 386	γ = 92 n = 511 Tot = 603	γ = 79 n = 532 Tot = 611	γ = 69 n = 550 Tot = 619
6) Bottom side, contact	γ = 40 n = 143 Tot = 183	γ = 35 n = 186 Tot = 221	γ = 28 n = 225 Tot = 253	γ = 17 n = 298 Tot = 315	γ = 45 n = 173 Tot = 218	γ = 43 n = 305 Tot = 348	γ = 37 n = 318 Tot = 355	γ = 33 n = 328 Tot = 361

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)	1.45 E-2	1.19 E-3	5.06 E-5	2.34 E-4	1.60 E-2
328 ft (100 m)	2.48 E-3	2.09 E-4	1.61 E-5	5.76 E-5	2.76 E-3
656 ft (200 m)	3.42 E-4	3.57 E-5	3.48 E-6	9.13 E-6	3.90 E-4
984 ft (300 m)	8.97 E-5	9.40 E-6	9.63 E-7	2.16 E-6	1.02 E-4
1476 ft (450 m)	1.67 E-5	2.46 E-6	1.69 E-7	3.09 E-7	1.96 E-5
1969 ft (600 m - OCA fence)	4.46 E-6	6.41 E-7	3.84 E-8	5.97 E-8	5.20 E-6

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

TABLE 7.3-6

DOSE RATES VERSUS DISTANCE FOR A SINGLE TRANSTOR STORAGE CASK *

(mrem/hr)

Distance from Cask Side (ft)	Cask Side Gamma Dose Rate	Cask Side Neutron Dose Rate	Cask Top Gamma Dose Rate	Cask Top Neutron Dose Rate	Total Dose Rate
50	6.30 E-2	3.71 E-4	1.44 E-3	2.36 E-5	6.48 E-2
200	4.32 E-3	2.24 E-5	1.91 E-4	2.93 E-6	4.54 E-3
1,000	9.70 E-5	4.09 E-7	4.75 E-6	8.88 E-8	1.02 E-4
2,000	4.24 E-6	1.84 E-8	1.87 E-7	4.04 E-9	4.45 E-6

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

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)**	1.13 E0	6.09 E-2	1.19
150 ft east or west (security fence)	9.15 E-1	6.01 E-2	0.98
2,119 ft north (OCA boundary)***	1.87 E-3	6.90 E-5	1.94 E-3
2,119 ft west (OCA boundary)***	1.11 E-3	6.15 E-5	1.17 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

DOSE RATES VERSUS DISTANCE FROM THE PFSF ARRAY OF 4,000 ASSUMED TRANSTOR STORAGE CASKS *

(mrem/hr)

Distance to Detector from Security Fence (ft)	Gamma Dose Rate from Cask Sides	Neutron Dose Rate from Cask Sides	Gamma Dose Rate from Cask Tops	Neutron Dose Rate from Cask Tops	Total Dose Rate
0 (at security fence)	2.66 E-1	1.37 E-3	1.85 E-1	2.96 E-3	4.55 E-1
50	1.86 E-1	9.35 E-4	1.34 E-1	2.17 E-3	3.23 E-1
200	1.08 E-1	5.24 E-4	6.78 E-2	1.14 E-3	1.77 E-1
500	3.17 E-2	1.43 E-4	1.77 E-2	3.20 E-4	4.99 E-2
1,000	4.99 E-3	2.12 E-5	2.95 E-3	5.90 E-5	8.02 E-3
1,500	1.11 E-3	4.80 E-6	9.38 E-4	2.01 E-5	2.07 E-3
1,969 (600 m) OCA** boundary	4.49 E-4	1.95 E-6	7.46 E-4	1.62 E-5	1.21 E-3

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

** OCA - Owner Controlled Area

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	2.5 2.5	2.5 1.3
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	2.5 5.0	5.0 8.0
4. Remove impact limiters and tiedowns.	2 Ops 1 HP	1.5	1.5 1.5	5.0 1.0	15.0 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	5.0 1.0	5.0 1.0
6. Sample enclosed cask gas and vent.	1 Op 1 HP	0.5	0.5 0.5	12.5 / 2.0 1.0	6.3 /1.0 0.5
7. Remove HI-STAR closure plate (lid) bolts.	2 Ops 1 HP	1.0	1.0 1.0	12.5 / 2.0 1.0	25.0 / 4.0 1.0
8. Remove HI-STAR closure plate (lid).	2 Ops 1 HP	0.2	0.2 0.2	12.5 1.0	5.0 0.2
9. Prep HI-STAR to mate with HI-TRAC transfer cask.	2 Ops 1 HP	0.2	0.2 0.2	12.5 1.0	5.0 0.2
10. Install canister lift cleats and attach slings.	2 Ops 1 HP	1.0	1.0 1.0	12.5 / 2.0 1.0	25.0 / 4.0 1.0

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 1 HP	0.5	0.5 0.5	2.0 1.0	2.0 0.5
12. Mount HI-TRAC on top of HI-STAR.	2 Ops 1 HP	0.5	0.5 0.5	2.0 1.0	2.0 0.5
13. Open HI-TRAC transfer cask doors.	2 Ops 1 HP	0.2	0.2 0.2	5.0 1.0	2.0 0.2
14. Attach slings to canister downloader hoist and raise canister.	2 Ops 1 HP	0.5	0.5 0.5	2.0 1.0	2.0 0.5
15. Close HI-TRAC doors and install pins.	2 Ops 1 HP	0.2	0.2 0.2	10.0 1.0	4.0 0.2
16. Lower canister onto HI-TRAC doors.	2 Ops 1 HP	0.2	0.2 0.2	1.0 1.0	0.4 0.2
17. Prep HI-STORM storage cask to mate with HI-TRAC transfer cask (including installation of HI-STORM shielding inserts).	2 Ops 1 HP	0.2	0.2 0.2	5.0 1.0	2.0 0.2
18. Move HI-TRAC from HI-STAR to HI-STORM.	2 Ops 1 HP	0.7	0.7 0.7	1.0 1.0	1.4 0.7
19. Raise canister and open HI-TRAC doors.	2 Ops 1 HP	0.5	0.2 0.5	10.0 1.0	4.0 0.5
20. Lower canister into HI-STORM storage cask.	2 Ops 1 HP	0.5	0.5 0.5	1.0 1.0	1.0 0.5
21. Disconnect lifting slings.	2 Ops 1 HP	0.2	0.2 0.2	7.5 1.0	3.0 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	5.0	2.0
	1 HP		0.2	1.0	0.2
23. Remove HI-TRAC from HI-STORM.	2 Ops	0.5	0.5	1.0	1.0
	1 HP		0.5	1.0	0.5
24. Remove canister lift cleats and HI-STORM shield inserts.	2 Ops	0.5	0.5	12.5 / 2.0	12.5 / 2.0
	1 HP		0.5	1.0	0.5
25. Install HI-STORM lid and lid bolts.	2 Ops	1.0	1.0	7.5 / 2.0	15.0 / 4.0
	1 HP		1.0	1.0	1.0
26. Perform dose survey and install HI-STORM lifting eyes.	2 Ops	0.5	0.5	2.0	2.0
	1 HP		0.5	1.0	0.5
27. Drive cask transporter in transfer cell.	1 Op	0.3	0.3	1.0	0.3
	1 HP		0.3	1.0	0.3
28. Connect HI-STORM to cask transporter.	2 Ops	0.5	0.5	1.0	1.0
	1 HP		0.5	1.0	0.5
29. Raise HI-STORM storage cask.	2 Ops	0.2	0.2	1.0	0.4
	1 HP		0.2	1.0	0.2
30. Transport HI-STORM cask to storage pad.	2 Ops	2.0	2.0	1.0	4.0
	1 HP		2.0	1.0	2.0
31. Position and lower HI-STORM cask on pad.	2 Ops	0.5	0.5	10.0	10.0
	1 HP		0.5	5.0	2.5
32. Disconnect HI-STORM cask from transporter and remove cask lifting eyes.	2 Ops	1.0	1.0	10.0	20.0
	1 HP		1.0	5.0	5.0

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.	2 Ops 1 HP	0.5	0.5 0.5	10.0 5.0	10.0 2.5
34. Perform cask operability tests.	1 Op 1 HP	48	1.0 1.0	10.0 5.0	10.0 5.0
TOTAL					245.4 / 176.6 [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.

TABLE 7.4-2
(Page 1 of 4)

ESTIMATED PERSONNEL EXPOSURES FOR TRANSTOR 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	3.0 3.0	3.0 1.5
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	3.0 5.0	6.0 8.0
4. Remove impact limiters and tiedowns and install cask rotation trunnions.	2 Ops 1 HP	1.5	1.5 1.5	2.2 1.0	6.6 1.5
5. Attach lifting yoke to crane and TranStor shipping cask. Upright shipping cask and move to transfer cell.	2 Ops 1 HP	1.0	0.5 1.0	2.0 1.0	2.0 1.0
6. Sample enclosed cask gas and vent.	1 Op 1 HP	0.5	0.5 0.5	13.5 / 2.0 1.0	6.8 / 1.0 0.5
7. Remove shipping cask closure lid bolts.	2 Ops 1 HP	1.0	1.0 1.0	13.5 / 2.0 1.0	27.0 / 4.0 1.0
8. Remove shipping cask closure lid.	2 Ops 1 HP	0.2	0.2 0.2	13.5 1.0	5.4 0.2
9. Prep shipping cask to mate with TranStor transfer cask.	2 Ops 1 HP	0.2	0.2 0.2	16.8 1.0	6.7 0.2

TABLE 7.4-2 (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)
10. Install canister lift eyes and attach slings.	2 Ops 1 HP	1.0	1.0 1.0	16.8 / 2.0 1.0	33.6 / 4.0 1.0
11. Attach lifting yoke to crane and transfer cask.	2 Ops 1 HP	0.5	0.5 0.5	2.0 1.0	2.0 0.5
12. Mount transfer cask on top of shipping cask, connect support struts, and disengage crane.	2 Ops 1 HP	0.7	0.7 0.7	2.0 1.0	2.8 0.7
13. Open transfer cask doors.	2 Ops 1 HP	0.2	0.2 0.2	5.9 1.0	2.4 0.2
14. Attach slings to crane and raise canister.	2 Ops 1 HP	0.5	0.5 0.5	2.0 1.0	2.0 0.5
15. Close transfer cask doors and install pins.	2 Ops 1 HP	0.2	0.2 0.2	5.9 1.0	2.4 0.2
16. Lower canister onto transfer cask doors and disconnect canister slings from crane hook.	2 Ops 1 HP	0.2	0.2 0.2	13.1 1.0	5.2 0.2
17. Attach lifting yoke to crane hook and engage transfer cask. Disconnect support struts.	2 Ops 1 HP	0.5	0.2 0.5	13.1 1.0	5.2 0.5
18. Move transfer cask from shipping cask to storage cask. Attach support struts to transfer cask and disengage crane.	2 Ops 1 HP	1.0	0.3 1.0	13.1 1.0	7.9 1.0
19. Engage crane to canister lifting slings, raise canister, and open transfer cask doors.	1 Op 1 HP	0.5	0.3 0.5	13.1 1.0	3.9 0.5

TABLE 7.4-2 (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)
20. Lower canister into TranStor storage cask.	2 Ops	0.5	0.5	1.0	1.0
	1 HP		0.5	1.0	0.5
21. Disconnect canister lifting slings.	2 Ops	0.2	0.2	1.0	0.4
	1 HP		0.2	1.0	0.2
22. Close transfer cask doors.	1 Op	0.2	0.2	5.9	1.2
	1 HP		0.2	1.0	0.2
23. Attach lifting yoke to crane and engage to transfer cask. Remove transfer cask from storage cask.	2 Ops	0.8	0.8	2.0	3.2
	1 HP		0.8	1.0	0.8
24. Remove canister lifting eyes.	2 Ops	0.5	0.5	16.8 / 2.0	16.8 / 2.0
	1 HP		0.5	1.0	0.5
25. Install storage cask shield ring, lid, and lid bolts.	2 Ops	1.0	1.0	15.2 / 2.0	30.4 / 4.0
	1 HP		1.0	1.0	1.0
26. Perform dose survey and install storage cask lifting eyes.	2 Ops	0.5	0.5	15.2 / 2.0	15.2 / 2.0
	1 HP		0.5	1.0	0.5
27. Drive cask transporter in transfer cell.	1 Op	0.3	0.3	1.0	0.3
	1 HP		0.3	1.0	0.3
28. Connect storage cask to cask transporter.	2 Ops	0.5	0.5	1.0	1.0
	1 HP		0.5	1.0	0.5
29. Raise storage cask.	2 Ops	0.2	0.2	1.0	0.4
	1 HP		0.2	1.0	0.2
30. Transport storage cask to storage pad.	2 Ops	2.0	2.0	1.0	4.0
	1 HP		2.0	1.0	2.0

TABLE 7.4-2 (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)
31. Position and lower storage cask on pad.	2 Ops 1 HP	0.5	0.5 0.5	10.0 5.0	10.0 2.5
32. Disconnect storage cask from transporter and remove storage cask lifting eyes.	2 Ops 1 HP	1.0	1.0 1.0	15.2 / 10.0 5.0	30.4 / 20.0 5.0
33. Connect cask temperature instrumentation.	2 Ops 1 HP	0.5	0.5 0.5	10.0 5.0	10.0 2.5
34. Perform cask operability tests.	1 Op 1 HP	48	1.0 1.0	10.0 5.0	10.0 5.0
TOTAL					306.1 / 182.9

1. Number of personnel typically includes 2 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.

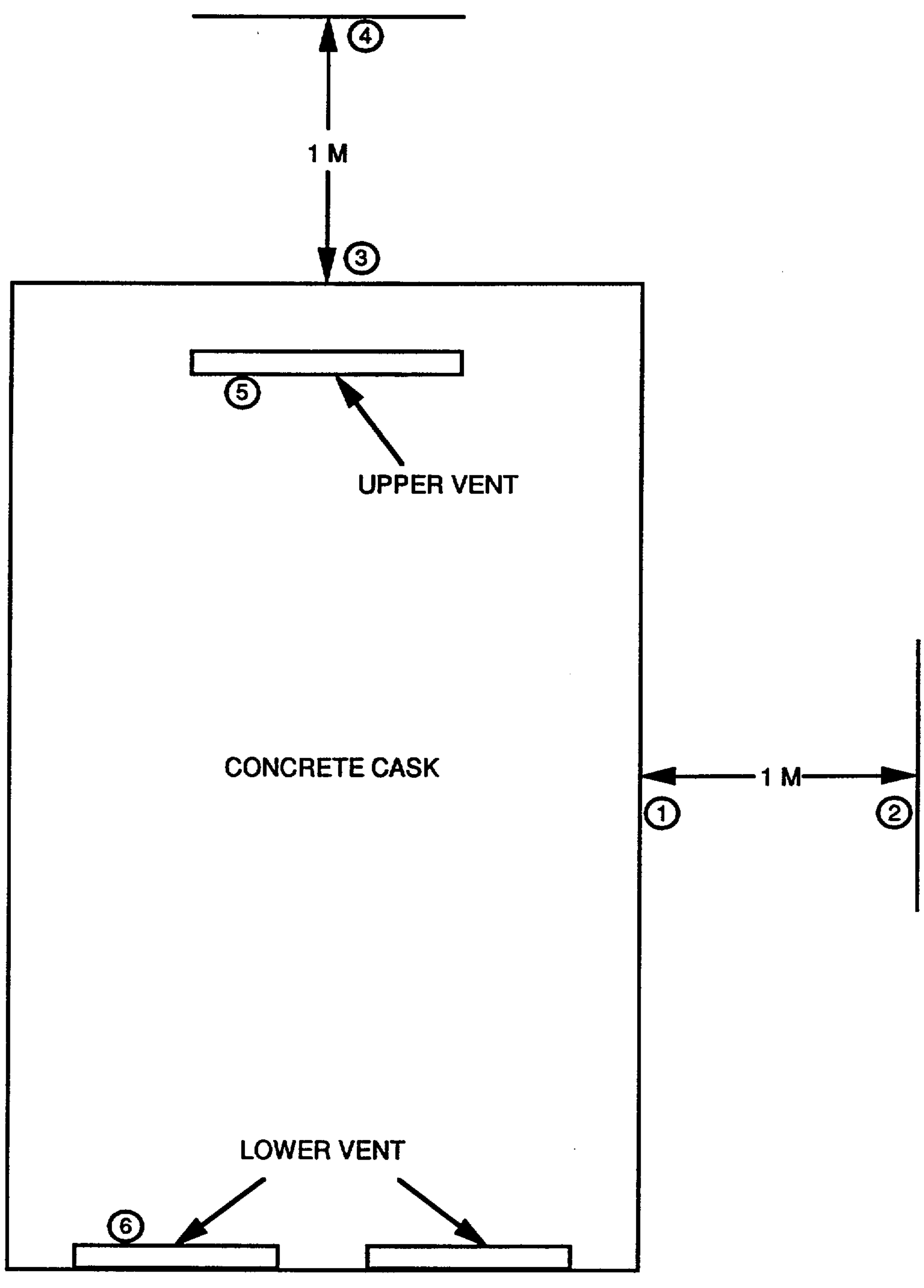

Figure 7.3-1

STORAGE CASK CALCULATED DOSE RATE LOCATIONS

PRIVATE FUEL STORAGE FACILITY
SAFETY ANALYSIS REPORT

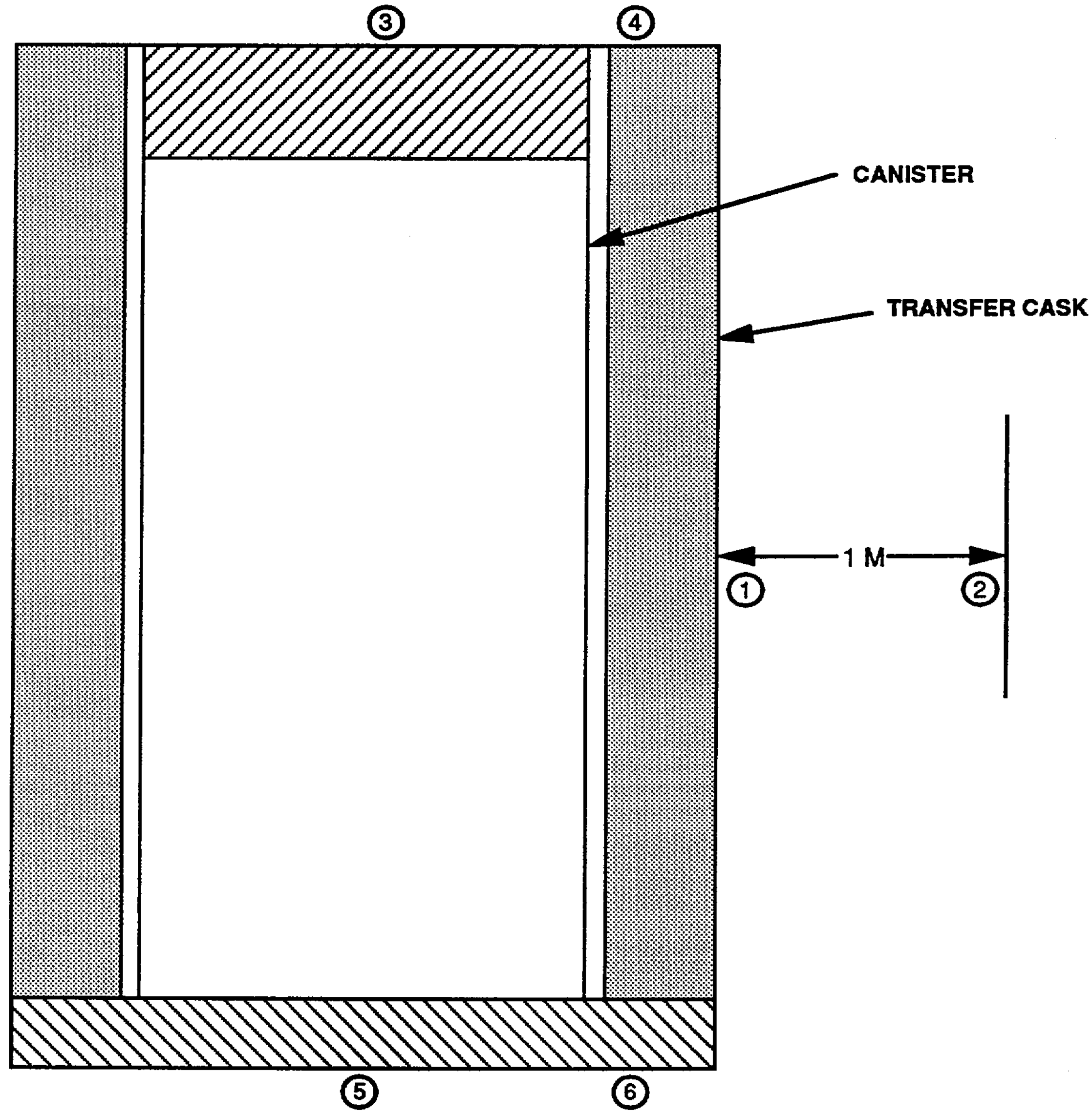

Figure 7.3-2

TRANSFER CASK CALCULATED DOSE RATE LOCATIONS

PRIVATE FUEL STORAGE FACILITY
SAFETY ANALYSIS REPORT

CHAPTER 8

ACCIDENT ANALYSES

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CHAPTER 8

ACCIDENT ANALYSIS

In the preceding chapters, the design and operational features of the PFSF storage and handling systems that are classified as Important to Safety were identified and discussed. This chapter provides a description of the analyses performed for off-normal operating conditions and for a range of hypothetical accidents. The evaluations of off-normal events and accidents demonstrate that the PFSF structures, systems and components (SSCs) classified as Important to Safety are capable of performing their required functions for a wide range of postulated conditions satisfying the requirements of 10 CFR 72.122(b).

ANSI/ANS 57.9 (Reference 1) defines four categories of design events that establish the requirements to satisfy operational and safety criteria. A Design Event I is associated with normal operation. Design Event I conditions are addressed in Chapter 4. A Design Event II is associated with off-normal operations that can be expected to occur with moderate frequency, or on the order of once during a calendar year of PFSF operation. The Design Event II conditions are described in Section 8.1. A Design Event III is associated with infrequent events that could be reasonably expected to occur during the lifetime of the PFSF. These are described in Section 8.2. A Design Event IV is associated with plant-specific design phenomena, including natural phenomena and man-induced low probability events. These are also described in Section 8.2.

The conservative nature of the assumptions and methods used in the analyses of off-normal and accident conditions represent an upper bound for the PFSF design basis events. The analyses demonstrate that the PFSF satisfies the applicable design criteria and regulatory limits. Therefore, the reported values of parameters, such as temperatures and stress levels, envelope the values that would actually be experienced for the various postulated accident conditions.

The results of the off-normal and accident analyses described in this chapter are based on analyses documented in more detail in the HI-STORM 100 Cask System SAR (Reference 2) and the TranStor Storage Cask System SAR (Reference 3).

8.1 OFF-NORMAL OPERATIONS

This section addresses events designated as Design Event II as defined by ANSI/ANS-57.9. The following are considered off-normal events:

- Loss of external electrical power,
- Off-normal ambient temperatures,
- Partial blockage of storage cask air inlet ducts,
- Operator error, and
- Off-normal contamination release.

There is no release of radioactive fission products from inside the canister or abnormal radiation levels associated with these off-normal operations. The only calculated consequence arises from the postulated release of surface contamination from the canister exterior, as discussed in Section 8.1.5. The resultant committed effective dose equivalent (CEDE) and the committed dose equivalent (CDE) to the maximally exposed organ at the Owner Controlled Area (OCA) boundary are shown to be less than 0.1 mrem in Section 8.1.5.3, well below the 10 CFR 72.106 criteria of 5 rem for accidents. Assuming an off-normal condition resulting in release of contamination to the atmosphere occurs on the order of once per year, total annual dose consequences at the OCA boundary from this event and radiation emanating from storage casks (Section 7.6) will not exceed 25 mrem, in accordance with 10 CFR 72.104.

8.1.1 Loss of External Electrical Power

A total loss of external AC electric power is postulated to occur as the result of a disturbance in the offsite electric supply system.

8.1.1.1 Postulated Cause of the Event

Loss of external power to the PFSF may occur as the result of natural phenomena, such as lightning, extreme wind, icing conditions, or as the result of undefined factors causing disturbances in the offsite electric grid that supplies the PFSF.

8.1.1.2 Detection of Event

Loss of the external electric power supply to the PFSF would be detected by noticing a loss of building lighting, loss of function of the powered equipment in the Canister Transfer Building and other equipment not automatically supplied with backup power, or by observation that the security diesel generator has automatically started operation.

8.1.1.3 Analysis of Effects and Consequences

Storage

There are no safety or radiological consequences of this condition because loss of power does not affect the integrity of the canisters and does not result in the release of radioactive material. PFSF spent fuel storage nuclear safety functions do not rely on electrical power for their accomplishment. Heat is removed from the canisters by passive means. None of the systems whose failure could be caused by loss of power are necessary for the continued performance of the PFSF spent fuel storage nuclear safety functions. The storage systems are qualified to safely protect the spent fuel for at least the 40-year PFSF operational life including adequate provisions for transfering decay heat from the spent fuel to the surrounding environment without the need for electrical power.

An emergency diesel-generator is provided as a backup power supply for the security system, designated emergency lighting fixtures, the storage cask temperature monitoring system, and certain other equipment items whose continued energization is desirable.

Canister Transfer Operations

It is postulated that a loss of external electrical power event could occur during the canister transfer operations that are conducted in the PFSF Canister Transfer Building. This could take place at any point in the transfer sequence. Consideration is given to the loss of power: (1) while a loaded shipping cask, with the impact limiters removed, is being unloaded off the heavy haul trailer or rail car; (2) while the canister is being raised from the shipping cask into the transfer cask; (3) while the loaded transfer cask is being moved from above the shipping cask to above the storage cask; and (4) while the canister is being lowered from the transfer cask into the storage cask.

Three lifting devices are used in the shipping cask unloading/loading and canister transfer operations. The 200 ton overhead bridge crane, described in Section 4.7.2, is used to a) move both HI-STAR and TranStor shipping casks to and from the heavy haul trailers or rail cars and canister transfer cells, b) handle the HI-TRAC and TranStor transfer casks, and c) can be used to raise and lower the TranStor canister during canister transfer operations. The 150 ton semi-gantry crane, described in Section 4.7.2, is also used to handle the HI-TRAC and TranStor transfer casks, and to raise and lower the TranStor canister during canister transfer operations. The Holtec canister downloader, described in Section 4.7.3.5.1, is bolted onto the top of the HI-TRAC transfer cask and functions to raise and lower the HI-STORM canister during canister transfer operations. As discussed in Sections 4.7.2.1 and 4.7.3.5.1, the overhead bridge crane, semi-gantry crane, and canister downloader are all designed to meet the criteria for single-failure-proof lifting devices. Whereas the HI-STORM and TranStor transfer operations use the same cranes to handle the shipping casks and transfer casks, they use different methods for raising and lowering the canisters, as discussed in Chapter 5. In the TranStor canister loading operation, the canister is raised from the shipping cask into the transfer cask, and lowered from the transfer cask into the storage cask, by means of the semi-gantry crane or the overhead bridge crane. In the HI-STORM canister transfer operation, the canister is raised and lowered by means of the canister downloader.

The overhead bridge crane and the semi-gantry crane are designed to hold the lifted load in the event of loss of electrical power, with the brakes automatically actuated. The canister downloader is also designed to fail-as-is upon loss of electrical power or loss of hydraulic pressure, with two redundant sets of anti-drop cam locks. Therefore, a load drop accident during canister transfer operations due to loss of electrical power is not a credible event.

With a canister loaded into a transfer cask, a loss of electrical power will delay the transfer operation but will not challenge the integrity of the canister or safe storage of the spent fuel in the canister. There are no safety concerns associated with storage of a canister in its transfer cask until electrical power is restored and the canister transfer operation can resume. The transfer casks are designed to provide adequate shielding and decay heat removal from the canisters. The sides of the HI-TRAC and TranStor transfer casks have both gamma and neutron shields, and the thick bottom shield doors are designed to prevent excessive dose rates below the transfer casks, as shown in Chapter 5 of the HI-STORM and TranStor SARs. In the event the transfer operation is interrupted due to loss of external electrical power, operators would take measures as necessary to assure adequate distance and/or additional shielding between themselves and the transfer casks to minimize doses until such time as electrical power is restored and the transfer process can resume. The overhead bridge crane, semi-gantry crane and canister downloader are all capable of supporting their rated loads indefinitely without electrical power. Thermal analyses discussed in Chapter 4 of the HI-STORM and the TranStor SARs indicate that under steady state conditions with a canister stored in a transfer cask, the temperatures of the fuel, canister, and transfer cask components are within allowable limits.

8.1.1.4 Corrective Actions

Following a loss of electric power to the PFSF, radiation protection personnel would take necessary measures to maintain exposures to personnel in the vicinity of halted canister transfer operations as low as is reasonably achievable (ALARA). Utility repair personnel would be informed and would restore service by conventional means. Such an operation is routine for utility personnel.

8.1.2 Off-Normal Ambient Temperatures

Performance of the storage casks has been conservatively evaluated assuming abnormally high ambient temperatures of sufficient duration for the storage systems to reach steady-state conditions.

8.1.2.1 Postulated Cause of the Event

In order to bound expected steady-state temperatures of the storage system during periods of abnormally high temperatures, analyses were performed by the storage system vendors to calculate the steady-state temperatures for the storage cask, canister, and fuel for a continuous 100°F ambient condition with solar insolation. The design basis spent fuel decay heat generation rates were used for these analyses. The postulated 100°F ambient condition bounds the average daily (day/night) maximum temperature of 93.2°F for cities in Utah (Reference 4). Since it would take 4 to 5 days for the storage systems to achieve steady-state thermal conditions, component temperatures resulting from the constant 100°F off-normal event with solar insolation bound those associated with the 93.2°F average daily maximum temperature condition.

8.1.2.2 Detection of Event

High ambient temperatures would be detected by normal weather monitoring and/or by evaluation of data from the storage cask temperature monitoring system. However, detection of off-normal ambient temperatures is not critical because there are no consequences, i.e., the storage system is designed to withstand such conditions.

8.1.2.3 Analysis of Effects and Consequences

Analyses have been performed for the HI-STORM and TranStor storage systems, assuming a continuous ambient temperature of 100°F for a sufficient duration to allow the system to achieve thermal equilibrium and design basis fuel with maximum decay heat. The analyses were performed using the ANSYS computer program (described in Chapter 4 of the vendors' SARs). The HI-STORM and TranStor SARs (Chapters 4 and 11 of References 2 and 3, respectively) provide the detailed temperature analyses for the off-normal ambient temperature condition.

The maximum steady-state temperatures of key storage system components for both vendors are provided in Table 8.1-1. As discussed in the HI-STORM and TranStor SARs, the component temperatures are all within the vendor temperature limits. The canister and storage cask temperatures pose no threat of fuel cladding failure, canister breach, or reduction in shielding provided by the storage cask.

8.1.2.4 Corrective Actions

The HI-STORM and TranStor storage systems are designed to accommodate component steady state temperatures that would result from continuous exposure to an ambient temperature of 100°F, and no corrective actions are required.

8.1.3 Partial Blockage of Storage Cask Air Inlet Ducts

A complete blockage of one-half of the air inlet ducts is postulated for this event. Both storage systems have four air inlet ducts located at or near the base of the storage casks, so this event considers complete blockage of two air inlet ducts.

8.1.3.1 Postulated Cause of the Event

The air inlet ducts are protected from incursion of foreign objects by screens. The HI-STORM storage cask has four air inlets, oriented 90° apart. The TranStor storage cask has four air inlets, with two located on opposing sides of the cask. Events such as high winds, tornado and heavy snow could potentially cause partial duct blockage. Significant duct blockage would be detected by the storage cask temperature monitoring system periodic surveillance and be removed before achieving the steady state temperatures considered in the vendor analyses. This scenario demonstrates the inherent thermal margin and stability of the storage systems.

8.1.3.2 Detection of Event

Temperatures representative of the thermal performance of each storage cask are remotely monitored by the storage cask temperature monitoring system and trended. Increased temperatures indicate possible blockage of the natural convection air flow path, most likely at the air inlet ducts, and personnel are dispatched to inspect storage casks with high temperatures. Also, quarterly surveillances consisting of visual inspections are performed for the purpose of detecting any blockage of the storage cask inlet and outlet ducts. Should blockage occur, it will be identified and removed in a timely manner.

8.1.3.3 Analysis of Effects and Consequences

Results of the analyses of the postulated 50 percent blockage condition are included in the HI-STORM and TranStor SARs (Chapter 11 of References 2 and 3, respectively). The maximum steady state temperatures of storage system components are provided in Table 8.1-2. As discussed in the HI-STORM and TranStor SARs, the component temperatures are all within the vendor temperature limits.

8.1.3.4 Corrective Actions

Upon receiving indication of high storage cask(s) temperatures, PFSF personnel will inspect the affected cask(s) ducts for blockage. Once an obstruction has been identified, PFSF personnel will remove the debris or other foreign material blocking the ducts. Since screening is provided for all air inlets, material blocking inlet ducts is expected to be on the outside and may be removed by hand or hand-held tools. Dose rates at the air inlets are higher than the nominal dose rates at the storage cask walls, so a worker clearing the vents will be subject to above-normal dose rates. As a worst case estimate, it is assumed that a worker kneeling with hands on the vent inlets requires up to 30 minutes to clear the vents. Assuming the highest dose rates associated with a storage cask containing design fuel (Tables 7.3-1 and 7.3-2), a worker could accrue approximately 35 mrem to the hands and forearms and approximately 25 mrem to the chest and body from the storage cask with blockage and from adjacent casks.

8.1.4 Operator Error

This event consists of off-normal operator load handling errors that develop from the canister impacting against the inside of the shipping, transfer, or storage cask.

8.1.4.1 Postulated Cause of the Event

Several postulated events involving off-normal handling have been considered, all caused by personnel error. Load drops by the overhead bridge crane, the semi-gantry crane, or the canister downloader are not considered credible because of the single-failure-proof design of these lifting systems. Postulated events are: (1) while lifting the canister out of the shipping cask and into the transfer cask, personnel error could result in lifting the canister too high so it contacts the top of the transfer cask; (2) during placement of the canister into the storage cask, improper operation of the crane or canister downloader may cause a lateral impact against the inside of the storage cask (this could also occur during transfer of the storage cask to a storage pad, where an inadvertent movement could cause lateral impact of the canister against the inside of the storage cask); and (3) during canister lowering into the storage cask with the transfer cask improperly aligned with the storage cask, the canister could encounter interference, such as catching on the edge of the storage cask.

8.1.4.2 Detection of Event

The off-normal handling event would be detected by facility operators and personnel monitoring canister transfer operations or storage cask movement from the Canister Transfer Building to a storage pad. Audible noises would be heard from the canister impacting a cask, and slackening of the slings that connect the canister to the crane hook or to the canister downloader would be observed.

8.1.4.3 Analysis of Effects and Consequences

Off-normal handling events are evaluated in the HI-STORM and TranStor SARs. The following is a summary of the evaluations of the different credible off-normal handling events.

Canister Contacts Top of Transfer Cask

In the TranStor canister transfer operation, either the overhead bridge crane or the semi-gantry crane is used to lift the canister out of the shipping cask into the transfer cask, whereas in the HI-STORM canister transfer operation the canister downloader is used to lift the canister into the transfer cask, as described in Chapter 5. This evaluation considers the possibility of lifting the canister into the top of the transfer cask and continuing to lift the canister and the transfer cask, which could possibly occur with the TranStor transfer operation. This is not a concern with the HI-STORM transfer operation, since the canister downloader is bolted to the top of the transfer cask.

It is assumed that the crane operator fails to stop the crane while lifting the TranStor canister into the transfer cask. This event would be detected by audible noise as the canister contacts the transfer cask, and/or by upward movement of the TranStor transfer cask. The worst case loading for this condition occurs when the canister starts to lift one side of the transfer cask. For the TranStor system, the arrangement of 8 lifting points (hoist rings and lifting slings) is redundant (see Section 4.7.4.5.1), i.e., only 4 points are needed to carry the load. It is conservatively assumed that only two lifting points out of the eight carry one-half the weight of the transfer cask (total weight = 126.6 kips) with the maximum canister load (84.5 kips) evenly distributed among the eight lifting points. The resulting load on each of the two lifting points, partially lifting the transfer cask, is 42.2 kips. The hoist rings that connect the canister to the crane hook have a manufacturer rated capacity of 24 kips with a safety factor of 5. Therefore, the breaking load for a ring is 120 kips (24 X 5) in any direction. This bounds the 42.2 kips loading postulated for this off-normal event by a factor of 2.8, which is acceptable for this off-normal occurrence.

The top annular cover plate of the TranStor transfer cask and its bolts were also reviewed to determine whether these have sufficient strength to support the weight of the transfer cask, since an inadvertent canister lift would result in lifting the entire transfer cask by the cover plate. Since this would not be a normal lift but rather an accident condition, the NUREG-0612 safety factors do not apply and the allowable stresses used are from the AISC Manual for the design of this component. The stresses on the inner and outer edges of the cover plate were calculated and compared to AISC allowables. The load on a single transfer cask lid bolt from the inadvertent canister lift consists of tension required to balance the transfer cask weight plus a prying force that results from the applied moment. Both of these forces were calculated and added together before comparison to AISC allowables. As shown in the TranStor SAR (Reference 3, Table 3.4-1), the cover plate stresses and bolt forces were determined to be within AISC allowables.

Horizontal Impacts of the Canister

The horizontal impact of the canister event assumes that the canister impacts the side of the storage cask at a speed of 2 ft/sec, which is equivalent to a drop from a height of 0.75 inch. The resulting deceleration is conservatively calculated to be 17.5 g for the TranStor system (Section 11.1.5 of Reference 3). This acceleration is bounded by those determined for the canisters in drop accidents. Therefore, the associated stresses resulting from this accidental impact are bounded by those for design basis drop accidents. Canister accelerations analyzed due to postulated side drop/tipover accidents are 45 g for HI-STORM (Reference 2) and 44 g for TranStor (Reference 5). Tables 11.1-1 (PWR) and 11.1-2 (BWR) of the TranStor SAR identify stresses in various canister components resulting from the 17.5 g acceleration, compare these with

ASME Boiler and Pressure Vessel Code (BPVC) Section III allowable stresses, and demonstrate that stresses from this postulated event are below allowables.

Interference During Canister Lowering Operations

The interference during canister lowering operations event postulates that the canister impacts the storage cask edge or side while the canister is lowered into the storage cask. Procedures to ensure alignment of the transfer cask with the storage cask should prevent this condition from occurring, but it is assumed that operator error results in inadequate clearance / misalignment. Since the only force acting on the canister during lowering is gravity, the worst case condition would be a load of 1 g on the canister bottom or side, if it were completely supported by the interference. The stresses applied to the canister in this scenario are again bounded by those assessed for the canister in drop accidents, analyzed in the HI-STORM SAR and the TranStor shipping cask SAR (Reference 5). The analyses determined that the canister vessel and its internals would maintain their structural integrity and continue to perform their safety functions for the drop accidents.

8.1.4.4 Corrective Actions

In the case of interference during canister lifting, the crane operator (TranStor) or canister downloader operator (HI-STORM) lowers the canister. Workers would inspect the alignment of the transfer cask on the shipping cask, make necessary adjustments, and complete the lift. If unable to satisfactorily correct the situation, workers would lower the canister back to the bottom of the shipping cask, lift the transfer cask off the shipping cask, and determine the cause of any interference/misalignment.

In the horizontal impact scenario, the canister is designed to withstand horizontal acceleration loads that bound the canister horizontal impacts on the storage cask discussed above. No corrective actions are necessary.

To recover from interference during the canister lowering situation, the crane or canister downloader operator would immediately stop lowering the canister, inspect the area for interference, and raise the canister back into the transfer cask. The personnel involved in the transfer operation would check the alignment of the transfer cask on the storage cask. If necessary, the transfer cask will be lifted off the storage cask to permit inspection for foreign objects.

8.1.5 Off-Normal Contamination Release

This event involves the postulated release of surface contamination from the exterior of a canister to the environment.

8.1.5.1 Postulated Cause of the Event

The canister may become slightly contaminated during loading operations of the spent fuel into the canister at the originating nuclear power plant. If this contamination is not detected and removed prior to shipment to the PFSF, it is possible for an impact of the canister to dislodge some of the removable surface activity resulting in a release to the atmosphere.

Contamination of the canister at the originating nuclear power plant is unlikely since during wet loading of fuel the canister is contained within a transfer cask when in the spent fuel pool and the annulus between the canister and the transfer cask is filled with clean water (or filtered pool water). The annulus is either sealed at the top or clean water is continuously pumped into the bottom of the annulus, causing clean water to overflow out of the top, thereby preventing entry of unfiltered contaminated pool water into the annulus. For dry transfers, it is less likely for contamination to occur since the canister is not inserted into a spent fuel pool. Surveys are performed at the originating nuclear power plant to assess removable contamination levels on the outside of the canister, and canisters having removable contamination levels in excess of specified limits are not permitted to be shipped to the PFSF. Upon receipt of canisters at the PFSF, accessible canister surfaces (canister lid and side several inches below lid) are again surveyed for removable contamination, and canisters having removable contamination levels in excess of specified limits (Section 10.2.2.1) are required to be returned to the originating nuclear power plant for decontamination.

8.1.5.2 Detection of Event

A release of some removable activity from the exterior surface of the canister could possibly occur as the result of impacts during the canister transfer operation. Significant impact of the canister during transfer operations would be observed by personnel associated with the transfer operation, which includes health physics coverage that would detect an activity release.

8.1.5.3 Analysis of Effects and Consequences

The following assesses the effects of postulated release of contamination from the external surfaces of a canister, conservatively assuming removable contamination levels of 1 E-4 $\mu Ci/cm^2$ (22,200 dpm/100 cm^2) over the entire external surface area of a canister, much higher than is anticipated for canisters received at the PFSF and slightly above the removable surface contamination limit for accessible canister surfaces specified in Section 10.2.2.1 (22,000 dpm/100 cm^2) for beta/gamma activity. It is conservatively assumed that an event causes 100 percent of the canister external surface contamination to be released to the atmosphere.

Assuming the contamination is Co-60 particulate activity evenly distributed at a concentration of 1 E-4 $\mu Ci/cm^2$ over the entire external surface of a HI-STORM canister (TranStor canister has a smaller area), with a surface area of approximately 312,000 cm^2 there would be a total activity inventory of approximately 31.2 μCi. The nearest distance from a storage pad to the OCA fence (site area boundary) is 646 meters, and the nearest distance from the Canister Transfer Building to the OCA fence is 500 meters. A χ/Q of 1.94 E-3 sec/cubic meter was calculated in accordance with Regulatory Guide 1.145 (Reference 6), assuming a distance of 500 meters to the dose receptor, a wind speed of 1 meter/sec, atmospheric stability class F, with no consideration for plume meander.

The dose conversion factor for intake of Co-60 is specified in EPA Federal Guidance Report No. 11 (Reference 7) as a committed effective dose equivalent (CEDE) of 5.91 E-8 Sv/Bq, equal to 219 mrem/μCi. The highest dose conversion factor for committed dose equivalent (CDE) to any organ from Co-60 is that for the lungs, 3.45 E-7 Sv/Bq, equal to 1,277 mrem/μCi. An adult breathing rate of 3.3 E-4 cubic meters per second is assumed in accordance with Reference 7. Assuming an individual is located within the plume 500 meters from the release point for the duration of the release, the individual would receive a CEDE of 4.4 E-3 mrem and a CDE to the lungs of 2.6 E-2 mrem. These doses are well below the 10 CFR 72.106 criteria of 5 rem for accidents. Assuming an off-normal condition resulting in release of contamination to the atmosphere occurs on the order of once per year, total annual dose consequences at the OCA boundary from this event and radiation emanating from storage casks (Section 7.6) will not exceed 25 mrem, in accordance with 10 CFR 72.104.

Onsite personnel located 150 meters from the release point would receive a CEDE of 0.03 mrem and a CDE to the lungs of 0.2 mrem, using the same assumptions noted above except for a calculated χ/Q of 1.40 E-2 sec/cubic meter.

8.1.5.4 Corrective Actions

Even if relatively high levels of contamination are encountered on the external surfaces of a canister, which is not anticipated, no corrective action is necessary. Doses at the OCA fence resulting from release of activity from a contaminated canister would be negligible.

8.2 ACCIDENTS

Design events of the third and fourth types as defined in ANSI/ANS-57.9 are considered in this section. A Design Event III consists of those infrequent events that could reasonably be expected to occur during the lifetime of the PFSF. A Design Event IV consists of natural phenomena and human-induced low probability events that are postulated because their consequences may result in the maximum potential impact on the immediate environs but are not necessarily credible. Hypothetical accidents, which are analyzed in this section, are also considered as Design Event IV. Their consideration establishes a conservative design basis for SSCs classified as important-to-safety.

The following accident or class III and IV design events are considered in this chapter:

- Earthquake,
- Extreme wind,
- Flood,
- Explosion,
- Fire,
- Hypothetical storage cask drop / tip-over,
- Hypothetical loss of confinement barrier,
- 100% blockage of air inlet ducts,
- Lightning, and
- Hypothetical accident pressurization.

Each of these accidents are described in the following sections. These evaluations show that the release of radioactive material is controlled in compliance with 10 CFR 72.106 and 72.126(d).

8.2.1 Earthquake

An earthquake is classified as a natural phenomenon Design Event IV as defined in ANSI/ANS-57.9.

8.2.1.1 Cause of Accident

Earthquakes are associated with faults in the upper crust of the earth's surface. Earthquake intensities and associated ground motions are based on historical data and are contained in maps and tables as referenced in Section 2.6. The PFSF is located west of the Rocky Mountain Front (approximately 104° west longitude) as described in 10CFR Part 72.102 and the site area has the potential for seismic activity. Consequently, the site has been evaluated for geological and seismological characteristics to determine the appropriate seismic design criteria identified as the ISFSI Design Earthquake (DE). SSCs classified as Important to Safety are required to be designed to resist the effects of seismic ground motions due to the DE in accordance with the requirements of 10CFR 72.122(b).

The storage system structural design bases, which identifies earthquake loads and the structural design of the storage systems, are contained in Section 4.2.1.5.1 (H) for HI-STORM and Section 4.2.2.5.1 (H) for TranStor.

8.2.1.2 Accident Analysis

The HI-STORM and TranStor storage casks are analyzed for a generic DE as selected by each cask vendor and as described in their respective SARs. The HI-STORM and TranStor storage casks are also analyzed for a PFSF site specific DE, which is based on a seismological evaluation of the siting area. The site specific DE is discussed in Section 3.2.10 and is represented by response spectrum curves developed specifically for the site with a zero period acceleration of 0.67 g horizontal (two directions) and 0.69 g vertical. Both the HI-STORM and TranStor storage casks are analyzed for these conditions by the vendors to assure structural strength of the cask and cask stability. Cask stability analyses provide assurance that the casks will not tip over or slide excessively in an earthquake.

In addition to the vendor's PFSF site specific cask stability analyses, a separate and independent site specific cask stability analysis was performed by a structural-mechanical engineering consultant specializing in seismic dynamic analysis of equipment and structures. The analysis was performed by J. D. Stevenson, Consulting Engineer, for the purpose of independently confirming the cask stability conclusions of the vendor's analyses. This bounding case analysis considered both the HI-STORM and TranStor storage casks. The analysis demonstrates the storage casks will not tip over or slide excessively in an earthquake and confirms the conclusions of the vendors.

A summary of the vendor's cask stability analyses and the independent cask stability analyses performed by J. D. Stevenson, Consulting Engineer, are as follows:

<u>HI-STORM Cask Stability Analysis</u>
The HI-STORM generic seismic cask stability analysis is described in Section 3.4.7 of the HI-STORM SAR. The analysis demonstrates the HI-STORM storage cask will not tip over or slide excessively when subjected to a seismic event characterized by Regulatory Guide 1.60 response spectra curves with a zero period acceleration of 0.80 g in three orthogonal directions. The cask is allowed to rock, rim, and slide during the event. The results show the cask centroid projection on the pad remains inside of the initial contact envelope and is therefore within the stability criteria established by the cask vendor.

The generic analysis determined that inertia loads produced by the seismic event are less than the 45 g loads due to the postulated HI-STORM vertical drop and non-mechanistic tipover events (Section 8.2.6). Stresses in the canister due to the seismic event are bounded by the 45 g deceleration load from the hypothetical end drop and side drop events described in Section 3.4.4.3.1 of the HI-STORM SAR. As discussed in Appendix 3.B of the HI-STORM SAR, shell and concrete stresses in the storage cask resulting from the seismic shear and moment forces are evaluated and determined to be acceptable.

The generic cask stability analysis in the HI-STORM SAR assumes a single cask resting on a rigid surface. The analysis does not consider soil-structure interaction, which will affect the dynamic properties and seismic response of the structural system. In order to verify the cask stability under site specific conditions, a site specific cask stability analysis was performed by the cask vendor (Reference 8).

The HI-STORM storage cask was analyzed using proprietary qualified software for the site specific seismic event characterized by site specific response curves with a zero period acceleration of 0.67 g in both horizontal directions and 0.69 g in the vertical direction. The analysis considered soil-structure interaction, actual storage pad size, and a variety of cask placements on the pad.

The site specific cask stability analysis was performed by developing three statistically independent acceleration time histories from the site specific response spectra. This seismic input was applied three-dimensionally to the structural system model, which included the storage pad, soil springs, and various cask placements to determine the worst case response. The site specific seismic analysis employs the finite element modeling of the cask geometry and boundary conditions and numerical integration of the dynamic equation.

The casks are modeled as a two body system with each overpack described by six degrees of freedom which captures the inertial rigid body motion of the overpack. Within each overpack the internal MPC is modeled by an additional five degree of freedom mass which is sufficient to define all but the rotational motion of the MPC about its own longitudinal axis, a motion which is of no significance in this analysis. Spring constants are developed to simulate the contact stiffness between the MPC and the overpack cavity. Interface spring constants are developed for the overpack-to-concrete pad linear compression only contact springs and for the associated friction springs at each of the 36 contact locations for each overpack on the pad.

Soil-structure interaction is incorporated into the model by the development of soil springs to reflect the characteristics of the underlying soil mass beneath the pad. Horizontal, vertical, rocking and torsional spring rates were calculated along with appropriate soil mass and damping values and applied at the pad-soil interface.

The cask stability analysis was performed by computer methods using cask-to-pad coefficients of friction equal to 0.2 (emphasizes sliding potential) and 0.8 (emphasizes tipping potential) to bound the maximum sliding and tipping behavior of the cask. The results of the site-specific analysis show that the storage casks will not tip over or slide to the extent of impacting adjacent casks during the site-specific DE.

For the limiting case with a 0.8 coefficient of friction (maximum tip), there is minimal rotation of the cask vertical centerline, except the case with a single cask on the pad. For this case, there is evidence the cask edges and rolls to a new position approximately 5 inches from its original position during the seismic event. The maximum tip, identified as the lateral motion of the cask top center point from its initial position, is approximately 13 inches for the single cask case and considerably less for all other cases. For the limiting case bounded by a 0.2 coefficient of friction (maximum slide), the maximum distance in which a cask will slide is shown to be 10.25 inches.

For both coefficients of friction considered, cask motions are generally in-phase with each other. The casks are spaced on the storage pad at 15 ft center-to-center which provides 47.5 inches clear between casks (cask diameter is 132.5 inches) and provides a considerable margin of safety against impacts between casks during a seismic event. The analysis also shows that for those cases where the cask(s) tip up slightly and then "slap" back down on their base, vertical deceleration forces, up to 8.43 g, are developed, but these values are well below the design basis deceleration forces for the HI-STORM 100 system.

The site specific cask stability analysis performed by the cask vendor demonstrates that the HI-STORM storage cask will not tip over in a seismic event. The calculated cask movements are much less than the cask spacing on the storage pad and as such, the storage casks are shown not to impact one another or move off of the storage pad in a seismic event. Therefore, no radioactive material would be released from the storage system when subjected to the DE. The HI-STORM storage system thus meets the general design criteria of 10CFR 72.122(b), as it relates to earthquakes.

TranStor Cask Stability Analysis

The TranStor generic cask stability analysis is described in Section 11.2.5 of the TranStor SAR. The analysis demonstrates the storage cask will not tip over when subjected to a seismic event characterized by Regulatory Guide 1.60 response spectra curves with a zero period acceleration of 0.75 g in two horizontal directions and 0.50 g in the vertical direction. The generic cask stability analysis is performed using a two-dimensional analysis, and as such, utilizes a horizontal response spectrum with a resultant zero period acceleration of 0.8 g and a vertical response spectrum with a zero period acceleration of 0.5 g. The horizontal response spectrum is the resultant of both 0.75 g horizontal accelerations combined using the 100-40 rule allowed by NUREG/CR-0098 (Reference 9). The analysis also concludes that with a maximum ground displacement of 20.0 inches (substantially less than 44 inches of clear space between the casks), sliding would not cause impact between adjacent casks.

The generic cask stability analysis in the TranStor SAR assumes a single cask resting on a rigid surface. The analysis does not consider soil-structure interaction, which will affect the dynamic properties and seismic response of the structural system. In order to verify the cask stability under actual conditions, a site specific cask stability analysis was performed by the cask vendor.

The TranStor storage cask was analyzed for the site specific seismic event characterized by site specific response curves with zero period accelerations of 0.67 g in both horizontal directions and 0.69 g in the vertical direction. The analysis considered soil-structure interaction, actual storage pad size, and a variety of cask placements on the pad.

The site specific cask stability analysis was performed using the ANSYS finite element program (discussed in TranStor SAR Section 11.2.5.2) and time history analysis. The time history is developed from the free-field seismic response spectra generated for the PFSF site. However, one of the important factors that must be considered when defining the structural response to an earthquake is that the ground input at the base of the foundation may be influenced by the presence of the structure itself. From the dynamics viewpoint, the seismic response of a dynamic system (free field soil) may be altered due to both inertial and kinematic interaction when another system (structure) is present. As a result, the motion at the structure base may be different from the free field motion. The soil-structure interaction effects are discussed in ASCE 4 (Reference 10), Section 3.3 and the Commentary. The problem can be analyzed as two separate sub-problems: 1) analysis of soil-structure interaction to determine the seismic input at the cask base, and 2) analysis of the cask stability under that excitation.

To take into account soil-structure interaction at the PFSF site, the SUPER SASSI/PC computer program (System for Analysis of Soil-Structure Interaction) (Reference 11) is used. This program is a specialized finite-element code originally developed at UC Berkeley by Prof. J. Lysmer and modified into a PC version by Stevenson and Associates. The analysis methodology is as follows:

1. Model the structure and soil in three dimensions. Embedment of the structure is adequately represented using 3-D solid elements. The casks are represented as rigid bodies because their natural frequencies have been shown to be well beyond the seismic cutoff of 33 Hz. The site specific soil properties (established by testing) are used for the top 120 feet and an elastic half-space is modeled below that level.

2. Develop free field vertical and two horizontal time histories which envelope the provided response spectra. The duration of 20 seconds has been selected for time histories used in this evaluation. Power Spectral Density (PSD) distribution is also evaluated to assure that the energy is reasonably distributed across the spectrum. All time histories are generated to be statistically independent.

3. Provide the generated free field time histories and the soil-structure model as input for SUPER SASSI/PC.

4. Obtain translational time histories at the base of the storage cask. These time histories are used in the cask tip over analysis.

5. Repeat the procedure for different pad loading patterns to determine the configuration that produces the highest input at the cask base. One, four, and eight cask patterns are considered.

The stability of the TranStor storage cask during the DE is evaluated using the ANSYS finite element program and non-linear time history analysis. Non-linearity of the system results from the fact that the storage pad is capable of producing a reaction in only one direction (upward). No hold-down restraint is available to prevent the cask from rocking. Non-linear gap elements are used in the finite element model to represent this condition. The most unfavorable cask geometry (highest center of gravity location) is conservatively used. The highest vertical and horizontal components from the different pad loading configurations are taken for the cask stability analysis.

The cask ANSYS model is the same as used to model the generic seismic event in the TranStor SAR. Since the model is two-dimensional, i.e. only one horizontal and one vertical excitation is applied, the horizontal time history must represent the geometrical sum of two independent horizontal components. This is accomplished by using the 100-40 rule allowed by NUREG/CR-0098 (Reference 9). According to this rule, due to statistical independence of two motions, the structure can be designed to take the combined effects of 100% in one particular direction and 40% of the effects corresponding to the other direction. As a result, the ZPA for the horizontal component used in the analysis is increased by the factor of $\sqrt{(1^2 + 0.4^2)} = 1.08$.

Results of the calculation demonstrate that the cask is stable under the loads of the site specific DE. Furthermore, a vertical ground displacement of approximately 5.6 feet would be required to move the center of gravity over the corner of the cask so that the cask would topple. This type of ground displacement and/or failure of the foundation is considered to be unrealistic and, hence, it is concluded that in addition to not toppling due to the kinetic energy of the earthquake, the cask will also not topple due to permanent failure and vertical movement of the foundation. Therefore, based on this analysis, it is concluded that the cask will not tip over during a seismic event.

The canister, its internals, and the concrete storage cask are very rugged and, since tip over is precluded, stresses due to the DE are relatively minor and bounded by the 17.5 g load during the off-normal handling event (Section 8.1.4.3). Section 11.1.5 of the TranStor SAR determined that stresses in the canister resulting from this event are within allowable limits. Therefore, the TranStor canister will not breach or suffer damage in the event of the DE.

Shell and concrete stresses in the storage cask from the seismic shear and moment forces are bounded by those resulting from the 19.8 g deceleration load from the hypothetical TranStor cask tip over analysis discussed in Section 8.2.6. Loads on the storage cask resulting from the DE are included in the load combinations discussed in Chapter 4, where it is shown that stresses resulting from required load combinations are within allowable stress capacities.

The seismic analyses show that the storage cask will not tip over and no damage will be sustained by either the canisters, their internals, or the storage casks in the event of an earthquake. Therefore, no radioactive material would be released from the storage system when subjected to the DE. The TranStor storage system thus meets the general design criteria of 10CFR 72.122(b), as it relates to earthquakes.

Independent Cask Stability Analysis

An independent cask stability analysis was performed by J. D. Stevenson, Consulting Engineer, for the purpose of confirming the conclusions of the vendors' site specific cask stability analyses. The analysis considered both the HI-STORM and TranStor storage casks to determine a controlling and bounding storage cask configuration. Although both storage casks are similar in overall dimensions and weight, the cask which was selected as bounding for evaluation of seismic stability was the HI-STORM MPC-32 canister. The HI-STORM MPC-32 canister is the heaviest and the tallest loaded canister and storage cask combination with a weight of 356,521 lb. and a center of gravity of 118 in. above the base of the cask.

The cask stability analysis was performed using a two step approach. First, the cask/pad/soil system was modeled using the SUPER SASSI/PC computer program (Reference 11) to include the effects of soil-structure interaction. The results of the SUPER SASSI/PC analysis were then used in a non-linear time-history analysis using the ANSYS (Reference 12) computer program. The ANSYS analysis was for a single cask, considered essentially as a rigid body, evaluated for overturning. Additional rigid body analysis was also considered, as suggested by Housner (Reference 13), to check the effects of cask tip over and sliding as a rigid body.

The independent cask stability analysis utilized the PFSF site specific response spectra curves, having a zero period acceleration of 0.67 g horizontal (two directions) and 0.69 g vertical. The free field ground surface response spectra were used to develop 3 independent synthetic time histories using the SPECTRA (Reference 14) computer program. These time histories were used as input to the SUPER SASSI/PC computer analysis to evaluate the soil-structure interaction. The model included the cask storage pad (3 ft. thick x 30 ft. wide x 64 ft. long) with eight casks in place. The casks were idealized by rigid sticks (beam elements) with translational and rotational inertia concentrated at the cask center of gravity. Rigid links were introduced to simulate the physical sizes of the cask bases. The model was excited by the three acceleration time-histories obtained from SPECTRA. The results of the SUPER SASSI/PC computer analysis were the time-history motions at the base of the cask. Computed results showed that the rocking motions of the storage pad were practically negligible in comparison with the translational motions.

The output time-history motions at the base of the cask were then applied to the non-linear ANSYS analysis using a single cask. Several analyses were performed using

ANSYS to evaluate the potential for the cask to tip during the seismic event as simulated by the storage pad seismic motions. The ANSYS models used included both two-dimensional and three-dimensional beam element models. Contact elements were used to model the interface between the cask and the storage pad. The cask was modeled using rigid beam elements with a single vertical element for the cask. The cask base was modeled with two horizontal elements at the base extending to the outer edges of the cask diameter for the two-dimensional model and multiple horizontal elements at thirty degrees spacing for the three-dimensional model. The mass of the cask was lumped at the cask center of gravity located on the vertical element. The model was loaded with gravity prior to earthquake loading. Friction effects were included using a coefficient of friction of 0.5 between the cask and the pad. The cask tip over analysis was based on a constant friction force (lateral resistance at the base of the cask model) active during the stability analysis which conservatively overestimates the overturning potential of the cask. In addition, some rigid body dynamic analyses were performed to provide insight into the behavior of the ANSYS models, and also to evaluate the sliding effects. Rigid body rocking analysis indicates a maximum angle of cask rotation less than 1.5 degrees. Rigid body sliding response of the cask was calculated and found to be less than one foot.

From the analysis performed, it was determined that the most conservative results (and therefore the greatest overturning response) were obtained with the two-dimensional model with the displacement motion applied only at the center base node of the model. The results of the two-dimensional analyses indicate that the rotation at the base of the cask model is approximately 7.2 degrees, which will cause the storage cask top to move laterally approximately 29.0 inches. For the three-dimensional model, a maximum cask rotation of approximately 1.3 degrees was obtained, which will cause the storage cask top to move laterally approximately 5.2 inches. It should be noted that for the three-dimensional model, two components of displacement loading were applied simultaneously. Whereas, for both the two-dimensional and three-dimensional models, the vertical earthquake effects were included as an acceleration time history.

The site specific cask stability analysis performed by J. D. Stevenson, Consulting Engineer, demonstrates that a bounding storage cask configuration, enveloping both the HI-STORM and TranStor storage casks, will not tip over or slide excessively in an earthquake. The computed tip angle of the cask was 7.2 degrees for the two-dimensional model and 1.3 degrees for the three-dimensional model for the prescribed seismic criteria and soil conditions. Lateral displacements of the top of the cask are well below the point at which the cask would tip over (approximately 28.85 degrees). And, using a coefficient of friction of 0.5, sliding is less than one foot. These cask movements are within the clear distances between casks (47.5 inches) and will preclude impact of adjacent casks.

Canister Transfer Operations

Canister transfer operations are performed in the Canister Transfer Building and described in Chapter 5. The overhead bridge crane, located inside the building, is used to handle the shipping casks and transfer casks of both vendors, and can also be used to handle the TranStor canister. The semi-gantry crane is used to handle the transfer casks of both vendors and the TranStor canister. The canister transfer operations of the two vendors are essentially the same, with the exception that the HI-STORM operation uses the canister downloader and the TranStor operation uses a crane to raise and lower the canister. The canister downloader is a hydraulically powered lifting device that is bolted onto the top of the HI-TRAC transfer cask.

The overhead bridge crane and the semi-gantry crane are designed to withstand the DE, as is the Canister Transfer Building which provides the structural support for the cranes. As discussed in Section 4.7.2, the overhead bridge crane and semi-gantry crane are designed to meet the criteria for single-failure-proof lifting devices. The canister downloader is also a single-failure-proof lifting device (Section 4.7.3.5.1), which is designed to withstand the DE. The overhead bridge crane, semi-gantry crane, and canister downloader are capable of withstanding the DE during the critical lift without toppling or dropping the load. Therefore, the DE will not cause a load drop accident during lifting of either vendor's shipping cask, transfer cask, or a canister.

As discussed in Chapter 5, the overhead bridge crane lifts either a HI-STAR or a TranStor shipping cask off the heavy haul trailer or rail car and moves it into one of the canister transfer cells, where it is placed upright on its bottom end in preparation for the canister transfer operation. Prior to disconnecting the crane and unbolting the lid, the shipping cask is secured in place by attaching seismic support struts between the cask and the transfer cell walls. The seismic support struts are designed to resist forces resulting from the DE and maintain the shipping cask in its upright position. Once the lid is unbolted and removed, the canister is accessible through the top of the shipping cask. Canister lifting attachments and hoist slings are installed on the canister lid and the transfer cask placed onto the shipping cask by means of the overhead bridge crane or semi-gantry crane. This stacked cask configuration that occurs during the canister transfer process, with a transfer cask resting on either a shipping or storage cask, was evaluated for stability in the event of a DE (see Section 4.7.4.5.1).

In the HI-STORM transfer operation, the HI-TRAC transfer cask, resting on either the HI-STAR shipping cask or the HI-STORM storage cask, is connected to either the overhead bridge crane or the semi-gantry crane throughout the transfer operation. Continuous connection of the crane to the transfer cask provides assurance that the transfer cask cannot topple in the event of forces associated with a DE.

In the TranStor transfer operation, during the stacked cask portion of the operation, the transfer cask is disconnected from the crane when the crane is used to lift the canister out of the shipping cask into the transfer cask, and when the crane is used to lower the canister from the transfer cask into the storage cask, as described in Chapter 5. In order to assure cask stability in the event of an earthquake, the crane is not disconnected from the transfer cask until seismic support struts are attached to the transfer cask, as discussed in Section 4.7.4.5.1. The seismic support struts are physically connected to the walls of the transfer cell and are designed to resist forces resulting from the DE and maintain the transfer cask in its upright position. Therefore, the stacked cask configuration of each vendor is stable and will withstand the forces associated with a DE without a drop accident.

8.2.1.3 Accident Dose Calculations

The DE is not capable of damaging the canisters or storage casks during canister storage operations. The Canister Transfer Building structure is designed to withstand the DE. Additionally, the overhead bridge crane, semi-gantry crane, and canister downloader are designed to comply with the single-failure-proof criteria, which requires them to withstand a DE with the maximum critical load in the lifted position during the seismic event, without dropping the load (Section 3.2.10.2.10). No radioactivity would be released in the event of an earthquake and there would be no resultant dose.

8.2.2 Extreme Wind

The extreme design basis wind is derived from the design basis tornado. Extreme wind is classified as a natural phenomenon Design Event IV as defined in ANSI/ANS-57.9.

8.2.2.1 Cause of Accident

Extreme winds due to passage of the design tornado, defined in Section 3.2.8, are postulated to occur as a severe natural phenomenon.

8.2.2.2 Accident Analysis

The site is located in tornado Region III as defined in Regulatory Guide 1.76 (Reference 15). The design basis tornado loading for this region is defined as a tornado with a maximum wind speed of 240 mph and a 1.5 psi pressure drop occurring at a rate of 0.6 psi/sec, including the effects of postulated Spectrum I tornado generated missiles that could be created by the passage of the tornado as identified in Section 3 of NUREG-0800 (Reference 16).

Storage Casks

The HI-STORM and TranStor storage systems are designed to withstand loads associated with the most severe meteorological conditions, including extreme winds, pressure differentials, and missiles generated by a tornado. Results of the evaluation of effects of a tornado on the HI-STORM and TranStor storage systems are described in their SARs (References 2 and 3, respectively). Both storage systems are designed to the design basis tornado criteria for tornado Region I (Maximum wind speed of 360 mph and 3.0 psi pressure drop occurring at a rate of 2.0 psi/sec), which substantially envelopes the Region III criteria for the PFSF.

The HI-STORM and TranStor SARs demonstrate that the 360 mph wind loading on the cask area produces insufficient forces to tip over the casks. Spectrum I missiles are assumed to impact a storage cask in a manner that produces maximum damage. The combination of tornado winds with the most massive Spectrum I missile, a 3,960 lb (1,800 kg) automobile traveling at 126 mph, was also evaluated in accordance with Section 3 of NUREG-0800. The wind tipover moment was applied to the cask at its maximum rotation position following the worst-case missile strike. Calculations presented in the HI-STORM and TranStor SARs determined that the restoring moment far exceeded the overturning moment and the storage casks would not tip over.

While the calculations demonstrate that design missiles could not cause the storage casks to tip over, they could inflict localized damage. The HI-STORM and TranStor SARs demonstrate that none of the Spectrum I design missiles are capable of penetrating the storage cask and striking the canister, and canister confinement would not be affected. However, design missiles could cause a localized reduction in shielding. SNC calculated worst case damage to a TranStor storage cask of 5.69 inch deep penetration from the 8 inch diameter design missile (TranStor SAR Section 11.2.3). The TranStor and HI-STORM SARs conclude that while tornado missiles could cause localized damage to the radial shielding of the storage casks resulting in increased dose rates on contact with the affected area, the damage will have negligible effect on the dose at the OCA boundary.

Based on the above, the HI-STORM and TranStor storage systems meet the general design criteria of 10 CFR 72.122(b), which states that SSCs classified as Important to Safety must be designed to withstand the effects of tornadoes without impairing their capability to perform safety functions. Since tornado winds and tornado generated missiles do not have the capability to damage the canister, a tornado strike on or about loaded storage casks will not result in a release of radioactivity.

Canister Transfer Building

The Canister Transfer Building shields and protects the SSC's housed within it and the canister transfer activities, which take place inside, from the effects of severe natural phenomena. The Canister Transfer Building is designed to withstand the effects of the Region III design basis tornado wind and pressure drop forces, as well as the effects of Spectrum I tornado missiles as defined in Regulatory Guide 1.76 and Section 3 of NUREG-0800 (see Section 3.2.8).

The building provides this protection by means of thick reinforced concrete walls and roof of sufficent strength to withstand the design basis wind, pressure drop, and missile forces. Additional missile protection is provided by the interior reinforced concrete walls and missile / shielding doors and/or labyrinths.

8.2.2.3 Accident Dose Calculations

Extreme winds in combination with tornado-driven missiles are not capable of overturning a storage cask nor of damaging a canister within a storage cask. The Canister Transfer Building is designed to withstand wind forces and missiles associated with the Region III design basis tornado, protecting canister transfer operations from the effects of tornadoes. Therefore, no radioactivity would be released in the event of a tornado. Dose rates at the OCA boundary would not be affected by damage to storage casks from tornado-driven missile strikes .

It is assumed that it would take two technicians 30 minutes to repair the worst case damage resulting from impact to a TranStor storage cask by a design missile (a penetration 5.69 inches deep and 8 inches in diameter), by filling the damaged area with grout. SNC shielding calculations (TranStor SAR Sections 5.4.8 and 11.2.3) predict surface dose rates of less than 300 mrem/hr in the center of the damaged area for all design fuel cases for a concrete cask with 5.69 inches of concrete removed. A conservative estimate of the dose rate one meter from the damaged area is 150 mrem/hr, and the total dose to repair the cask is estimated to be 150 person-mrem.

8.2.3 Flood

Flood is classified as a natural phenomenon Design Event IV as defined in ANSI/ANS-57.9.

8.2.3.1 Cause of Accident

The probable maximum flood is considered to occur as a severe natural phenomenon.

8.2.3.2 Accident Analysis

Both the HI-STORM and the TranStor storage cask systems are designed to withstand severe flooding, including full submergence. However, the PFSF site will remain dry in the event of a flood because of the site location and site design measures (Section 3.2.9). The upper surfaces of the storage pads and the floor of the Canister Transfer Building, and other PFSF buildings, are situated above the elevation of the Probable Maximum Flood from offsite sources. The site area is designed to assure adequate drainage for heavy rainfall, including the 100-year event. Therefore, a flood will not impact spent fuel storage or transfer operations.

8.2.3.3 Accident Dose Calculations

The Probable Maximum Flood will not have any affect on PFSF operations because of the location and design of the PFSF site. There will be no releases of radioactivity and no resultant doses.

8.2.4 Explosion

Explosion is classified as a human-induced Design Event IV as defined in ANSI/ANS-57.9.

8.2.4.1 Cause of Accident

Section 2.2 "Nearby Industrial, Transportation and Military Facilities", indicates that the only facility which could contribute to the potential for significant explosions located within 5 miles of the PFSF is the Tekoi Rocket Engine Test facility. There are no chemical processing plants, petroleum refineries, natural gas facilities, or munition depots that could contribute to the potential for significant explosions located within 5 miles of the PFSF. The Tekoi Test facility is located approximately 2.5 miles south-southeast of the PFSF. This facility is used periodically to test engines mounted on stationary bases. Hickman Knolls, with an elevation of approximately 4,800 ft, is situated directly between the PFSF (elevation 4,460 ft) and the Tekoi Test facility (elevation approximately 4,600 ft). Overpressures resulting from the Tekoi Test facility would be substantially deflected and dispersed by the intervening Hickman Knolls and would not produce significant overpressures at the PFSF 2.5 miles away.

The northern perimeter of the Dugway Proving Grounds is approximately 9 miles from the PFSF and the Tooele Army Depot (south area) is approximately 22 miles from the PFSF. There is no interstate highway, railroad (other than the rail which may be installed specifically for shipments of spent fuel shipping casks to and from the PFSF), or river traffic within the vicinity of the PFSF. The nearest interstate highway and commercial rail line are about 24 miles to the north of the facility. The Skull Valley Road, which runs north and south through the Skull Valley Indian Reservation to the east of the PFSF and provides entrance to the site access road, is 1.9 miles from the Canister Transfer Building and 2.0 miles from the nearest storage pad. The worst-case explosion potential at the PFSF is considered to be from an accident associated with the transportation of explosives along the Skull Valley Road (elevation approximately 4,580 ft, with no obstacles intervening between PFSF).

8.2.4.2 Accident Analysis

Regulatory Guide 1.91 (Reference 17) provides guidance for calculating safe distances from transportation routes, based on calculated overpressures at the nuclear site created by postulated explosions from transportation accidents. The Regulatory Guide indicates that overpressures which do not exceed 1 psi at the storage site would not cause significant damage and states that "under these conditions, a detailed review of the transport of explosives on these transportation routes would not be required." Using the methodology of Regulatory Guide 1.91, the nearest transportation routes are located much further from the PFSF than the distances required to exceed 1 psi overpressure. Based on this Regulatory Guide, the maximum probable hazardous solid cargo for a single highway truck is 50,000 lb, and detonation of this quantity of explosives could produce a 1 psi overpressure at a distance of approximately 1,660 ft (0.31 mile) from the detonation. Since the Skull Valley Road is 1.9 miles from the Canister Transfer Building and 2 miles from the nearest storage pad, explosions invoving vehicles travelling on this road would not produce significant overpressures at these locations.

The effects of explosions on the storage systems are discussed in the HI-STORM and TranStor SARs, and it is determined that the canisters are protected from the effects of explosions. Overpressures of substantially greater than 1 psi would be required to cause damage to the cask storage systems. The Canister Transfer Building is designed to withstand extreme winds, pressure drops of 1.5 psi, and missiles associated with the design tornado. The effects of credible explosions occurring on the Skull Valley Road, with resultant overpressures less than 1 psi at the PFSF, would not challenge the

Canister Transfer Building's structural integrity. Therefore, the canister storage and transfer systems meet the general design criteria of 10 CFR 72.122(c), as it applies to explosion, which states that structures, systems, and components Important to Safety must be designed and located so that they can continue to perform their safety functions effectively under credible fire and explosion exposure conditions.

8.2.4.3 Accident Dose Calculations

Since there is no potential for significant overpressures occurring at the PFSF as a result of nearby explosions, there would be no damage to the cask storage or transfer systems and no resultant dose.

8.2.5 Fire

Fire is classified as a human-induced Design Event IV as defined in ANSI/ANS-57.9.

8.2.5.1 Cause of Accident

The only combustible material at the PFSF storage pads during storage operations is insulation on the temperature monitoring instrumentation wiring, which is present in insignificant quantities at each storage cask. No combustible or explosive materials are allowed to be stored on or near the storage pads. The PFSF Restricted Area (RA) is cleared of vegetation and the entire RA surfaced with compacted gravel. Movement of a storage cask from the Canister Transfer Building to a storage pad involves the use of a diesel-powered cask transporter, whose fuel tank has a capacity of 50 gallons of diesel fuel. The worst-case fire at the storage pads involves a postulated spill and ignition of this diesel fuel in the vicinity of a storage cask. The accident scenario involving a storage cask in the following section assumes that the fuel tank of the transporter vehicle ruptures, resulting in 50 gallons of diesel fuel spilled, which is postulated to ignite and burn.

The combustibles of key concern in the Canister Transfer Building are the transient combustibles associated with the diesel fuel tanks of the cask transporter and the heavy haul vehicle tractor. For rail delivery/retrieval of shipping casks, the train locomotives are required by administrative procedure to stay out of the Canister Transfer Building. The design of the building and its surroundings will assure that any diesel fuel spilled outside the building will not flow into the building, which could create a fire hazard. The heavy haul vehicle tractors have saddle tanks with a total capacity of up to 300 gallons of diesel fuel. Spillage of diesel fuel does not create the potential for explosions in the Canister Transfer Building, due to this fuel's low volatility. It should be noted that diesel fuel is difficult to ignite, and it is highly unlikely that spillage of diesel fuel would result in a fire. The following assumes that spillage of diesel fuel is somehow ignited, and considers fires in the Canister Transfer Building associated with postulated rupture of the cask transporter's fuel tanks, with up to 50 gallons of fuel spilled in a transfer cell, and postulated rupture of a heavy haul tractor's fuel tanks, with up to 300 gallons of diesel fuel spilled in the cask load/unload bay.

8.2.5.2 Accident Analysis

Storage System

A fire is assumed to occur when the fuel tank of the cask transporter ruptures spilling diesel fuel in the vicinity of a storage cask that is at its location on a storage pad, or enroute from the Canister Transfer Building to its storage location, and the diesel fuel is postulated to ignite and burn. This scenario is analyzed in Section 11.2.4 of the HI-STORM SAR. From IAEA requirements (Reference 18), the "pool" of fuel is assumed to completely encircle a storage cask and extend 1 meter beyond the cask surface. Based on the minimum outer cask diameter of 132.5 inches (HI-STORM), this spill would result in a ring of fuel with a pool surface of about 21,260 sq in around the storage cask. A fuel consumption rate of 0.15 in/min was assumed (Reference 19) based on gasoline/tractor kerosene experimental burning rates. This translates into a fuel consumption rate of approximately 14 gal/min. Therefore, the 50 gallons of fuel would sustain a fire for about 3.6 minutes.

The storage system designs are highly resistant to the effects of fires. The thick concrete walls are not significantly affected by short-term exposure to fire induced temperatures, and the thermal diffusivity is such that any fire would be required to burn for many hours before much of the wall thickness would be affected. HI-STORM SAR Section 11.2.4 describes the results of a transient analysis of the effects of a diesel fuel fire encircling a storage cask assumed to burn for 15 minutes. The analysis concludes that because of the comparatively small temperature rise at the concrete wall inner surface and the thermal inertia of the canister, the effect of a fire accident on the canister temperature is negligible. The ability of the HI-STORM system to cool the spent fuel within design temperature limits during post-fire equilibrium is not compromised. Intense heat from the fire only partially penetrates the storage cask wall, and the majority of concrete experiences a relatively minor temperature increase. Concrete exposed to extreme temperatures would experience a minor reduction of its neutron shielding capability, which would not significantly increase the dose rate from the cask. Based on this analysis, the effects of a fire of approximately 3.6 minutes duration on the storage system, which is less than the 15 minute fire condition analyzed in the HI-STORM SAR, will have a negligible effect on canister and fuel temperatures and cause no reduction in nuclear safety.

TranStor SAR Section 2.3.6 states in regards to the effects of fire on a storage cask: "... the TranStor Storage System design is highly resistant to the effects of fire. The thick concrete walls are capable of protecting the basket containing irradiated fuel. Although the exposed layer of concrete may lose a portion of its strength, it would not disintegrate from an exposure to flame temperatures on the order of 1,500°F (as specified in 10 CFR 71). In addition, any fire would be required to burn for a long time (days) before much of the wall thickness would be affected."

Canister Transfer Building

A fire in the Canister Transfer Building would have a negligible effect on storage casks on the storage pads because of the concrete construction of the building walls and the distance between the Canister Transfer Building and the storage pads. The Canister Transfer Building is approximately 425 ft from the nearest storage pad.

The Canister Transfer Building contains minimal combustible loading, except when a heavy haul tractor or cask transporter is present in the building. Transient combustibles associated with these vehicles are up to 300 gallons of diesel fuel inside the saddle tanks of the heavy haul tractor, and up to 50 gallons of diesel fuel inside the fuel tank of the cask transporter. In the event of rail delivery/retrieval of shipping casks, the train engines are required by administrative procedure to stay out of the Canister Transfer Building. Although it is highly unlikely that a fuel tank could rupture and spilled diesel fuel ignite, the design of the Canister Transfer Building includes provisions to address these scenarios, as discussed below.

The first postulated fire scenario in the Canister Transfer Building is assumed to involve 300 gallons of diesel fuel from ruptured fuel tanks of a heavy haul tractor in the shipping cask load/unload bay. The heavy haul vehicles enter and exit the cask load/unload bay at the south end of the Canister Transfer Building and do not approach a transfer cell where canister transfer operations are conducted. Building design measures assure that any diesel fuel spilled in the cask load/unload bay will remain in the bay and cannot enter a transfer cell. The Canister Transfer Building design includes automatic fire detection and suppression systems that provide complete building coverage, including the cask load/unload bay and the transfer cells (Section 4.3.8.1). The fire suppression system consists of a sprinkler system. A worst-case fire involving 300 gallons of diesel fuel would be detected and automatically extinguished in less than 15 minutes, and would not threaten the integrity of a canister in a shipping cask. The shipping casks are required to be demonstrated capable of safely withstanding the effects of an exposure fire that burns at 1475°F for 30 minutes per 10 CFR 71.73(c)(3).

The second postulated fire scenario in the Canister Transfer Building is assumed to involve 50 gallons of diesel fuel from ruptured fuel tanks of the cask transporter in one of the three canister transfer cells. A fire involving up to 50 gallons of diesel fuel could burn for up to 5 minutes duration, consuming the entire fuel inventory (without credit for suppression by the automatic fire detection/suppression system). The cask transporter enters a transfer cell for the purposes of moving an empty storage cask into the cell, and moving a loaded storage cask out of the cell and out to the storage pad. During canister transfer operations, the cask transporter is prevented from entering a transfer cell by shield doors on either side of the transfer cell. PFSF procedures will require that the shield doors remain closed when a canister transfer operation is in progress. Building design measures assure that any diesel fuel spilled in the Canister Transfer Building main bay outside of a transfer cell will not run into a transfer cell. A cask transporter could enter a transfer cell when the canister is in the shipping cask and its lid bolted in place, or when the canister is in the storage cask and the storage cask lid has been bolted in place. As noted above, the shipping casks are required by regulation to be demonstrated capable of safely withstanding the effects of an exposure fire that burns at 1475°F for 30 minutes, with spent fuel remaining within temperature limits and no breach of the confinement barrier. Therefore, short duration fires in a transfer cell resulting from postulated rupture of the cask transporter's diesel fuel tanks and ignition of the pool of fuel would not result in breach of the shipping cask confinement and there would be no release of radioactivity. Fires involving shipping casks can result in reduction of neutron shielding, as discussed in Chapter 5 of both vendors' shipping cask SARs (References 5 and 20). Storage casks are relatively impervious to the effects of fires, as discussed above, and there would be no damage to the canister confinement or the spent fuel for fires in the vicinity of a loaded storage cask. The occurrence of a fire in a transfer cell while a canister is in a transfer cask is precluded, since the cask transporter can not enter a transfer cell during the canister transfer operation and the cask transporter represents the only significant combustible loading in a transfer cell.

Based on the above, the canister storage and transfer systems meet the general design criteria of 10 CFR 72.122(c), which states that structures, systems, and components Important to Safety must be designed and located so that they can continue to perform their safety functions effectively under credible fire exposure conditions.

8.2.5.3 Accident Dose Calculations

The temperature of the canister would not significantly change in the event of a credible fire near a storage cask or in the Canister Transfer Building. Therefore, canister integrity would be retained in the event of fires and no activity released.

8.2.6 Hypothetical Storage Cask Drop / TipOver

The hypothetical drop / tipover of a storage cask is classified as Design Event IV as defined by ANSI/ANS-57.9. As discussed below, storage cask tipover events, and vertical end drop events from heights greater than 10 inches, are not credible.

8.2.6.1 Cause of Accident

The stability of the loaded storage casks in the upright position on the PFSF concrete storage pad is demonstrated in Chapter 4 of this SAR. The effects of earthquakes, tornado wind, and missiles are described in the HI-STORM and TranStor SARs, where it is shown that the loaded storage casks will not tip over under the severe design basis natural phenomena specified in Chapter 3 of this SAR. Seismic analyses by both vendors confirm the casks will not tip over in the event of the site-specific DE (Section 8.2.1).

The storage casks are moved from the Canister Transfer Building to the storage pad using the cask transporter. The bottom of a storage cask is only raised approximately 4 inches above the ground during movement of a loaded storage cask. The cask transporter is designed to mechanically prevent a storage cask lift of more than 10 inches above the ground. As discussed in the following paragraphs, storage cask end drops of up to 10 inches would not result in canister breach, and the storage cask would retain its structural integrity and continue to provide shielding and natural convection cooling for the canister.

Storage cask tipover accidents, and storage cask vertical end drop accidents from heights greater than 10 inches, are hypothetical events, since there are no credible causes. A storage cask tipover, and storage cask vertical end drop from 10 inches, are analyzed in order to assess potential consequences of such accidents.

8.2.6.2 Accident Analysis

Analyses of the hypothetical storage cask drop and/or tipover are documented in the HI-STORM and TranStor SARs. SNC analyzes the tipover accident in TranStor SAR Section 11.2.10 and Holtec analyzes tipover and vertical end drop accidents separately in HI-STORM SAR, Chapter 3.

HI-STORM Storage Cask

Holtec establishes design basis vertical and horizontal acceleration values for the HI-STORM storage cask system of 45 g. With the cask weight, cask dimensions, and center of gravity being known quantities, and the cask conservatively treated as a rigid body, calculations (Appendix 3.A of Reference 2) determine the storage pad stiffness (target hardness) that produces an acceleration of 45 g in the hypothetical tipover accident. Using this approach, the limiting stiffness for the storage pad was calculated to be 30.65 E6 lb/inch. This value bounds the calculated PFSF storage pad stiffness value of 10.8 E6 lb/inch. Therefore, hypothetical tipover of a HI-STORM storage cask at the PFSF would result in an acceleration less than 45 g and lower stresses than those evaluated in the HI-STORM SAR.

Based on the limiting storage pad stiffness calculated in the HI-STORM SAR for the hypothetical tipover accident, Holtec calculated the maximum drop height for a vertical end drop of the HI-STORM storage cask that would result in a deceleration of 45 g. This height was determined to be 10 inches (Appendix 3.A of Reference 2). Since storage casks can not be lifted above 10 inches by the cask transporter and the PFSF storage pads have a stiffness value less than that considered for drop analyses in the HI-STORM SAR, end drop accidents at the PFSF involving HI-STORM storage casks will produce decelerations less than the 45 g analyzed in the HI-STORM SAR.

For the canister, the peak acceleration of 45 g established for the side and end drops is bounded by the 60 g acceleration calculated for drop accidents analyzed in the HI-STAR Transport SAR (Reference 20). Since the accelerations are bounding, the stresses (produced by 60 g vertical and horizontal accelerations) analyzed in the HI-STAR stress analyses and determined to be acceptable also bound stresses that would result from the HI-STORM tipover and end drop accidents. The canister would retain its integrity and the canister and canister internals would continue to perform their safety functions (i.e. confinement; $k_{eff} < 0.95$; transfer of decay heat from the spent fuel assemblies to the canister shell; and shielding, especially in the top axial direction).

For the storage cask, the HI-STORM SAR evaluates the buckling capacity of the cask based on the 10 inch vertical end drop and resulting 45 g acceleration. No credit was taken for the structural stiffness of the radial concrete shielding. The minimum factor of safety for material allowable stresses for all portions of the cask structure is 1.13. The only criteria for the tip over event evaluated in the HI-STORM SAR is that the cask lid must remain in-place due to the 45 g horizontal acceleration. Chapter 3 of the HI-STORM SAR demonstrates that the minimum factor of safety for the cask lid and lid bolts is 1.194. It is considered that the tipover accident could cause some localized damage to the radial concrete shield and outer steel shell where the storage cask impacts the surface.

As stated in Section 3.5 of the HI-STORM SAR: “Studies of the capability of spent fuel rods to resist impact loads [3.5.1] indicate that the most vulnerable fuel can withstand 63g’s in the most adverse orientation. Therefore, designing the HI-STORM 100 System to a maximum deceleration of 45g’s will ensure that fuel rod cladding integrity is maintained during all normal, off-normal, and accident conditions.” The referenced document is included as Reference 21 of this Chapter.

TranStor Storage Cask

The TranStor storage cask was evaluated for the tipover accident in the TranStor SAR, Section 11.2.10, assuming the cask is rigid and the storage pad absorbs the energy. An equivalent horizontal cask drop height of 63 inches was calculated and analyzed, producing a calculated 17 g acceleration, with a bounding acceleration of 19.8 g for any drop height utilized for stress analyses. The maximum dynamic moment and shear in the cask wall were determined and evaluated in combination with other loads to the criteria of ACI-349 (Reference 22) and ANSI/ANS 57.9 (Reference 1). The results are presented in TranStor SAR Table 3.4-8, which shows that the storage cask design is adequate to withstand the tipover loads produced by the 19.8 g acceleration. The storage cask would not sustain significant damage due to the hypothetical tipover event, and would continue to perform its safety functions. It was determined that this analysis bounds the effects of a postulated 63 inch horizontal drop of a TranStor storage cask onto the PFSF storage pad. Using PFSF site-specific concrete and soil parameters, and applying the same target hardness methodology used by SNC, an acceleration of 16.2 g is calculated to result from this side drop onto a PFSF storage pad.

SNC calculated the maximum crush depth of the TranStor storage cask concrete by conservatively postulating that all energy of impact is absorbed by crushing of cask concrete, with the pad assumed to be absolutely rigid (TranStor SAR Section 11.2.10.2). The maximum crush depth was calculated to be 2.07 inches. This number is conservative since, in reality, a significant portion of the energy would be absorbed by crushing of the pad, taken by flexural deformation of the pad and cask, and dissipated as heat.

The canister and its internals were analyzed using the ANSYS finite element code and hand calculations to evaluate canister components in accordance with the ASME BPVC, Section III, Subsections NC and NG as applicable (References 5). The analyses determined that the acceleration experienced by the canister in the tipover accident (represented by the 63 inch horizontal drop) is bounded by the canister design acceleration of 44 g for a horizontal drop, stresses in the various canister components are within code allowables, the canister would retain its integrity, and the canister and its internals would continue to perform their safety functions (i.e. confinement; $k_{eff} < 0.95$; transfer of decay heat from the spent fuel assemblies to the canister shell; and shielding, especially in the top axial direction).

The TranStor storage cask was not explicitly analyzed for a vertical end drop accident. However, Section 12.2.2.8 of the TranStor SAR states that "The NRC evaluation of the cask drop analysis for VSC-24 concurred that drops between 18 and 80 inches can be sustained with acceptable damage (i.e. without breaching the confinement boundary, preventing removal of fuel assemblies, causing a criticality accident, or causing a structural failure of the concrete cask so it can not maintain its shielding function). Based on the engineering judgment, drops from heights up to 18 inches are not considered to be a concern. Since the TranStor cask and basket are similar to the VSC-24 and designed for the same drop loads, no additional analysis is required." The TranStor canister is analyzed and shown to withstand a vertical deceleration of 50 g, analyzed to result from vertical end drop of a TranStor shipping cask onto its impact limiters (Reference 5). SNC has performed additional calculations which indicate that stresses in the various canister components remain within code allowables, and the canister and its internals would continue to perform their safety functions, for a vertical deceleration of 124 g (Reference 23).

8.2.6.3 Accident Dose Calculations

Based on the results of the analyses described above, the cask/canister storage systems would retain their confinement integrity and there would be no release of radioactivity and no resultant doses in the event of hypothetical drop/tipover of a fully loaded storage cask. For tipover of a HI-STORM storage cask, it is considered that localized damage to the radial concrete shield and outer steel shell where the cask impacts the pad could result in an increased surface dose rate due to the damage. However, this would not produce a noticeable increase in the dose rates at the RA fence or OCA boundary because the affected area would likely be small (HI-STORM SAR, Section 11.2.3). The maximum concrete crush depth of 2 inches calculated for the TranStor storage cask would approximately double the dose rates in the localized area, but would not significantly affect the overall dose rates from the storage cask (TranStor SAR Section 11.2.10).

In the hypothetical event of a storage cask tipover / drop accident that is postulated to result in damage to a storage cask, the PFSF staff would evaluate the extent of damage and if needed would remove a canister from the damaged storage cask and transfer the canister to a new storage cask in the Canister Transfer Building utilizing a transfer cask to provide canister shielding and a single-failure-proof crane.

8.2.7 Hypothetical Loss of Confinement Barrier

The hypothetical loss of confinement barrier is classified as Design Event IV as defined by ANSI/ANS-57.9. This is not a credible accident at the PFSF.

8.2.7.1 Cause of Accident

The HI-STORM and TranStor canisters are totally sealed, integrally welded pressure vessels, designed to Section III of the ASME BPVC. There are no gaskets, mechanical seals, or packing that could provide a potential leakage path for the radioactive fission products contained within the fuel cladding. The canisters are provided with multiple lids to confine the radioactive fuel. Following welding of the lids, the canisters are tested to verify their leaktight integrity. No components are required to penetrate the sealed canisters after helium backfilling is completed and the outer closure is welded in place. The postulated failure of the cladding of all fuel rods in a canister and release of gases normally contained in the fuel rod cladding under pressure would not challenge the integrity of the canisters (Section 8.2.10). Loss of the confinement boundary is considered to be a non-credible event, which will not occur over the life of the PFSF. Nevertheless, a non-mechanistic loss of the confinement boundary is hypothesized and analyzed below.

8.2.7.2 Accident Analysis

Although no credible accident scenario has been identified that would result in the rupture of the confinement barrier, a hypothetical case was evaluated based on the assumption that the cladding of all fuel rods in a canister ruptures with concurrent, non-mechanistic failure of canister lid welds. The spent fuel is stored in a manner that complies with the general design criteria 10 CFR 72.122(h), in that the spent fuel cladding is protected during storage against degradation that could lead to gross ruptures. The space internal to the confinement boundary is filled with an inert gas (helium) without the presence of air or moisture that might produce the potential for long term degradation of the spent fuel cladding. The spent fuel storage systems are designed to assure that fuel is maintained at temperatures below those at which fuel cladding degradation occurs, under normal, off-normal, and accident conditions. It is therefore highly unlikely that a spent fuel assembly with intact fuel cladding will undergo cladding failure during storage, and the assumption of complete cladding failure of all rods in a canister is extremely conservative.

The analysis assumes the following radionuclides are released at the specified release fractions, based on guidance in Section 7 of NUREG-1536 (Reference 24):

Nuclide	Fraction of Total Radionuclide Inventory Assumed to be Released
H-3	0.30
Kr-85	0.30
I-129	0.10
Cs-137	2.3 E-5
Cs-134	2.3 E-5
Sr-90	2.3 E-5
Ru-106	1.5 E-5
Co-60	0.15

The source of Co-60 is assumed to be crud deposited on the fuel rods. The total Co-60 inventory is based on a surface concentration of 140 $\mu Ci/cm^2$ over the total surface area of all PWR fuel rods in a canister, and 600 $\mu Ci/cm^2$ over the total surface area of all BWR fuel rods in a canister, in accordance with NUREG-1536.

The quantities of the remainder of radionuclides identified above are based on the fission product inventories of PWR and BWR fuel assemblies. The following radionuclide inventories were derived from ORIGEN-S calculations by Holtec for HI-STORM design fuel, as described in Section 7 of the HI-STORM SAR. The PWR values are based on the B&W 15X15 fuel assembly with a conservative burnup of 45,000 MWd/MTU and 5-year cooling time. Holtec determined that this bounds PWR fuel assembly types which will be loaded into the HI-STORM system. Similarly, bounding values for BWR fuel types are based on the GE 8X8R fuel assembly with a burnup of 45,000 MWd/MTU and 5-year cooling time. The following radionuclide inventories from the HI-STORM SAR bound those modeled in the TranStor SAR for this hypothetical accident condition.

Radionuclide	Inventory Associated with 24 PWR Design Fuel Assemblies* (μCi)	Inventory Associated with 68 BWR Design Fuel Assemblies* (μCi)
H-3	5.74 E9	5.84 E9
Co-60	5.66 E8	2.33 E9
Kr-85	9.70 E10	9.79 E10
Sr-90	1.02 E12	1.05 E12
Ru-106	2.93 E11	2.55 E11
I-129	5.04 E5	5.19 E5
Cs-134	5.23 E11	4.85 E11
Cs-137	1.51E12	1.56 E12

* 45,000 MWd/MTU burnup and 5-year cooling time

Based on Table XIX of Reference 25, it is assumed that 90 percent of particulate and volatile fission products (Co-60, Sr-90, I-129, Ru-106, Cs-134, and Cs-137) are held up within the breached canister following release from the fuel assemblies, and do not escape to the atmosphere. Ten percent of these radionuclides and 100 percent of the H-3 and Kr-85 are assumed to escape from the canister breach to the atmosphere.

The radionuclides postulated to be released from the hypothetical breached canister are dispersed in the atmosphere, reaching the downwind dose point and resulting in a dose to a hypothetical individual standing at the nearest point of the OCA boundary for the duration of the release.

8.2.7.3 Accident Dose Calculations

The nearest distance from a PFSF storage pad to the OCA fence (site area boundary) is 646 meters, and the nearest distance from the Canister Transfer Building to the OCA fence is 500 meters. A χ/Q of 1.94 E-3 sec/cubic meter was calculated in accordance with Regulatory Guide 1.145 (Reference 6), assuming a distance of 500 meters from the release source to the dose receptor, a wind speed of 1 meter/sec, and atmospheric stability class F, with no consideration for plume meander.

The dose conversion factors for Committed Effective Dose Equivalent (CEDE) and Committed Dose Equivalent (CDE - organ) due to inhalation are specified in Reference 7. An adult breathing rate of 3.3 E-4 cubic meters per second is assumed (Reference 7). Based on Table XX of Reference 25, 95 percent of Co-60 and Sr-90 particulates are greater than 10 microns aerodynamic diameter and are non-respirable. Therefore, a respirable factor of 0.05 was applied to these particulates to account for inhalation of those particulates having an aerodynamic diameter less than 10 microns. In addition to inhalation dose equivalents, immersion doses were also calculated that result from exposure to radiation emitted by the radionuclides in the plume. These immersion doses were calculated using the Passive/Evolutionary Regulatory Consequence Code ("PERC2" - Reference 26) based on radionuclide concentrations calculated at the OCA boundary. Gamma exposures resulting from immersion in the plume were added to the inhalation dose equivalents to arrive at total dose equivalents. Assuming an individual is located within the plume 500 meters (at the closest point of the OCA boundary to the Canister Transfer Building) from the breached canister for the duration of the release, it is calculated that the individual would receive a total CEDE of 547 mrem from hypothetical breach of a PWR canister containing 24 design fuel assemblies and a total CEDE of 752 mrem from hypothetical breach of a BWR canister containing 68 design fuel assemblies. The CDE to the lungs, the maximally exposed organ, was calculated to be 2,470 mrem from a breached PWR canister and 3,480 mrem from a breached BWR canister. Consistent with Reference 7, the doses from H-3 include a factor of 1.5 to account for absorption through the skin.

The above doses at the OCA boundary calculated to result from a hypothetical canister breach accident are below the 5 rem to the whole body or any organ criteria specified in 10 CFR 72.106 (b). Note that although the consequences have been evaluated, this is not considered to be a credible event for the PFSF.

8.2.7.4 Recovery Plan for a Hypothetical Canister Breach

As discussed in Section 8.2.7 above, the breach of a canister is not considered credible. However, for a hypothetical canister breach, a plan has been developed to recover from such an event.

The primary method of recovery from a breached canister would be to return the canister via shipping cask to the originating nuclear power plant or another facility having the capability to handle individual spent fuel assemblies. Transport of a breached canister would be in a licensed spent fuel shipping configuration in which the shipping cask provides the confinement boundary, with no reliance on the canister for fission product confinement. Since radioactive material that could potentially escape from the breached canister would be confined within the shipping cask, which is qualified to maintain its integrity under normal conditions of transport as well as accident conditions, transport of a breached canister would have no adverse environmental impacts.

The additional method of recovery from a breached canister would be to enclose the breached canister inside of another confinement vessel. This action can accomplished onsite without the need to handle individual spent fuel assemblies inside the canister.

Holtec has designed and is licensing the HI-STAR shipping cask for both transportation and storage of spent fuel. The cask is metal and is designed to provide the confinement boundary in the shipping configuration. For the hypothetical breach of a Holtec canister, the canister would be transferred to a HI-STAR metal storage cask to re-establish the storage confinement boundary. The HI-STAR cask would be vacuum-dried, backfilled with helium, sealed closed, and shipped or placed in storage at the PFSF for shipment offsite at a later date. The procedure is addressed in HI-STORM SAR, Section 8.4. This recovery method would use steel storage casks with mechanical closure and would require a site specific seismic analysis, equipment to vacuum dry and backfill the HI-STAR cask with helium, and a pressure monitoring system to ensure the integrity of the mechanical seal.

Sierra Nuclear Corporation (SNC) has designed a canister overpack, a second metal canister that a breached TranStor canister can be inserted into, which functions as the confinement barrier. Once the leaking canister is placed in the canister overpack, the lid of the canister overpack would be welded to the shell and the overpack system would be vacuum dried and helium backfilled. The backfill port would then be welded closed and the welds would be tested. The sealed canister overpack, now loaded with the breached canister, would be transferred into a TranStor storage cask and moved to the storage pad. When the canister is shipped offsite, it would be necessary to cut the canister overpack open to remove the breached canister and transfer it into a TranStor shipping cask where it could be sealed (shipping cask provides the confinement barrier) and shipped offsite. Appropriate radiological controls and airborne surveys would be taken during the transfer process to ensure protection of PFSF workers. This procedure is addressed in Reference 27, Section 5.1.1.5 for the TranStor storage system. This recovery method would require equipment to weld, vacuum dry and backfill the canister overpack with helium, perform required tests, and a pressure monitoring system to ensure the integrity of the single closure canister overpack.

As an alternative to placing a breached TranStor canister inside of the overpack, the breached canister could be sealed in a TranStor shipping cask and stored at the PFSF for shipment offsite at a later date, as with the HI-STAR cask recovery method. This would require an amendment to the TranStor SAR Certificate of Compliance. This recovery method would use a steel storage cask with mechanical closure and would require a site specific seismic analysis, equipment to vacuum dry and backfill the shipping cask with helium, and a pressure monitoring system to ensure the integrity of the mechanical seal.

Another method of recovery from a breached canister would be to utilize a portable dry transfer system at the PFSF site. A dry transfer system is not part of the PFSF design, however, several portable dry transfer system units are expected to be in operation by the Private Fuel Storage L.L.C. (PFSLLC) for use at power plants with restrictions on lifting heights or crane capacity. One of these dry transfer system units could be brought to the PFSF as needed to recover from a hypothetical canister breach at the PFSF. The dry transfer system would enable onsite transfer of individual fuel assemblies from a breached canister to a new canister.

The portable dry transfer system consists of a transfer vessel, canister shield adapter, and load/unload frame. The transfer vessel would be an upright steel cylindrical vessel that provides complete confinement for a limited number of spent fuel assemblies. The transfer vessel contains a single-failure-proof hoist and grapple within its confinement boundary that are used to lift or lower the fuel assemblies. The transfer vessel is mounted on top of the canister shield adapter, which is mounted on top of the cask. A shield door is located on the bottom end of the transfer vessel to enable fuel assemblies to be moved between the transfer cask and canister. The canister shield adapter is a cover that encloses the top of the canister in lieu of the canister lid, which is removed. The adapter also provides a shield barrier to limit the radiation doses out the top of the canister. A load/unload platform frame provides support for the canister shield adapter and transfer vessel and allows personnel access to operate the equipment. This recovery method would require the dry transfer system, equipment to weld the new canister, vacuum dry and backfill the canister with helium, perform tests, monitor for airborne contamination, and dispose of the opened canister.

Although a canister breach is a non-credible, hypothetical event at the PFSF, this recovery plan provides several reasonable alternatives of recovery from such a hypothetical event with negligible impacts on public health and safety.

8.2.8 100% Blockage of Air Inlet Ducts

Complete blockage of the air inlet ducts is classified as Design Event IV as defined by ANSI/ANS-57.9.

8.2.8.1 Cause of Accident

This event involves postulated complete blockage of all four storage cask air inlet ducts. Heat is normally removed from the canister shell by natural convection, and the heated air flows up the annulus by natural convection to four top outlet ducts, where the hot air exits the storage cask.

Since the HI-STORM storage casks have four air inlet ducts 90° apart and the TranStor storage casks have four air inlet ducts, with two located on opposing sides of the cask, it is highly unlikely that all air inlet ducts could become blocked by blowing debris, snow, rodents, or other material. A severe windstorm could possibly blow debris against the bottom of the storage casks and possibly clog one or two of the inlet screens exposed to the wind, but the inlets on the leeward side of the cask would be expected to remain relatively free of dirt and debris. If a large sheet of plastic or a tarpaulin were to blow against a storage cask (which is unlikely since the RA is surrounded by two 8-ft high chain link fences that would be expected to catch such items), it could wrap partially around the storage cask and block, or partially block, the air inlet ducts on the windward side, but ducts on the opposite side would be expected to remain open.

One means of cutting off normal convection airflow would be a flood in which the height of the water exceeded the tops of the air inlet ducts. However, since the PFSF location and design assures that the upper surfaces of the storage pads are at an elevation above the elevation of the probable maximum flood in this area, blockage of the inlet ducts by flooding is not credible.

8.2.8.2 Accident Analysis

Analyses of this event are included in the HI-STORM and TranStor SARs. The analyses assume complete blockage of all four air inlet ducts, preventing air flow through the normal circulation paths. The Holtec analysis (HI-STORM SAR Section 11.2.13) considers the large thermal margin between normal peak fuel cladding temperatures and the short-term limit (1,058°F) in a HI-STORM storage cask and demonstrates that the HI-STORM canister temperatures can heat up over 300°F before the cladding short-term peak temperature limit is reached. With all inlet and outlet air ducts blocked, Holtec determined a 300°F temperature rise of the HI-STORM storage system would take 92 hours assuming adiabatic conditions and the highest heat load canister (MPC-24). Duct blockage would be detected by the cask temperature monitoring system and removed before temperatures could approach the maximum temperatures considered in this analysis.

The SNC analysis (Section 11.2.7 of the TranStor SAR) calculates the TranStor maximum steady-state temperatures for the 100 percent inlet duct blockage case and shows that temperatures do not exceed short-term allowable limits, with consideration for some cooling by natural convection air flow into and out of the air outlet ducts at the top of the cask. It is estimated that the duct blockage condition would have to exist for 4 to 5 days to reach steady state temperatures.

8.2.8.3 Accident Dose Calculations

Adiabatic analyses of 100 percent duct blockage conditions determined that fuel cladding temperatures do not reach the short-term limit for several days. In the extremely unlikely occurrence of 100 percent blockage of the storage cask inlets, periodic surveillance of the storage cask temperature monitoring system would identify any duct blockage and the blockage would be expeditiously removed before short-term temperature limits are exceeded. The canister would maintain its confinement integrity, and there would be no releases of radioactivity. Therefore, no offsite doses would result from this accident.

The radiation dose to PFSF workers who remove debris blocking the inlet ducts are estimated to be double those conservatively estimated for the analysis of one-half the inlet ducts blocked in Section 8.1.3.4, or approximately 70 person-mrem.

8.2.9 Lightning

Lightning is classified as a natural phenomenon Design Event IV as defined in ANSI/ANS-57.9.

8.2.9.1 Cause of Accident

This event would be caused by meteorological conditions at the site. Lightning striking one of the storage casks is not a likely event, because there are grounded metal light fixtures/poles in the vicinity of the storage pads that are substantially higher than the storage casks (approximately 120 ft high).

8.2.9.2 Accident Analysis

If a storage cask were hit by lightning, the likely path to ground would be from the steel cask lid to the steel base plate via the steel cask liner(s) and the steel air inlet ducts. The canister is surrounded by these steel structures and would not provide a likely ground path. Therefore, a lightning strike would not affect canister integrity. The absorbed heat would be insignificant due to the very short duration of the event. If the lightning entered or exited the cask via the concrete shell, some local spalling of concrete might occur. Storage cask operation would not be adversely affected.

8.2.9.3 Accident Dose Calculations

The canister would retain its confinement integrity, and there would be no releases of radioactivity. Therefore, no offsite doses would result from this accident. The effects of localized shielding loss due to spalling of storage cask concrete would be bounded by dose rates discussed in Section 8.2.2.3 for worst case tornado missile penetration.

8.2.10 Hypothetical Accident Pressurization

Accident pressurization is classified as a hypothetical Design Event IV as defined by ANSI/ANS-57.9. This is not a credible accident.

8.2.10.1 Cause of Accident

The spent fuel is stored in a manner that complies with the general design criteria 10 CFR 72.122(h), in that the spent fuel cladding is protected during storage against degradation that could lead to gross ruptures. The space internal to the confinement boundary is filled with an inert gas (helium) without the presence of air or moisture that might produce the potential for long term degradation of the spent fuel cladding. The spent fuel storage systems are designed to assure that fuel is maintained at temperatures below those at which fuel cladding degradation occurs, under normal, off-normal, and accident conditions. It is therefore highly unlikely that a spent fuel assembly with intact fuel cladding will undergo cladding failure during storage, and the assumption of complete cladding failure of all rods in a canister is extremely conservative. Failure of the cladding of all fuel rods contained in a canister is considered to be a non-credible event. Nevertheless, a hypothetical breach of all fuel rods in the canister and subsequent release of their fission and fill gases to the canister interior is analyzed. This would pressurize the canister shell and lids.

8.2.10.2 Accident Analysis

The analysis of this accident entails calculation of the free volume in the canister as well as the quantities of fill and fission gases in the fuel assemblies. The canister pressure is then determined based on the addition of 100 percent of the fuel rod fill gas and a conservative fraction of the fission gases to the helium already present in the canister. The fuel rods are initially assumed to be at a bounding fill pressure.

The evaluation of the canister pressurization accident is provided in HI-STORM Section 11.2.9 and TranStor SAR Section 11.2.6. Table 4.3.4 of the HI-STORM SAR and Section 11.2.6 of the TranStor SAR identify the fractions of fission product gases assumed to be released from fuel rods into the canister. The vendors' structural analyses evaluate the canister confinement boundary for this accident condition. The structural analyses show that stresses resulting from accident pressure, or the canister design basis internal pressure that exceeds accident pressure, are within applicable ASME BPVC Section III allowables.

8.2.10.3 Accident Dose Calculations

Since the analyses determined that the canisters would retain their integrity, there are no radiological consequences for this accident.

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8.3 SITE CHARACTERISTICS AFFECTING SAFETY ANALYSIS

Site characteristics have been considered in the formation of the bases for these safety analyses. The PFSF site layout was considered in determining conservative χ/Q atmospheric dispersion factors to estimate doses from accidents involving postulated and hypothetical releases of radioactivity to a hypothetical individual located at the closest point of the OCA boundary to the source of radioactivity for the duration of the releases. The site location, relative to the nearest major highway, was considered in the assessment of effects of postulated explosions resulting from transportation accidents.

Thermal analyses of the effects of abnormally high ambient temperatures on the storage system considered climactic conditions of the area, and temperatures were selected to bound day/night average maximum temperatures that could occur over a period of several days (Reference 4).

Regional and site geology and seismology were used to define the DE. Regional meteorology was considered in the determination of the design basis tornado parameters (Reference 15). The evaluation of the potential for fires is based on characteristics of the area surrounding the concrete storage pads, as well as the systems that will be used to transfer canisters and storage casks.

Information associated with aircraft flights in the vicinity of the PFSF, presented in Section 2.2 of this SAR, is based on data obtained from the Dugway Proving Grounds and its associated Michael Army Air Field and on flight path data issued by the Federal Aviation Administration (FAA) and the National Oceanic and Atmospheric Administration (NOAA). As discussed in Section 2.2, the calculated probability of an aircraft impacting the PFSF is below the applicable guidance and therefore is not considered to be a credible event.

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TABLE 8.1-1

STORAGE SYSTEM OFF-NORMAL MAXIMUM AMBIENT TEMPERATURE EVALUATION

	HI-STORM Storage System Temperatures (°F)	TranStor Storage System Temperatures (°F)
Ambient Air	100	100
Storage Cask Air Outlet	226	200
Storage Cask Outer Surface	166	141
Storage Cask Inner Concrete	287	222
Canister Outer Surface	452	299
Fuel Clad	652	688

Note: The above results are for the bounding canister temperature case (PWR or BWR) for the respective vendor.

TABLE 8.1-2

PARTIAL BLOCKAGE OF STORAGE CASK AIR INLET DUCTS TEMPERATURE EVALUATION

	HI-STORM Storage System Temperatures (°F)	TranStor Storage System Temperatures (°F)
Ambient Air	80	75
Storage Cask Air Outlet	227	189
Storage Cask Outer Surface	150	87
Storage Cask Inner Concrete	288	201
Canister Outer Surface	451	283
Fuel Clad	656	673

Note: The above results are for the bounding canister temperature case (PWR or BWR) for the respective vendor.

CHAPTER 9

CONDUCT OF OPERATIONS

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CHAPTER 9

CONDUCT OF OPERATIONS

9.1 ORGANIZATIONAL STRUCTURE

The Private Fuel Storage L.L.C. (PFSLLC) is a limited liability company organized under the laws of the State of Delaware. The PFSLLC is owned by eight member utilities contributing equity in equal amounts throughout the pre-license application phase and in varying amounts after submittal of the License Application to the NRC. Capital contributions are invoiced and paid quarterly in advance of expenditure, and no pre-construction debt is incurred. Each member utility (members must be U.S. utilities) selects one member of the Board of Managers. The Chairman is selected by the Board.

Spent fuel storage customers (both PFSLLC members and non-members) will enter into Service Agreements with the PFSLLC that will provide for the funding of facility construction and operation. An indexed one time fee will be assessed in three payments. The first payment will be due at the commencement of construction, the second approximately one year before shipment, and the final payment at time of shipment. All payments will be received by the PFSLLC prior to receipt of spent fuel at the PFSF. Customers will also pay transportation fees, an indexed annual fee for storage, a per-canister decommissioning fee (paid in advance), and a decommissioning assessment if the customer is responsible for any contamination found at the PFSF. No customer Service Agreements may be assigned by the customer without express PFSLLC permission.

9.1.1 PFSLLC Organization

The PFSLLC organization at the time of filing of the License Application consists of a Board of Managers, with specific committees made up of members of the Board or representatives of their companies, as well as contracted support for engineering and design, public affairs, legal counsel and other necessary expertise. The Board committees were established in several functional work areas to manage and accomplish needed tasks to prepare this license application and associated reports. The PFSLLC will name additional officers and hire additional employees as the project moves from the pre-licensing phase to the licensing and construction phase, and subsequently, into the operational phase. Figures 9.1-1, 9.1-2 and 9.1-3 identify the PFSLLC organization that will be in place through each of these three phases. The Board of Managers directs and oversees activities in all three phases, and is described in the following paragraphs:

9.1.1.1 PFSLLC Functions, Responsibilities and Authorities

The PFSLLC organization is structured to be operated by a Board of Managers during the pre-licensing, licensing and construction, and operational phases of the PFSF. Representatives to the Board of Managers are chosen by the member utilities. The Board is under the direction of a chairman, who is selected by the Board members. Voting rights of each representative are in proportion to the associated member utility's respective ownership interest in the Company.

The Board of Managers is responsible for:

- Providing the executive functions of the PFSLLC and exercising those functions through a Chairman of the Board. In the operational phase, the Chairman of the

Board, along with supporting staff in the Office of the Chairman, will perform the role of supervisor of the General Manager/Chief Operating Officer (Figure 9.1-3).

- Performing the long-range planning necessary to ensure stable resources for the operation of the facility. The Board will ensure that appropriate financial stability is maintained on an operating basis.

- Preparation and submittal of the PFSF License Application, including the Safety Analysis Report, Environmental Report and Emergency Plan; securing inside and/or outside expertise to assist in preparation of the License Application; and development of responses to address requests for additional information from the NRC (pre-licensing phase).

- Ensuring that the Quality Assurance Program is properly established, documented, approved, and effectively implemented by trained personnel with adequate resources, and that the Quality Assurance Committee/staff performs its designated oversight function and reports to the Board on matters affecting quality. The Board will assess the adequacy of the Quality Assurance Program implementation on a regular basis.

- Ensuring compliance with the conditions of the PFSF license, and ensuring that the Safety Review Committee (operational phase) reports to the Board and advises the Board on matters Important-to-Safety.

- Ensuring that decommissioning is properly funded through the escrow account, the letter of credit, and external sinking fund, and that decommissioning funding remains current by means of annual decommissioning funding reviews.

- Ensuring compliance with the terms of the Lease Agreement with the PFSF site host, the Skull Valley Band of Goshute Indians.

- Ensuring that License standards for engineering and design, construction, quality assurance, testing, and operation are met through oversight by the Chairman, utility staff, member utilities, expert consultants reporting directly to the Board, and the Quality Assurance Committee/staff. The Board shall maintain a direct relationship with the Quality Assurance Committee prior to PFSF construction and with the Quality Assurance staff throughout PFSF operation. A Project Manager is responsible to the Board for ensuring that all engineering and design tasks, as well as licensing support tasks, are completed on schedule.

9.1.1.2 PFSLLC In-House Organization

PFSLLC organization charts are provided in Figures 9.1-1, 9.1-2, and 9.1-3 for the organizations that will be in place during the pre-licensing, licensing and construction, and operational phases, respectively. The Board of Managers provides management direction and oversight throughout the various phases.

9.1.1.2.1 Pre-licensing Organization

Prior to licensing, the oversight of design and other project work activities rests with the committees of the Board of Managers and utility-provided PFSLLC staff, described in Section 9.1.1.4. The committees reporting to the PFSLLC Board of Managers consist of several functional groups, as outlined in Figure 9.1-1, each with a committee chair. The Committee Chairs work directly with the Chairman of the Board and the PFSLLC Project Manager to manage and complete activities in support of the license application. Typical committees are Licensing and Regulatory, Quality Assurance, and

Technology. Additional committees may be established as needed. Committees may be discontinued after their work is completed. The Quality Assurance Committee is responsible for the implementation of the Quality Assurance Program, including the preparation of procedures and maintenance of appropriate records. The Quality Assurance Committee conducts audits to ensure that requirements of the Quality Assurance Program are being met.

9.1.1.2.2 Licensing and Construction Organization

During construction, the PFSLLC will have a team of three persons available for oversight of PFSF design, procurement, and construction. This staff will be led by the PFSLLC Project Manager and will include a construction engineer and a procurement specialist. They will ensure oversight of the Architect/Engineer (A/E), contractors, and vendors and will be assisted as needed (at the discretion of the PFSLLC Project Manager) by utility staff from the member utilities in a full range of specialties appropriate to the design, construction, start-up, and operation of an independent spent fuel storage installation. These three persons will be available for initial training of the site staff prior to facility operation. Off-site support from the A/E, storage cask vendors, and others will be provided to the PFSF Project Manager, as needed.

During construction of the PFSF, the A/E will oversee the installation in accordance with quality assurance standards. As shown on Figure 9.1-2, the A/E reports to the PFSLLC Project Manager, who in turn reports to the Board of Managers. During construction of the PFSF, site administrative and engineering staff will be responsible for the administrative requirements of the site, including the maintenance of records in accordance with conditions of the License. Administrative staff will be responsible for the necessary personnel functions, ensuring that adequate business records and services are contracted for and maintained, and appropriate applicable hiring standards are followed in the selection of staff members. The oversight of construction shall be monitored by the Board of Managers and by the initial staff members, several of whom will be hired in the preconstruction design phase. The A/E will perform the oversight role on a daily basis in addition to the construction General Contractor.

During construction, Quality Assurance will be responsible for ensuring that structures, systems and components (SSCs) Important-to-Safety are designed, procured, fabricated, inspected, and tested in accordance with the Quality Assurance Program and in compliance with applicable codes, standards and requirements, including the maintenance of appropriate records. Quality Assurance will ensure that appropriate steps are added to site operation and maintenance procedures to ensure that all activities are performed and monitored in accordance with the site License.

9.1.1.2.3 Operational Organization

Following construction and prior to operation, the site organization will be managed by a General Manager who reports to the Chairman of the Board. The organization will consist of several functional groups, each with a leader. The leaders will function collectively with the General Manager to constitute the site Operations Review Committee (ORC) which will perform on-site safety assessment and review. All persons employed at the site will be hired and trained in compliance with the site License and Safety Analysis Report and other applicable guidelines.

During the operational phase, the General Manager will function as the Chief Operating Officer, responsible for day-to-day management of all PFSF operations, including canister receipt, handling, storage, and maintenance and surveillance activities. The General Manager/Chief Operating Officer reports directly to the Board of Managers, as shown on Figure 9.1-3. The Safety Review Committee (Section 9.1.2.1.1) advises the

Board of Managers on issues Important-to-Safety and is responsible for reviewing and approving modifications, plans, procedures and other activities that have elements that are Important-to-Safety. The ORC, consisting of the leaders of the various departments, will support the General Manager/Chief Operating Officer (who will serve as Chairman of the ORC) in the review/assessment of site operations. The ORC will consist of the lead persons from each department as indicated on Figure 9.1-3. A quorum consisting of a majority of ORC members will be required for the ORC to perform its function of operational assessment and safety oversight. Offsite nuclear engineering support during PFSF operations is discussed in Section 9.1.2.1.8.

Storage cask vendors will be inspected by PFSLLC representatives to ensure compliance with the 10 CFR 71 and/or 10 CFR 72 Certificates of Compliance and approved Quality Assurance Programs. Spent fuel shipment preparation at individual nuclear power plant sites will be overseen by the PFSLLC nuclear engineering staff.

9.1.1.3 Interrelationships with Contractors and Suppliers

As shown in Figures 9.1-1 and 9.1-2, the A/E reports to the PFSLLC Project Manager, who in turn reports to the Board of Managers, for the pre-licensing, and licensing and construction phases. The PFSLLC Project Manager is responsible to the Board for managing technical work activities and ensuring that engineering and design tasks, as well as licensing support tasks, are completed on schedule.

The PFSLLC and its A/E work under a contractual relationship which requires that all appropriate work by the A/E be performed in accordance with the A/E's approved Quality Assurance Program. Any subcontractor's support analysis or design work is required to have an approved Quality Assurance Program or to be directly supervised by the A/E.

The A/E Project Manager reports to the Project Manager of the PFSLLC, who is responsible for contractual compliance. The PFSLLC Quality Assurance and Technical Committees review and audit the work of the A/E. These committees consist of staff members from PFSLLC member utilities with specific expertise in the area of oversight. They are appointed by the Chairman of the Board. In addition, routine telephone conferences between the PFSLLC and the A/E are utilized to monitor progress and communicate issues.

The entire process of canister and storage cask selection in the pre-license phase of the project will be overseen by staff provided from PFSLLC member utilities. This mechanism will bring close industry scrutiny from a variety of potential users and reinforces the normal quality assurance oversight on this important function. Storage cask vendors are required by the contract with the PFSLLC to perform work under the approved Quality Assurance Program with oversight by the PFSLLC's Quality Assurance and Technical Committees.

Other suppliers contract with the A/E or the PFSLLC directly. All PFSLLC contracts and purchase orders relating to the License are reviewed for quality assurance applicability and standards by a member of the Quality Assurance Committee. Contracts and purchase orders are signed by the Chairman of the Board.

9.1.1.4 Technical Staff

The PFSLLC technical staff is provided by the member utilities. These staff members support the review of activities performed by the A/E and storage cask vendors. They also provide review for "Requests for Proposal" specifications to ensure transportation, dry transfer equipment, and on-site transfer equipment properly interface with the facilities of the individual nuclear power plant licensees.

Over 50 engineers have been involved in one or more phases of the project. They include nuclear engineers up to the Ph.D. level, several with 20 or more years of industry experience; operations personnel with direct fuel handling experience; and health physics staff with plant experience to the level of Radiation Protection Manager.

Security and mechanical design as well as instrumentation design oversight has been provided by staff at several member utilities. Safety Analysis Report preparation and review were a joint effort between these staff members and the A/E. Resumes of PFSLLC support personnel are on file with the PFSLLC.

9.1.2 Operating Organization, Management, and Administrative Control System

The following section describes the PFSLLC organization during the operational phase of the project.

9.1.2.1 On-Site Organization

Figure 9.1-3 details staff composition and job functions. During the operational phase, the Board of Managers has overall responsibility for safe operation of the PFSF, and the authority to ensure continued safe operation. The General Manager shall also function as the Chief Operating Officer during the operational phase, responsible to the Board of Managers for managing the PFSF in a manner that ensures safe and efficient operations and maintenance activities. The functions represented by the PFSLLC organization have the authority to control various aspects of the PFSF including engineering and design, quality assurance, fuel accountability, maintenance, radiation protection, training, operations, and decommissioning. This organization will ensure the continued safe operation of the PFSF during all normal, off-normal, and accident conditions.

Staff members in the various departments provide backup for the lead person in their respective department. Each specialty has more than one qualified individual and they are responsible for performing the lead responsibilities during the absence of the principal. The General Manager/Chief Operating Officer shall designate a lead person as a backup during his/her absence and rotate this responsibility among various leads to develop senior capability for site direction. Training will be available to General Plant Workers for the various technical specialties should an opening occur.

9.1.2.1.1 Safety Review Committee

As specified in the PFSF Technical Specifications, the PFSF Safety Review Committee is responsible for reviewing and advising the Board of Managers on all matters relating to Structures, Systems, and Components (SSCs) Important to Safety. The committee's responsibilities include, but are not limited to, the review of:

- Safety evaluations for procedures and changes thereto,
- Changes to SSCs classified as Important to Safety,
- Tests or experiments involving SSCs classified as Important to Safety,
- Review of Quality Assurance Audits related to safety
- Proposed changes to the technical specifications or the license, and
- Violations of codes, regulations, orders, license requirements, or internal procedures/instructions which pertain to SSCs classified as Important to Safety.

The committee consists, as a minimum, of members from the following functional areas:

Chairperson: PFSF General Manager/Chief Operating Officer
Quality Assurance
Radiation Protection
Nuclear Engineering
Maintenance/Operations

9.1.2.1.2 Mechanical Maintenance / Operations

The mechanical maintenance/operations staff will provide operations coverage for those periods of time in which fuel is being handled and routine site maintenance and surveillance when fuel is not being handled. The Mechanical Maintenance/Operations staff will also provide persons to operate railroad locomotives from the railroad mainline, or heavy haul vehicles from the intermodal transfer point, as needed.

Mechanics will be responsible for the mechanical maintenance of the facility. This will include performance of those maintenance functions required to maintain buildings, fencing, and operate mechanical equipment. When specialists within the area of Mechanical Maintenance are contracted for, it will be the responsibility of the Mechanical Maintenance personnel to oversee their work and ensure that the criteria of quality assurance is met. Mechanical Maintenance shall be responsible for ensuring that the appropriate records as outlined in the Quality Assurance Program or required by procedure are prepared and maintained in their area of responsibility to the standards delineated in the Safety Analysis Report.

9.1.2.1.3 Electrical and Instrument Maintenance

The Electrical and Instrument Maintenance staff will provide operations coverage for those periods of time in which fuel is being handled and routine site maintenance and surveillance when fuel is not being handled. The Electrical and Instrument Maintenance staff will also provide persons to operate railroad locomotives from the railroad mainline, or heavy haul vehicles from the intermodal transfer point, as needed.

Electrical and Instrument Maintenance personnel will be responsible for the electrical and instrument maintenance of the facility, including the performance of those maintenance functions required to maintain site electrical equipment as well as site instrumentation. When specialists within the area of Electrical and Instrument Maintenance are contracted for, it will be the responsibility of the Electrical and Instrument Maintenance personnel to oversee their work and ensure that quality assurance criteria are met. The Electrical and Instrument Maintenance staff shall be responsible for ensuring that appropriate records are maintained in accordance with the standards delineated in the Safety Analysis Report.

9.1.2.1.4 Radiation Protection

Personnel in the Radiation Protection group will be responsible for the functions of radiation safety and industrial safety. Radiation Protection staff will ensure compliance with the conditions of the License and the applicable personnel radiation safeguards such as 10 CFR 20. The Radiation Protection Manager shall be responsible for ensuring that appropriate records are maintained in accordance with the standards delineated in the Safety Analysis Report.

9.1.2.1.5 Security

The site security force will be trained in accordance with 10 CFR 73. The site security force will be responsible to maintain the security of special nuclear materials that are within the physical confines of this site. They will also be responsible for initial responses to both security intrusions and fires as outlined in the Physical Protection Plan. They shall be trained in advanced first aid and in fire fighting so that they are able to provide support in the area of the Skull Valley Indian Reservation to prevent range fires from entering the facility.

The security staff will function to initiate and support the response to any off-normal site event as outlined in the emergency plan and procedures. The security staff is also responsible for monitoring the storage cask temperature monitoring system and reports alarm conditions to the designated personnel.

9.1.2.1.6 Quality Assurance

The Quality Assurance staff will be responsible for the implementation of the Quality Assurance Program including the maintenance of appropriate records. The Quality Assurance staff will ensure that the appropriate steps are added to site procedures for operation and maintenance to ensure and monitor that all activities are performed in accordance with the site License.

9.1.2.1.7 Site Administrative and Engineering Staff

Site administrative and engineering staff will be responsible for the administrative requirements of the site including the maintenance of records in accordance with the conditions of the License. Administrative staff will be responsible for the necessary personnel functions ensuring that business records and services are contracted for and maintained and that appropriate hiring standards are followed in the selection of staff members. The site nuclear engineer will be responsible for the oversight of facility modifications as outlined in the Quality Assurance Program and procedures.

9.1.2.1.8 Off-Site Nuclear Engineering Support

The off-site nuclear engineering support for PFSLLC will provide monitoring at each nuclear power plant that ships spent fuel to the site. They will be familiar with applicable procedures of the individual plants and be knowledgeable of the requirements of the PFSF site. The contracting agreements with nuclear power plant licensees permitting shipment of their fuel to the PFSF will contain the necessary authority delegations to PFSF off-site engineers to permit them to have overall control of the PFSLLC property on the licensee's site and to ensure that all fuel is prepared and loaded into sealed canisters in accordance with the conditions of the PFSF License, the Service Agreements, and the shipping/storage cask Certificates of Compliance.

9.1.2.2 Personnel Functions, Responsibilities, and Authorities

The site staffing for the operational phase of the facility is described in the following paragraphs.

9.1.2.2.1 General Manager/Chief Operating Officer

The General Manager/Chief Operating Officer, supported by the facility's administrative and technical staffs, is responsible to the Board of Managers for managing the PFSF to ensure the safe and efficient operation and maintenance of the facility. The General Manager/Chief Operating Officer exercises direct control over all facility activities. This position has the authority and responsibility for providing staff resources (or contract support, as needed) and management direction to ensure the safe and efficient operation and maintenance of the facility. This position is responsible for document control and storage, training, security and the licensing interface with the NRC. This position is also responsible for the liaison between the PFSLLC and local governments in accordance with the Emergency Plan. The General Manager/Chief Operating Officer reviews proposed facility modifications, procedural changes, and tests, and has the authority to approve them for implementation, unless it is determined that the proposed modifications, changes or tests may involve an unreviewed safety question. This position approves procedures for facility operations, maintenance, equipment inspections, administration and security. The General Manager/Chief Operating Officer ensures that all subordinate or delegated responsibilities, assignments, authorities and relationships are understood and implemented by his/her lead people, engineers, and other staff members. The General Manager/Chief Operating Officer is familiar with all pertinent rules, regulations, codes, and procedures and ensures compliance as applicable.

The General Manager/Chief Operating Officer coordinates the activities of the facility with the Board of Managers and outside support services, and keeps the Board advised of facility performance. All unusual occurrences, incidents, or abnormalities in facility operation are reported to this position. The General Manager/Chief Operating Officer has the authority to shut down the facility operation and initiate emergency procedures or curtail operations in any emergency situation which should arise.

The General Manager/Chief Operating Officer is responsible for the scheduling and procurement of all equipment and materials (including special nuclear and source materials) necessary for the operation of the facility; the development of plans and procedures for facility administration, operation, and maintenance; the selection and hiring of facility personnel; and the development of appropriate training programs to ensure facility personnel are qualified.

9.1.2.2.2 Radiation Protection Manager

The Radiation Protection Manager is responsible to the General Manager/Chief Operating Officer for radiation safety at the PFSF, including the planning and direction of the facility radiation protection and ALARA programs and procedures, the operation of the health physics laboratory, and the technical and functional supervision of the Radiation Protection Technicians. The Radiation Protection Manager is responsible for all routine and special radiation monitoring for the protection of personnel and ensures that packaging, storage, and shipment of solid radioactive waste complies with applicable regulations. The Radiation Protection Manager advises and informs the General Manager/Chief Operating Officer on all matters pertaining to radiological safety, including the status of radiological health aspects of facility operation and maintenance and identification of potential radiological concerns. The Radiation Protection Manager is responsible for maintaining and monitoring all radiation protection related records for any trends which may affect facility operation.
The Radiation Protection Manager has the authority and responsibility to order cessation of hazardous work involving radiological materials until the General Manager/Chief Operating Officer is appraised of the situation and the appropriate precautions are taken. The Radiation Protection Manager has the authority to initiate and direct facility emergency procedures if required to protect personnel or the general public.

9.1.2.2.3 Radiation Protection Technicians

The Radiation Protection Technicians are responsible for the actual monitoring of radiation and environmental conditions and for performing chemical and radiochemical analyses under the direction of the Radiation Protection Manager. The Radiation Protection Technicians determine and evaluate radiation hazards in relation to prescribed limits and perform constant or intermittent radiation monitoring of work areas as the need arises. The Radiation Protection Technicians develop and recommend control or protective measures and check for compliance with appropriate procedures. The Radiation Protection Technicians perform periodic and special radiation surveys of facility areas and equipment to define existing or potential hazards. The Radiation Protection Technicians also package and store any solid radioactive waste in accordance with applicable regulations. The Radiation Protection Technicians perform periodic calibration of survey and analytical instruments.

The Radiation Protection Technicians immediately advise the Radiation Protection Manager of any abnormal radiological condition which could result in a serious hazard and, in the absence of the Radiation Protection Manager, assume responsibility for radiation monitoring during emergency conditions.

The Radiation Protection Technicians develop and implement personnel monitoring activities including the maintenance of personnel exposure records and environmental survey records. The Radiation Protection Technicians maintain the radiation protection and chemistry logs and perform chemical and radiochemical analyses as required. The Radiation Protection Technicians also perform detailed investigations of instances of abnormal contamination or personnel exposure and report findings and recommendations to the Radiation Protection Manager for corrective action.

9.1.2.2.4 Lead Mechanic/Operator

The Lead Mechanic/Operator is responsible to the General Manager/Chief Operating

Officer for the proper maintenance and operation of all facility mechanical equipment necessary to transport, transfer and store spent fuel canisters, including the shipping, transfer, and storage casks, shipping cask trailers and/or rail cars, and the cranes which handle these equipment items. The Lead Mechanic/Operator directs the maintenance crew in the performance of maintenance and operations involving this equipment, and advises the General Manager/Chief Operating Officer of activities being performed. The Lead Mechanic/Operator reviews maintenance records to ensure that proper procedures are being followed and acts on work requests. This person schedules all non-routine and routine maintenance, subject to the review of the General Manager/Chief Operating Officer. The Lead Mechanic/Operator is responsible for executing a preventative and predictive maintenance program and assuring that personnel in this group comply with established procedures and regulations. This person reviews maintenance logs and records to ensure that maintenance required for safe facility operation is performed in a proper manner. The Lead Mechanic/Operator coordinates mechanical activities with all group leaders and engineers, including coordination with the Radiation Protection Manager to ensure that radiation doses to Mechanics are controlled to levels that are as low as is reasonably achievable (ALARA). The Lead Mechanic/Operator keeps the General Manager/Chief Operating Officer advised of any conditions which require his/her attention or that of higher authority.

9.1.2.2.5 Mechanic

The Mechanics are responsible to the Lead Mechanic/Operator for performing facility maintenance and operations associated with the transport, transfer and storage of spent fuel canisters, including the shipping, transfer, and storage casks, shipping cask trailers and/or rail cars, and the cranes which handle these equipment items. Mechanics are responsible for ensuring that maintenance, operation and radiation protection procedures and regulations are followed. The Mechanics are also responsible for maintaining the equipment and maintenance records.

9.1.2.2.6 Lead Instrument and Electrical Technician

The Lead Instrument and Electrical Technician is responsible to the General Manager/Chief Operating Officer for the proper testing and maintenance of facility instrumentation and electric equipment. This person directs the activities of the Instrument and Electrical Technicians and keeps the General Manager/Chief Operating Officer informed of all matters requiring his/her attention. The Lead Instrument and Electrical Technician reviews maintenance records to ensure that proper procedures are followed and acts on work requests; schedules all non-routine maintenance subject to the General Manager/Chief Operating Officer's approval; is responsible for executing a preventative and routine instrumentation testing and maintenance program and assuring that personnel in this group comply with established procedures and regulations; is responsible for assuring that maintenance logs and records are kept and for reviewing these records to ensure that maintenance required for facility safety is performed in the proper manner; and makes recommendations to the General Manager/Chief Operating Officer with regard to instrument improvements or testing procedure changes to improve facility operation.

9.1.2.2.7 Instrument and Electrical Technicians

The Instrument and Electrical Technicians are responsible to the Lead Instrument and Electrical Technician for the repair, testing, maintenance, and approved modification of: facility instrumentation and controls, motors, lighting, and switchgear. These persons conduct test programs, prepare appropriate check lists, and maintain appropriate records of activities; maintain adequate spare parts inventory and ensure that electronic, pneumatic, and electrical test equipment is functioning properly. The Instrument and Electrical Technicians conduct the routine and preventative maintenance programs in a manner consistent with established maintenance procedures and regulations; and keep the Lead Instrument and Electrical Technician informed of all matters requiring his/her attention.

9.1.2.2.8 Lead Quality Assurance Technician

The Lead Quality Assurance Technician has direct access to the Board of Managers (as indicated by the dotted line in Figure 9.1-3) on all quality assurance related issues to ensure independence of quality assurance functions and the effective execution of the Quality Assurance Program. The Lead Quality Assurance Technician reports to the General Manager/Chief Operating Officer for administrative direction relative to the implementation and conduct of the Quality Assurance Program. The Quality Assurance Program is designed to meet Title 10 CFR Part 72, Subpart G, Quality Assurance requirements. The Lead Quality Assurance Technician shall be responsible for assuring that an appropriate Quality Assurance Program is established and effectively executed which verifies by procedures such as monitorings, inspections, and audits that activities classified as Important-to-Safety are performed correctly and in compliance with governing procedures, standards, and regulations. This person will be assisted in quality assurance functions by the Quality Assurance Technician and Quality Assurance Auditor, as necessary.

9.1.2.2.9 Quality Assurance Technician and Quality Assurance Auditor

The Quality Assurance Technician and Quality Assurance Auditor shall be responsible to the Lead Quality Assurance Technician in assisting with administration of the Quality Assurance Program.

9.1.2.2.10 Lead Nuclear Engineer

The Lead Nuclear Engineer is the staff technical expert on nuclear engineering and nuclear physics. This person is concerned with the detailed performance of facility systems which affect nuclear safety, including criticality safety, fission product confinement, decay heat removal from the canisters, and shielding of the canisters. This position reviews all procedural changes and modifications affecting nuclear safety and makes appropriate recommendations to the General Manager/Chief Operating Officer; periodically reviews facility operating data to seek trends which could potentially affect nuclear safety and performs detailed investigations of abnormalities or unusual occurrences related to spent fuel shipping, canister transfer operations, and canister storage; keeps the General Manager/Chief Operating Officer advised of all matters requiring his/her attention or that of higher authority; is responsible for fuel accountability and management; and prepares reports related to the nuclear engineering function.

9.1.2.2.11 Nuclear Engineers

The nuclear engineers ensure compliance with License procedures in the off-site loading of canisters. They are responsible for ensuring that the PFSF "Start Clean / Stay Clean" philosophy is maintained.

9.1.2.2.12 Administrative

The General Manager/Chief Operating Officer is supported by an administrative staff which performs management and clerical services for the facility. Included in this group are the administrative assistant, transportation specialist, secretary, public relations coordinator, and financial/purchasing specialist.

9.1.2.2.13 Security Captain

The Security Captain is responsible to the General Manager/Chief Operating Officer for establishing and maintaining physical security in accordance with the Security Plan approved by the NRC. The Security Captain is responsible for the overall performance of the security force in accordance with the NRC approved Security Force Training and Qualification Plan and Safeguards Contingency Plan, and for ensuring that the security force meets the requirements of the general criteria for Security Personnel. This position manages, directs, and supervises assigned personnel. The Security Captain is responsible for administration, training, reporting and disciplinary control over assigned personnel. The Security Captain receives applications from prospective employees, screens applications, interviews applicants, administers pre-employment tests, completes the background investigations, and hires new security officers for the PFSF. The Security Captain monitors the completion of all required tasks associated with security alarm systems, communication systems, closed circuit television systems, and entry control systems in accordance with security procedures. This position inspects and evaluates security facilities and equipment to ensure that they are being maintained properly and that all systems are functioning / operating in accordance with approved procedures. The Security Captain oversees the involvement of security personnel in implementing the emergency plan and procedures. The security staff is also responsible for monitoring the storage cask temperature monitoring system and reports alarm conditions to the designated personnel.

9.1.2.2.14 Emergency Preparedness Coordinator

The Emergency Preparedness Coordinator is responsible to the General Manager/Chief Operating Officer for ensuring the PFSF is maintained in a state of readiness for effective emergency response in accordance with the Emergency Plan

(EP). This includes responsibility for ensuring the adequacy of EP implementing procedures, that PFSF site personnel are adequately trained in emergency response, that emergency response facilities and equipment are maintained in a state of readiness, and for coordination with offsite agencies and support organizations who may be called upon to provide assistance in the event of an emergency. This position will coordinate drills and exercises, as discussed in the EP, in which individuals demonstrate the ability to perform assigned emergency response functions.

In addition to emergency preparedness, this position is also responsible to the General Manager/Chief Operating Officer for conduct of the overall PFSF site training program, including general orientation training, training on operating/maintenance procedures, and radiological training. The Emergency Preparedness Coordinator is a qualified radiation protection technician, capable of providing backup support to Radiation Protection with radiological monitoring functions on an as-needed basis.

9.1.3 Personnel Qualification Requirements

The staff individual qualification requirements are drafted for the minimum levels. Actual functional resumes will be available once staff is selected during construction in advance of operation under the License.

9.1.3.1 Minimum Qualification Requirements

9.1.3.1.1 General Manager/Chief Operating Officer

At the time of appointment to the active position, the General Manager/Chief Operating Officer shall have ten years of responsible experience within the nuclear power industry. A maximum of four years of the ten years may be fulfilled by academic training on a one-for-one basis. To be acceptable, this academic training shall be in an engineering or scientific field generally associated with the nuclear industry.

The General Manager/Chief Operating Officer shall have a recognized Baccalaureate (Bachelor) or higher degree in an engineering or scientific field generally associated with nuclear power production, fuel storage or radiation protection.

9.1.3.1.2 Radiation Protection Manager

At the time of appointment to the active position, the responsible person shall have a minimum of ten years experience in radiation protection within the nuclear power industry. A maximum of four years of this ten years experience may be fulfilled by related technical or academic training on a one-for-one basis. This person shall have a Bachelor or higher degree in radiation protection or a related field.

9.1.3.1.3 Radiation Protection Technicians

Technicians in responsible positions shall have a minimum of four years of working experience in radiation protection. These personnel should also have a minimum of one year of related technical training.

9.1.3.1.4 Lead Mechanic/Operator

At the time of appointment to the active position, this person shall have a high school diploma or equivalent and a minimum of six years of experience in mechanical maintenance. A Lead Mechanic/Operator shall also be a licensed locomotive operator if a rail spur is developed to the site, and will become a certified storage facility operator prior to facility operation, and a certified welder.

9.1.3.1.5 Mechanics

Mechanics shall hold a high school diploma or equivalent and a minimum of four years experience in mechanical maintenance. All mechanics shall become certified facility operators prior to operating cask handling equipment, and shall become licensed locomotive operators if a rail spur is developed to the site. One mechanic must become a certified welder to provide backup for the Lead Mechanic/Operator. Mechanics should possess a high degree of manual dexterity and the ability to learn and apply new skills in maintenance operations as they are developed and incorporated into facility operations.

9.1.3.1.6 Lead Instrument and Electrical Technician

At the time of appointment to the active position this person shall have a high school diploma or equivalent and a minimum of six years of experience in instrumentation and electrical work. This person should possess a two-year associate degree in the instrumentation, control and electrical field.

9.1.3.1.7 Instrument and Electrical Technicians

Technicians shall have a high school diploma or equivalent and a minimum of four years of working experience in this field. They should have minimum of one year of related technical training in addition to their experience.

9.1.3.1.8 Lead Quality Assurance Technician

This person shall have a high school diploma or equivalent and a minimum of six years experience within the nuclear power industry in a quality assurance position. This person should also have two years technical education in this area.

9.1.3.1.9 Quality Assurance Technician and Quality Assurance Auditor

The Quality Assurance Technicians shall have a high school diploma or equivalent and a minimum of four years experience in the quality assurance field within the nuclear power industry.

9.1.3.1.10 Lead Nuclear Engineer

This person shall have a minimum of a Bachelor degree in nuclear engineering and four years experience in the nuclear power industry.

9.1.3.1.11 Nuclear Engineers

This person shall have a Bachelor degree in Nuclear Engineering.

9.1.3.1.12 Security Captain

This person shall be a High School graduate, a qualified weapons instructor (NRA or equivalent), and have received supervisory skills training. Should have 6 years experience in security at least three of which must be at a nuclear facility.

9.1.3.1.13 Emergency Preparedness Coordinator

This person shall have a high school diploma or equivalent and a minimum of two years experience in emergency preparedness. In addition, this person should have experience in providing training.

9.1.3.2 Personal Qualification Requirements

Each member of the site staff involved with important safety activities will be required to meet the minimum qualifications of the License. Programs for additional site familiarization training and ongoing training and retraining will be maintained to provide a continuously qualified staff. The training program as coordinated by the Emergency Preparedness Coordinator is under the overall direction of the General Manager/Chief Operating Officer.

9.1.4 Liaison with Outside Organizations

During the pre-licensing phase, most of the outside technical support for the PFSF engineering and licensing was employed through the A/E. Oversight was provided by a nuclear engineer on the Technical Committee. The outside organizations providing technical expertise on site selection were directed by the Project Manager; the Chairman of the Board monitored the interface.

The outside organizations supplying the cask storage systems are directed by the Technical Committee and the PFSLLC Project Manager. Their review is performed in accordance with the Quality Assurance Program and is audited by the Quality Assurance Committee.
During construction, the outside organization for installation and construction and the A/E are overseen by the PFSLLC Project Manager. The system to monitor the design includes Technical Committee review and Quality Assurance Committee audits as well as routine telephone conferences.

During the operational phase, the General Manager/Chief Operating Officer shall be responsible for day-to-day contacts with the U.S. Nuclear Regulatory Commission and other regulatory bodies. The authority to hire necessary consulting staff within the guidelines approved by the Board will rest with the General Manager/Chief Operating Officer after consulting the Board Chairman. The acquisition of outside consulting expertise or services will be done in full accordance with the standards outlined in the Quality Assurance Program. All work performed on SSCs that are Important-to-Safety, will be strictly governed by the Quality Assurance Program.

The outside organizations for canister manufacturing are overseen by the General Manager/Chief Operating Officer and technical staff. Their review is performed in accordance with the Quality Assurance Program and is audited by the Quality Assurance staff. Fabrication of canisters and storage/transfer/transportation technology to appropriate quality standards is monitored by facility staff.

9.2 PREOPERATIONAL TESTING AND OPERATION

Prior to loading the PFSF with spent fuel canisters, preoperational, startup, and performance tests will be developed and implemented. The tests will verify the functional operation of structures, systems and components important to safety, including spent fuel shipping and receipt, canister transfer, onsite transport, and storage operations as well as the performance of the storage system components. The tests will verify that the PFSF shipping, transfer, onsite transport and storage systems operate safely and effectively. The results of the tests will be reported as a supplement to this section.

9.2.1 Administrative Procedures for Conducting Test Program

Test procedures will be developed for conducting the tests at the PFSF to ensure that structures, systems, and components satisfactorily perform their required functions. These test procedures will further ensure that the PFSF has been properly designed and constructed and is ready to operate in a manner that will not endanger the health and safety of the public.

The test procedures will include the elements in Section 9.4.1.2 and will detail the type of test, the response expected and the validation method for each component or system tested. All test procedures will be reviewed and approved by the responsible line manager. Procedures involving structures, systems or components important to safety will be reviewed and approved by the site Operations Review Committee (ORC). Revisions necessitated by operational experience, changes to systems or components, new requirements, clerical errors, etc., will be reviewed and approved in the same manner as the original procedure. The test results will be will be documented by the individuals performing the test and will be reviewed and evaluated by the appropriate line organization. Test results for structures, systems and components important to safety will be approved by the ORC.

9.2.2 Preoperational Test Plan

The preoperational test plan will ensure that preoperational tests and dry runs are performed on all PFSF operations involving spent fuel prior to operation and that preoperational tests continue to be conducted on new structures, systems, and components that handle spent fuel to verify performance throughout the life of the facility.

The preoperational test plan will ensure that preoperational test procedures address all aspects of the PFSF, including testing specified in the technical specifications, construction testing, physical facilities testing and operatioal testing. The preoperational test plan will clearly define test objectives, methods for accomplishing the objectives, prerequisites for performing the tests, and acceptance criteria used to evaluate test results. The results of preoperational testing will be used to make necessary changes or modifications to equipment and procedures.

Preoperational tests will be performed in accordance with approved procedures, which will be developed and implemented in accordance with the QA Program described in Chapter 11. The QA Program meets the requirements of 10 CFR 72, Subpart G. Thus, the PFSF meets the general design criteria of 10 CFR 72.122(f), as it relates to testing.

9.2.2.1 Construction Testing

Construction testing will verify requirements for configuration, materials, performance, and quality for structures, systems and components important to safety at the PFSF.

The purpose for construction testing is to ensure that structures constructed at the PFSF meet the requirements of their design, specifications, and code criteria. Construction, materials, operation, or quality that is found not in accordance with the criteria will be referred to the appropriate engineer for resolution, which may include rework, repair, or replacement. Construction testing will be performed on the following:

- Cask storage pad construction,
- Canister transfer building construction, and
- PFSF yard and yard infrastructure construction.

9.2.2.2 Physical Facilities Testing

Spent fuel storage shipping, transfer, transport and security equipment at the PFSF will be functionally tested prior to operation and as required by their applicable codes throughout the life of the facility. The tests will ensure that all equipment that handles spent fuel and all interface systems are working properly as designed and certified. Tests will be performed for both the HI-STORM and the TranStor storage systems equipment.

The purpose for the functional testing of the physical facilities, components, and equipment used at the PFSF is to ensure that they operate properly and will perform as designed in accordance with their perspective licenses, code, and/or vendor criteria. Acceptance criteria and corrective actions for test margins and response times will be specified by the equipment vendors. Components not in compliance will be returned to the vendor for correction or repaired onsite. Correction of components classified as important to safety will be accompanied by documentation identifying resolution of any safety concerns prior to reuse.

The PFSF facilities, components and equipment to be tested and inspected to ensure their proper functioning include:

- Storage system transfer casks,
- Canister downloader equipment,
- Lifting yokes,
- Canister transfer building overhead and bridge cranes and interlocks,
- Storage cask transporter vehicles,
- Heavy haul transport trailers,
- Concrete storage casks,
- Storage cask temperature monitoring equipment,
- Area radiation monitoring equipment,
- Electrical power system,
- Standby diesel generator,
- Security systems equipment,
- Communications systems,
- Fire truck and fire protection equipment

9.2.2.3 Operational Testing

Actual storage system components without fuel will be utilized for preoperational testing prior to PFSF operations. These tests will be performed using both the HI-STORM and the TranStor storage systems. The operational tests will be conducted by appropriate personnel as part of their certification training, and will include full-load testing of all rigging and attachments, and limits of travel on lifting and transfer equipment. These tests will determine the adequacy of plant procedures and estimated worker exposures for ALARA considerations and radiation protection analyses. The results of these tests will be evaluated for potential improvements and alternatives.

The purpose for operational testing is to ensure that the equipment performs as required and that personnel involved in spent fuel shipping, receipt, canister transfer, onsite transport, and storage operations perform their intended tasks in accordance with approved procedures, with ALARA awareness, with efficiency, and without compromising personnel or public safety. This will ensure that canisters containing spent fuel are safely handled and maintained in storage. Failed operational testing will result in repair or replacement of components by their respective vendors and additional training, retraining, or dismissal of personnel. The PFSF operational tests will include:

- Removal of the personnel barrier, impact limiters, and shipping cask from the heavy haul trailer or rail car using the canister transfer overhead bridge crane.
- Uprighting the shipping cask on the shipping cradle and moving the cask from the shipping cradle to the canister transfer building floor using the shipping cask lifting yoke and overhead crane.
- Moving the shipping cask from the cask unloading bay into one of the canister transfer cells using the overhead crane.
- Unbolting the shipping cask lid using automated wrenches and inserting lifting attachments on the canister.
- Setting the transfer cask on top of the shipping cask using the transfer cask lifting yoke and overhead crane.
- Transferring the canister from the shipping cask to the transfer cask using the vendor's canister lifting slings and equipment.
- Moving the transfer cask from the top of the shipping cask to the top of the concrete storage cask using the overhead crane.
- Transferring the canister from the transfer cask into the storage cask using the vendor's canister lifting slings and equipment.

- Ensuring that all steps throughout the transfer process are performed in an ALARA manner to minimize radiation doses, as identified in Chapter 7.
- Transporting the storage cask from the canister transfer building cell to the storage pads and back again using both the cask transporter vehicle and a combination of the overhead crane and cask transporter.
- Transferring the canister from the storage cask back to the shipping cask using the overhead crane as required when shipping fuel offsite.

9.2.3 Operational Readiness Review Plan

An operational readiness review (ORR) will be performed by the PFSF staff prior to operations in order to verify the readiness of the PFSF to begin the shipping, receipt, canister transfer, onsite transport, and storage of spent fuel. The ORR will consist of a programmatic and procedure review, a hardware and staffing review, and a performance assessment of operators, equipment, support staff, and management. The ORR will be conducted in the following major areas:

- Engineering and Technical Support - Onsite technical staff available, design control procedures written and approved, vendor information and manuals available, calculations verified and completed, and drawings as-built and available.
- Licensing - Confirmation of license condition conformance.
- Construction - Construction activities complete as required, drawings entered into document control system, open items resolved, non-conformances corrected, acceptance construction tests completed and approved, and inspections performed and accepted.
- Operations - Operating, off-normal, surveillance, and emergency response procedures written, approved and operationally tested, procedures entered

into document control system, no personnel concerns outstanding, operator aids posted, preoperational testing completed, equipment functional and calibrated.

- Training - Traning procedures written and approved, all personnel tranined to procedures as required, and training material revised to latest PFSF procedures and drawings.
- Radiological Controls - Radiation protection procedures written and approved, health physics personnel trained, radiation postings completed, and radiological monitoring equipment tested and operational.
- Security Controls - Security procedures written and approved, security personnel trained, security equipment tested and operational, and security drills to detect and assess intrusion performed.
- Maintenance and Surveillance - Maintenance and surveillance procedures written and approved, spare parts itentified and available on site, post-maintenance testing completed as required, surveillance inspections and testing completed or ready as required, and startup test plan actions completed.
- Organization and Management - Procedures affecting organization and management are written and approved, management available, fitness for duty requirements completed, staffing adequate, personnel trained and qualified, and security program and personnel in place.
- Fire Protection - Fire protection procedures written and approved, fire detection systems tested and operational, fire protection systems tested and operational including fire truck, fire pumps, and sprinkler systems, fire personnel trained and available, and fire drills performed and determined acceptable.

The ORR will consist of a team leader and safety and technical experts representing the areas of operations, engineering and technical support, maintenance and surveillance, and organization and management. The ORR team is expected to conduct internal meetings with the applicable organizations as required to ensure that all activites reviewed in the ORR are accomplished prior to operation. The ORR team will prepare and issue a report addressing the scope of the ORR and all conclusions, findings, and observations of each review item. The report will be signed off by the ORR Team Leader, PFSF General Manager, and other appropriate managers and made available to the NRC.

9.2.4 Operating Startup Plan

An operating startup plan will be initiated to prepare and implement procedures necessary for the initial arrival of spent fuel and operations to transfer the fuel to storage. The plan will identify specific actions unique to the initial spent fuel loading. The operating startup plan will include tests and reviews of the operating procedures, radiation exposure times and received doses, measured radiation levels of the casks and shielding methods, verification of heat removing features in accordance with the technical specifications, and notification to the NRC of the first loaded cask placed in storage.

The operating startup plan will be implemented for the initial loading of both the HI-STORM storage system and TranStor storage system. Upon completion of the plan, procedures, actions, and equipment will be evaluated for improved operations of subsequent spent fuel shipments.

9.3 TRAINING PROGRAM

The purpose of the training program is to provide the training necessary to ensure the safe, reliable, and cost effective operation of the storage site and to protect the health and safety of site personnel and the general public.

9.3.1 Program Description

The training program will be developed using a "Systematic Approach to Training" which is based upon analysis of the job performance requirements to establish the knowledge level and skills that are required for each position. Explicit learning objectives and performance measures are generated from this analysis. Training plans are then developed which identify training settings, sequences, and materials required. The training program is implemented by conducting the training activities, documenting the training and evaluating the program's effectiveness.

Job descriptions will detail the training, education, and experience requirements for each position. An individual assessment of the employee's needs will be conducted relative to the identified training requirements for each of the respective positions. Training will consist of classroom and on-the job training (OJT), as appropriate, for all individuals, commensurate with their job duties and responsibilities. Training will be structured to meet specific needs of various management and support groups.

It is the intent to hire individuals with the training, education and experience which enable them to perform the assigned tasks, and to provide additional training, as appropriate. There will be an adequate complement of trained and certified personnel prior to the receipt of spent fuel for storage, and throughout the period of the NRC operating license.

The training program depends on a constant evaluation of the job or task to be performed, the work environment, and the training provided, to determine whether the program is effective in producing and maintaining competent employees. Data from these evaluations are used to identify and correct deficiencies and to accommodate changing needs.

The training program will consist of initial training and continuing or refresher training at a frequency of not less than every two years. The training program will be implemented prior to the receipt of spent fuel at the PFSF and during the life of the facility, until termination of the NRC license.

9.3.2 Initial Training

Initial training will consist of general employee training (GET) and job-specific training to provide individuals with the skill and knowledge required to perform their particular duties and responsibilities. For example: it is anticipated that the mechanics will become licensed locomotive operators, and therefore this training will need to be completed prior to operating the locomotives.

9.3.2.1 General Employee Training (GET)

The following topics will be addressed in GET:

- Facility operations and design,
- Instrumentation and controls,
- Emergency Plan and Procedures,
- Security Plan and Procedures,

- Radiation Control Procedures and Practices, including: The nature and sources of radiation and contamination, interactions of radiation with matter, biological effects of radiation, methods of detecting and controlling radiation and contamination, ALARA concepts, facility access and visitor controls, decontamination procedures, use of monitoring and personal protective equipment, regulatory and administrative exposure and contamination limits, and site specific hazards,
- Environmental Protection,
- Quality Assurance,
- Administrative Procedures,
- Normal and Off-Normal Procedures,
- Safety, and
- Fire Brigade.

9.3.2.2 Job Specific and Certification Training

Individuals who operate equipment and controls that have been identified as "important to safety" in the Safety Analysis Report and in the NRC license must be trained and certified. Supervisory personnel who direct the operation of equipment and controls that are "important to safety" must also be certified. These individuals shall be certified as specified in their respective job descriptions. The Operator Training Program will address the following:

- Canister transfer system design and operations,
- Canister transfer system normal and off-normal procedures,
- Storage facility normal and off-normal procedures,
- Transportation normal and off-normal procedures,
- Maintenance,

- Storage cask temperature monitoring system,
- Radiation detection, monitoring, sampling and survey instruments,
- Facility layout and functions,
- Operator responsibility and authority,
- Technical Specifications,
- Normal and emergency communications,
- Transportation,
- Topics covered in General Employee Training (GET), addressed with specific emphasis on operations.

Other tasks that will be unique to this facility may require special training prior to performing them. Some of this training may have to be contracted, either on site or at a vendor's facility.

Whenever additional or job-specific training is required, the "Systematic Approach to Training" should be used to develop the necessary training materials. Training materials will be developed by the site personnel qualified on those particular tasks. Training will also be delivered by individuals qualified on the particular tasks, or by appropriate contractors.

Data collected from test results, job performance results, instructor and trainee critiques will be evaluated and necessary adjustments made to ensure program effectiveness.

9.3.3 Continuing Training

All site personnel will receive GET retraining at a frequency of not more than every two years. The topics selected for GET retraining will include, at a minimum, the subjects covered in initial GET, which appear in Section 9.3.2.

Job Specific and Certification retraining will be provided at a frequency of not more than every two years. Topics for this continuing training may be selected from initial training, NRC bulletins and information notices, major equipment and procedure changes, relevant industry events, and topics designed by the General Manager or requested by other site personnel. All facility processes that could affect the training program should be monitored and analyzed for impact, and the training program adjusted accordingly.

9.3.4 Administration and Records

The Emergency Preparedness Coordinator is responsible for the administration of the training program and for maintaining the training records. The Emergency Preparedness Coordinator will be the primary instructor for GET, and will ensure appropriate, qualified instructors from on site or contracted training services conduct all training activities.

Training records include: dates and hours of training received, outline of subjects (i.e., lesson plan), job performance statements, copies of written examinations, information pertaining to walk-through examinations, practical factors and OJT completed, and re-testing information. Training records will be maintained during the period of an individual's employment and for two years thereafter.

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9.4 NORMAL OPERATIONS

9.4.1 Procedures

Operations important to safety will be conducted at the PFSF in accordance with detailed written and approved procedures. All pre-operational, normal operating, maintenance, testing, surveillance and emergency procedures will be in effect prior to operation of the PFSF. These procedures and changes thereto will be reviewed and approved by the Health Physics amd Quality Assurance organizations, independent of the operating function. Procedures will contain sufficient detail to allow qualified and trained personnel to perform the actions without incident or abnormal event.

9.4.1.1 Categories Of Procedures

9.4.1.1.1 Administrative Procedures

Administrative procedures will provide rules and instructions to all PFSF personnel to provide a clear understanding of operating philosophy and management policies. These procedures will include instructions pertaining to personnel conduct and procedures to develop, review, change and approve the other facility procedures. Administrative procedures include activities to ensure that personnel safety, the working environment, procurement, and other general activities of the facility are carried on at a high degree of readiness, quality and success.

9.4.1.1.2 Radiation Protection Procedures

Radiation protection procedures are used to implement the radiation control program. Radiation protection procedures will assure compliance with 10 CFR 20 and ALARA principles. The radiation protection procedures will describe the acquisition of data, use of equipment, and qualifications and training of personnel to perform radiation surveys, measurements, and evaluations for the assessment and control of radiation hazards associated with the PFSF.

Entrance to and work performed inside the PFSF restricted area will require a radiation work permit and will be reviewed and controlled by radiation protection personnel. Existing radiation protection procedures from the utilities utilizing the PFSF will be used in developing PFSF specific procedures. The radiation protection procedures will ensure safety of personnel and operations at the PFSF.

The operation and use of radiation monitoring instrumentation at the PFSF, including area monitors, personnel monitoring equipment, air sampling, and measurement and sampling techniques will be described in written procedures.

9.4.1.1.3 Maintenance and Surveillance Procedures

Procedures will be established for performing preventative and corrective maintenance and for surveillance of PFSF equipment and instrumentation. Preventative maintenance and surveillance, including calibrations, will be performed on a periodic basis to preclude the degradation of PFSF systems, equipment, and components. Corrective maintenance will be performed to rectify any unexpected system, equipment, or component malfunction, and will be initiated as the need arises.

Procedures will describe the expertise or training required to perform tasks important to safety, special equipment needed, and operational controls. Any projected radiation exposure will be identified, along with the ALARA principles to be applied to minimize such exposure.

9.4.1.1.4 Operating Procedures

The operating procedures will provide instructions for all routine and projected contingency (off-normal) operations, including handling, loading, transporting, and storing of spent fuel, and for all other operations important to safety. Operating procedures will include chemical safety, off-normal occurrences, operation of the cask temperature monitoring system and all operations identified in the technical specifications. The requirements for certification of personnel operating equipment and controls important to safety will be specified in the operating procedures.

9.4.1.1.5 Quality Assurance Procedures

Quality assurance procedures will prescribe the necessary elements of quality oversight to ensure activities important to safety are conducted in a controlled manner, in accordance with 10 CFR 72.122 and all applicable regulations, the PFSF license and technical specifications, the radiation protection program, and approved procedures. The quality assurance procedures will clearly communicate that the responsibility for quality rests with each individual, employee or visitor, who enters the facility.

9.4.1.2 <u>Procedure Preparation</u>

Procedures will be generated for all activities important to safety, and will include:

- Requirements for coordination with and notification of other affected site organizations (i.e., radiation protection, security); before, during and after performance of the activities,
- Personnel and staffing required,
- Instruments, tools, and special equipment needed,

- Identification of functions to be completed,
- Prerequisites for performance of activities,
- Sequence of operations, including projected results,
- Records to be completed and recordkeeping requirements,
- Completion and acceptance criteria, and
- Contingency actions for off-normal conditions.

9.4.1.3 Training On Procedures

All personnel involved in activities important to safety will be trained on the associated procedures prior to conducting the activity. Formal training of personnel on facility procedures will be substantially complete prior to the receipt of radioactive materials at the PFSF. Personnel performing activities important to safety will be certified to perform such functions and will undergo refresher training and testing a minimum of every two years.

9.4.2 Records

9.4.2.1 Records Management System

Records relating to the historical operation of the facility will be maintained by the Technical Support organization, under the responsibility of the Technical Support Manager. Records will be stored in the Administration Building, with copies of records required to be maintained in duplicate, as noted below, maintained in the Security and Health Physics Building. Unless otherwise noted, records will be maintained until termination of the facility license by the NRC.

9.4.2.2 Records To Be Maintained

The following PFSF records will be maintained in accordance with procedures developed specifically for the PFSF. The regulatory reference for each category of records is provided in parentheses.

1. Radiation protection program and survey records(10 CFR 20.2101 to 20.2110);
2. Records associated with reporting of defects and noncompliance (10 CFR 21.51);
3. Records important to decommissioning (10 CFR 72.30(d)), consisting of:
   - Records of spills or off-normal occurrences involving the spread of contamination,
   - As-built drawings and modifications of structures and equipment involved in the use and/or storage of radioactive materials, and locations of possible inaccessible contamination,
   - A document, which is updated a minimum of every 2 years, containing a list of all areas designated at any time as restricted areas as defined in 10 CFR 20.1003, and a list of all areas outside of restricted areas involved in a spread of contamination,
   - Records of decommissioning cost estimates and the funding method used.
4. Records of changes to the physical security plan made without prior NRC approval, which must be maintained for a period of three years from the date of the change (10 CFR 72.44(e) and 72.186(b));
5. Records of changes, tests and experiments, and of changes to procedures described in the SAR (10 CFR 72.48(b)(1));

6. Records showing the receipt, inventory, location, disposal, acquisition and transfer of all spent fuel (10 CFR 72.72(a)), which must be maintained in duplicate as long as the spent fuel is stored at the PFSF and for five years after it is removed or transferred from the PFSF;
7. A copy of the current inventory of all spent fuel in storage at the PFSF (10 CFR 72.72(b), which must be kept in duplicate;
8. A copy of the current material control and accounting procedures (10 CFR 72.72(c));
9. Other records required by license conditions or by NRC rules, regulations or orders (10 CFR 72.80);
10. Records of the occurrence and severity of important natural phenomena that affect the PFSF design must be retained until the license is issued (10 CFR 72.92(b));
11. Quality assurance records, including records pertaining to the design, fabrication, erection, testing, maintenance and use of structures, systems and components important to safety, and results of reviews, inspections, tests, audits, monitoring of work performance, and material analyses (10 CFR 72.174);
12. A copy of the current physical security plan, plus any superseded portions of the plan, which must be maintained for three years after a change is made (10 CFR 72.180);
13. A copy of the current safeguards contingency plan procedures, plus any superseded portions of the procedures, which must be maintained for three years after a change is made (10 CFR 72.184);
14. Operating records, including maintenance, alterations or additions made;
15. Records of off-normal occurrences and events;
16. Environmental survey records;
17. Records of employee qualifications and certifications;

18. Record copies of:
    - SAR and updates,
    - Reports of accidental criticality or loss of special nuclear material,
    - Material status reports,
    - Nuclear material transfer reports,
    - Reports of pre-operational test acceptance criteria and results,
    - Procedures,
    - Environmental Report, and
    - Emergency Plan.
19. Construction Records

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9.5 EMERGENCY PLANNING

An Emergency Plan (EP) has been prepared for the PFSF with an outline and content that complies with the requirements of 10 CFR 72.32(a). The PFSF EP applies specifically to emergencies that could occur at the site.

All accidents and off-normal events evaluated in Chapter 8 of this SAR were considered in the planning basis for development of the PFSF EP. The planning basis includes credible events as well as hypothetical accidents whose occurrence is not considered credible, so as not to limit the scope of emergency planning. Evaluation of the consequences of credible and hypothetical accidents postulated to occur at the PFSF determined that releases of radioactivity would not require response by an off-site organization to protect persons beyond the boundary of the PFSF owner-controlled area. There is a single emergency classification level for events at the PFSF, the Alert classification, which is based on the worst case consequences of potential accidents which are postulated to occur at the PFSF.

Should an off-normal event or accident occur, the PFSF EP requires personnel stationed at the PFSF to notify appropriate emergency response personnel. The emergency response personnel are then responsible for classifying the event in accordance with classification procedures in the PFSF EP and notifying the NRC and local authorities, as stated in the PFSF EP. The emergency response personnel are also responsible for calling out personnel, as necessary, who assemble at the PFSF site to take actions to mitigate the consequences of the emergency, assess radiation and radioactivity levels in the vicinity of the PFSF, and return the PFSF to a safe and stable condition. The design of the PFSF provides for accessibility to equipment on-site and availability of off-site emergency facilities and services in accordance with 10 CFR 72.122(g). The Administration Building at the southeast corner of the PFSF site serves as the emergency response facility, from which emergency response actions are coordinated.

As detailed in the EP, should an emergency event occur, the General Manager (during normal working hours) or the Security Sergeant (at all other times) assumes the position of Emergency Response Leader. The Emergency Response Leader assumes responsibilities for declaring an Alert, as appropriate, and activation of the Emergency Response Organization (ERO), as well as communicating with on-site emergency response personnel and appraising them of the situation at the PFSF. The EP identifies responsibilities and staffing of the on-site ERO and for requesting off-site assistance. Members of the PFSF ERO will be trained on how to respond to various emergencies at the site, as established in the EP.

In order to expedite response to a fire, a fire pumper truck is stationed at the PFSF site, and members of the on-site fire brigade are trained in its operation. An additional fire truck is stationed at the Goshute Skull Valley Reservation and is available for use at the PFSF, if needed. An ambulance is also located at the PFSF to expedite the transport of any seriously injured individuals.

Off-site assistance may be requested as necessary from the Tooele Regional Medical Center, Tooele County Fire Department, and Tooele County Sheriff, all of which are located in Tooele, Utah. Other offset assistance may be requested from industry or the NRC, as specified in the EP.

The Tooele County Emergency Operations Plan was consulted in the development of the PFSF EP, and meetings were held with PFSF personnel and Tooele County officials responsible for emergency response operations to discuss accidents that could possibly occur at the PFSF and gain input in the development of the PFSF EP. The EP was submitted to Tooele County officials for their review and comment in accordance with 10 CFR 72.32(a)(14). Comments were received from the Tooele County, Utah, Department of Emergency Management in a letter dated June 3, 1997, a copy of which, as well as the letter documenting the response to the comments, are included as an attachment to the Emergency Plan.

The EP does not cover actions to be taken for security related events at the PFSF (though it does provide guidance for classifying such events). These actions will be governed by the PFSF Security Plan.

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9.6 DECOMMISSIONING PLAN

9.6.1 PFSF Decommissioning Plan

Prior to the end of the PFSF life, canisters loaded with spent fuel will be transferred from storage casks into shipping casks and transported off site. Since the canisters are designed to meet DOE guidance applicable to multi-purpose canisters for storage, transport and disposal of spent fuel, the fuel assemblies will remain sealed in the canisters such that decontamination of the canisters is not required. Following shipment of the canisters off site, the PFSF will be decommissioned by identification and removal of any residual radioactive material, and performance of a final radiological survey. Additional details on decommissioning are found in License Application Appendix B, Preliminary Decommissioning Plan."

9.6.2 Cost of Decommissioning and Funding Method

10 CFR 72.30(b) requires that the proposed decommissioning plan include a decommissioning cost estimate, a funding plan, and method of assuring the availability of decommissioning funds.

The cost of decommissioning the PFSF facilities and site, excluding the storage casks, is estimated to be $1,631,000. The cost of decommissioning the storage casks is estimated to be $17,000 each. Decommissioning of the PFSF facilities and site will be funded by a letter of credit coupled with an external sinking fund. Decommissioning of the storage casks will be funded by prepayment of $17,000 into an externalized escrow account for each cask to be utilized.

9.6.3 Decommissioning Facilitation

The design features of the dry cask storage concept, to be utilized at the PFSF, provide for the inherent ease and simplicity of decommissioning the facility in conformance with 10 CFR 72.130. Details on these design features and measures that will be taken to both minimize the potential for contamination and facilitate any decontamination efforts which may be required are found in License Application Appendix B, "Preliminary Decommissioning Plan."

9.6.4 Recordkeeping for Decommissioning

Records important to decommissioning, as required by 10 CFR 72.30(d), will be maintained until the PFSF is released for unrestricted use. See Section 9.4.2 for the type of records that will be maintained for the PFSF. These records will be maintained in a secure storage area.

9.7 PHYSICAL SECURITY AND SAFEGUARDS CONTINGENCY PLANS

The purpose of the PFSF Security Program is to establish and maintain physical security capabilities for protecting spent fuel at the PFSF. This Program meets the requirements contained in 10 CFR 72, Subpart H, "Physical Protection," and applicable portions of 10 CFR 73.

The PFSF Security Program is described in the following PFSF documents:

- Security Plan, to include physical protection designs,
- Guard Training & Qualification (T&Q) Plan, and
- Safeguards Contingency Plan.

The Security and Contingency plans are classified as safeguards information and are therefore controlled, and withheld from public disclosure in accordance with 10 CFR 73.21. The T&Q plan is classified under 10 CFR 2.790(d) and as such is also withheld from public disclosure. These plans will be submitted for NRC review under separate cover. A summary description of the PFSF physical protection measures that does not include safeguards information follows.

The Security Force controls access to the PFSF Restricted Area (RA). Access to the PFSF is limited to individuals who require access to perform work related activities. The PFSF Security Force maintains a list of approved individuals authorized unescorted access. Individuals granted access to the PFSF RA are badged to indicate whether access is granted with or without escort. Authorized individuals are required to display issued identification dedicated solely for access to the PFSF. An escort is not required for these individuals. All other personnel are considered visitors, and must sign in on a visitor log before entering the PFSF RA. The log documents the visitor's name, date, time, purpose of visit, employment affiliation, citizenship and name of escort. Visitors are issued display badges that indicate an escort is required. Authorization for access to the PFSF is also contingent upon the individual's meeting PFSF prescribed radiation protection briefings or training. Personnel, hand-carried articles and vehicles that enter the PFSF RA are searched to detect the presence of firearms, explosive and incendiary devices.

The PFSF RA perimeter has an intrusion detection system to immediately detect unauthorized entry, including penetration by stealth. This system is protected against circumvention and tampering. Staffed alarm stations support the Security Program by monitoring perimeter alarms, coordinating security communications and performing closed circuit television surveillance and alarm assessment. Detailed descriptions and capabilities of the PFSF physical protection systems are contained in the PFSF Security Plan.

In accordance with 10 CFR 72.184, the PFSF Safeguards Contingency Plan addresses security responses to a spectrum of threats. For planning purposes, these threats include generic and postulated, site specific contingencies, to include attempted radiological sabotage. Contingency event categories include: (1) Loss of Security Effectiveness, (2) Threats, and (3) Adversary Actions. The Safeguards Contingency Plan provides a Responsibility Matrix that details specific Security Force actions for neutralizing each contingency event. PFSF Security contingency planning involves detailed response procedures, and assistance from local law enforcement, when requested.

As stipulated in Appendix B to 10 CFR 73.55, provisions for training and qualifying Security Force members are contained in the PFSF Guard Training and Qualification (T&Q) Plan. The T&Q Plan identifies all crucial security tasks and the associated

Security Force positions that must be trained and qualified in the respective crucial task. In addition to initial and recurring Security Force training requirements, the T&Q Plan also describes a screening program to determine if the Security Force member's background and physical/mental qualifications meet criteria defined in this plan.

Each commitment made in the Security, T&Q, and Contingency Plans are implemented by written procedures. The regulatory basis for implementing procedures is 10 CFR 73.55(b)(3)(i). Implementing procedures, which are developed, approved and maintained by PFSF Security management, ensure accurate and organized day-to-day security operations.

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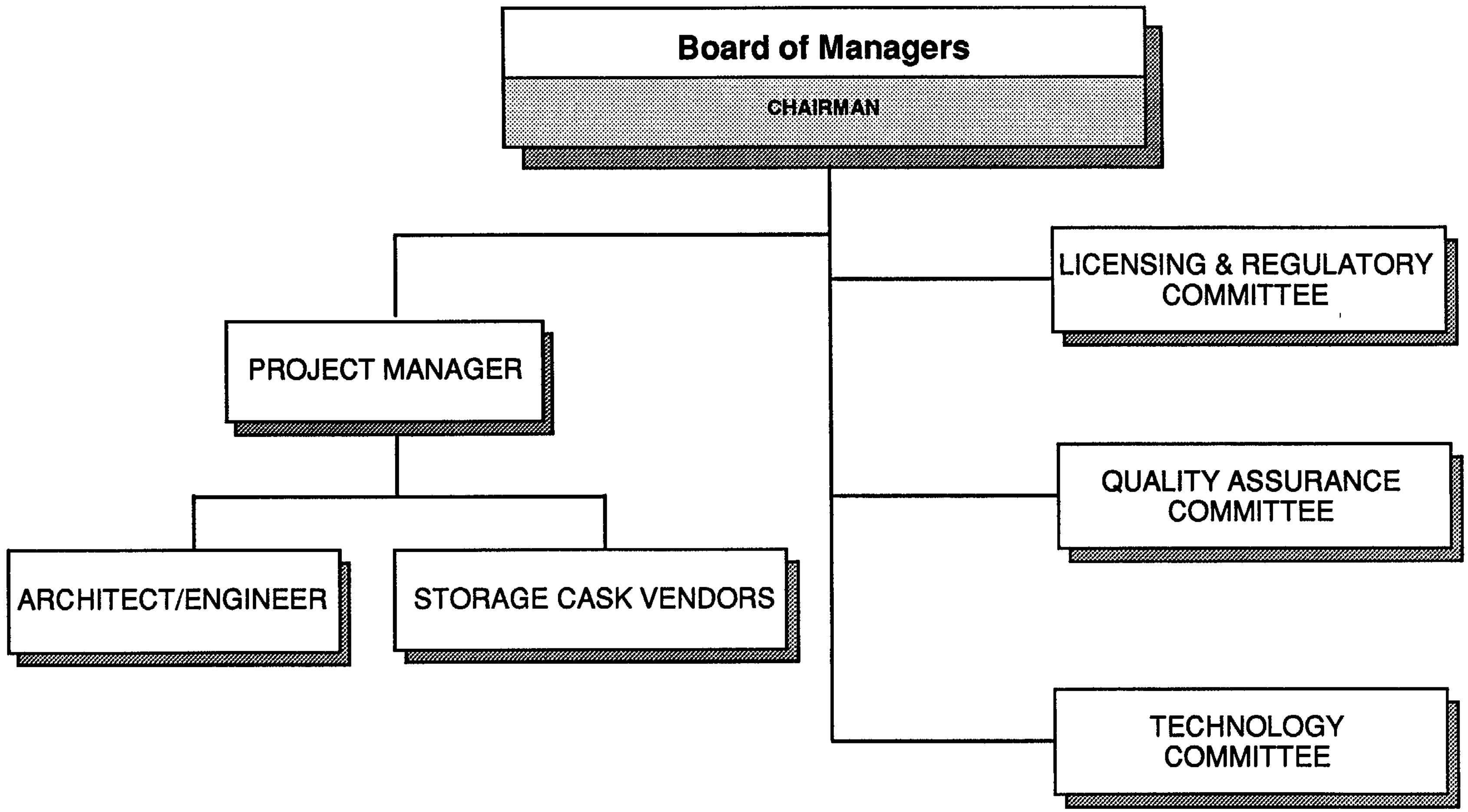

Figure 9.1-1

PRE-LICENSING ORGANIZATION

PRIVATE FUEL STORAGE FACILITY
SAFETY ANALYSIS REPORT

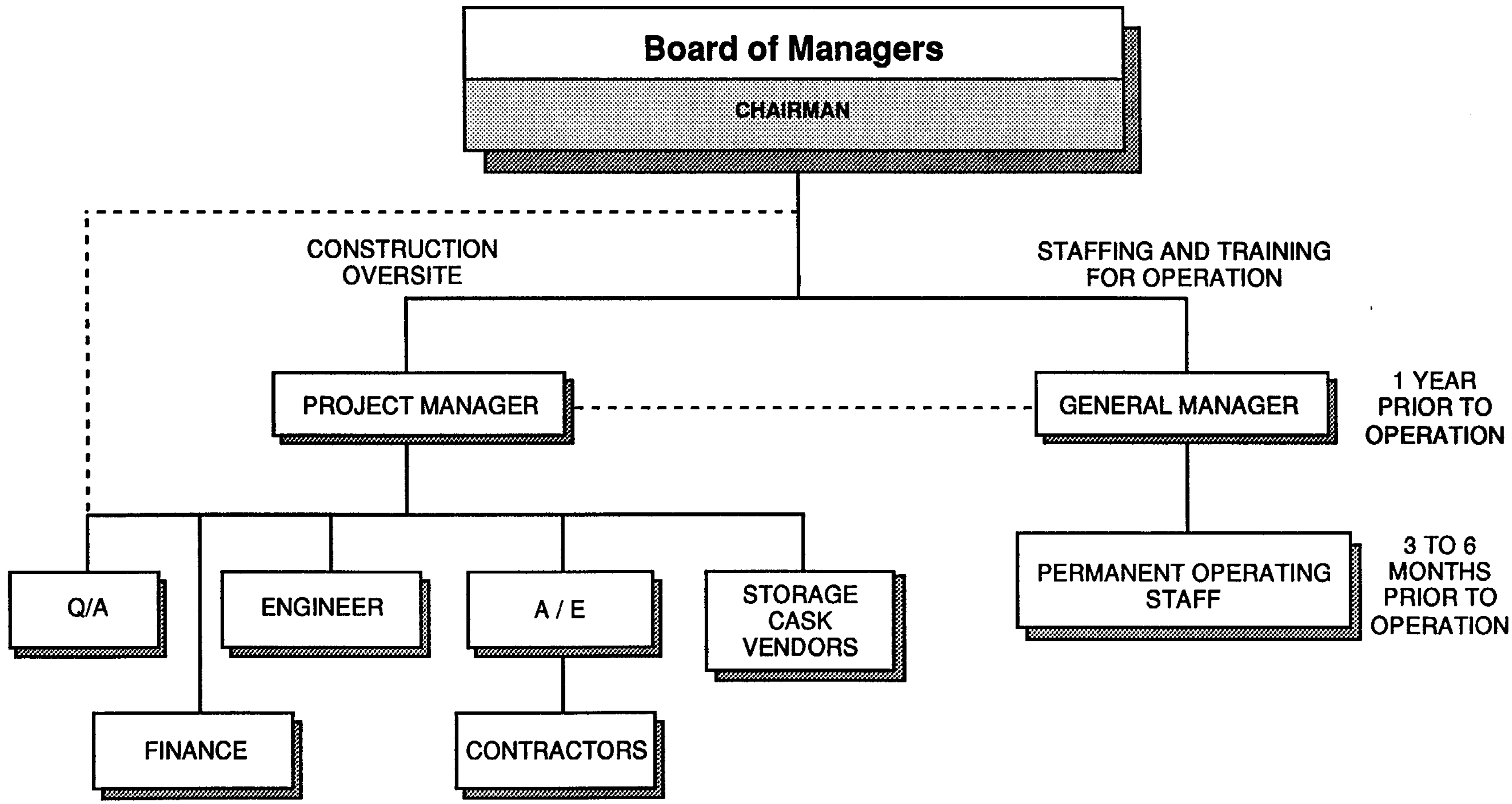

Figure 9.1-2

LICENSING & CONSTRUCTION ORGANIZATION

PRIVATE FUEL STORAGE FACILITY
SAFETY ANALYSIS REPORT

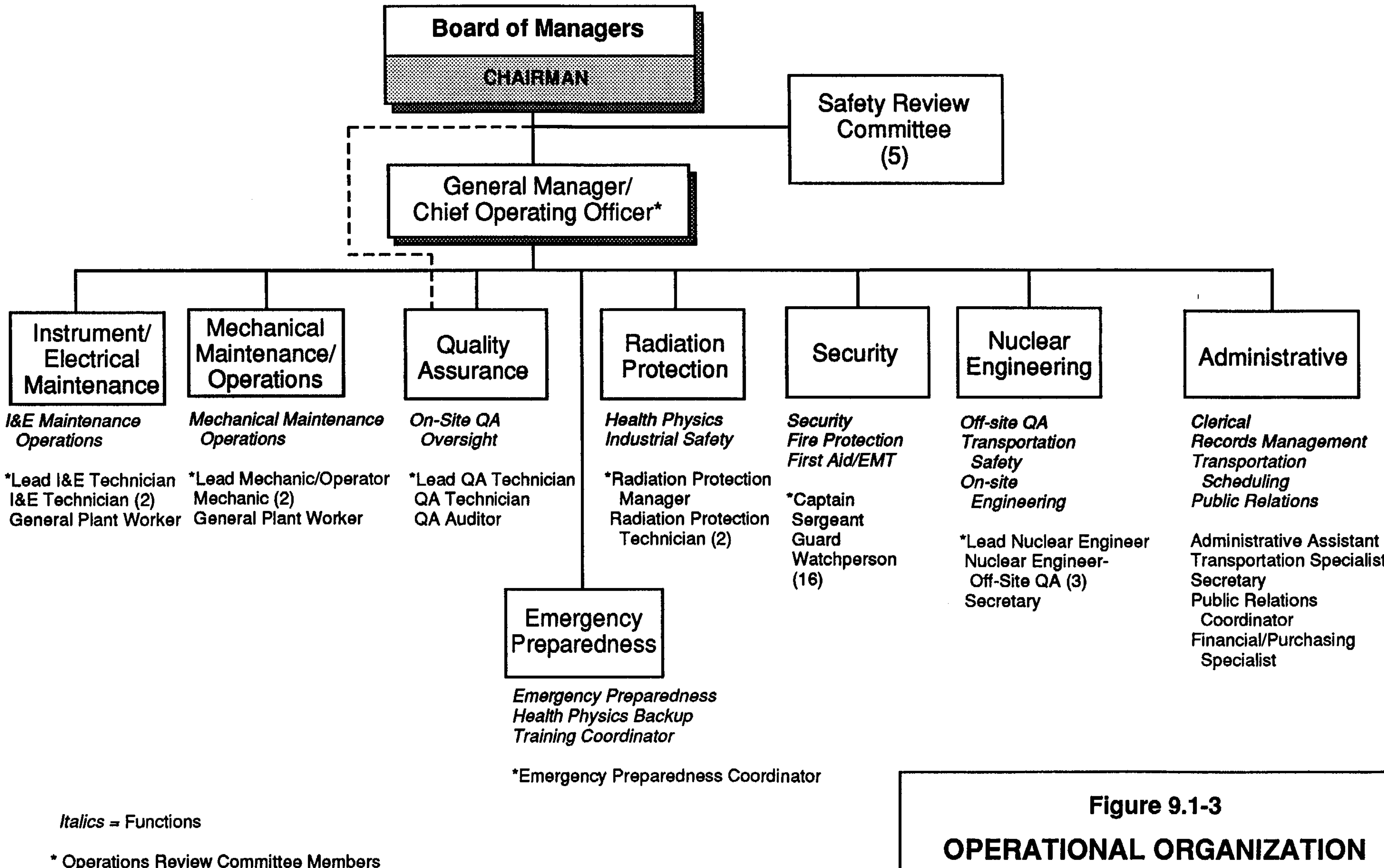

Figure 9.1-3

OPERATIONAL ORGANIZATION

PRIVATE FUEL STORAGE FACILITY
SAFETY ANALYSIS REPORT

CHAPTER 10

OPERATING CONTROLS AND LIMITS

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CHAPTER 10

OPERATING CONTROLS AND LIMITS

10.1 OPERATING CONTROLS AND LIMITS

The dry cask storage systems used at the Private Fuel Storage Facility (PFSF) are passive and require few operating controls during transfer and storage operations. The areas where controls and limits are necessary to ensure safe operation of the storage systems are shown below.

AREAS FOR OPERATING CONTROLS AND LIMITS	CONDITIONS TO BE CONTROLLED
Fuel Characteristics	• Type/Condition • Fuel cladding • Initial enrichment • Burnup • Cooling time • Decay heat • Fuel assembly weight
Concrete Storage Cask	• Surface dose rates • Lift height • Air outlet temperature-initial installation • Air vent operability • Placement on storage pad
Transfer Cask	• Minimum temperature for lifting
Canister	• Authorized types • External surface contamination

The conditions noted are selected based on their importance to the safe operation of the PFSF and on the safety assessments for both normal and accident conditions.

Operating controls and limits for spent fuel loading, canister closure, vacuum drying, helium backfilling, leak testing, and preparation for transportation of the canisters to the PFSF are not addressed in this section since these operations are performed at the originating nuclear power plants and not at the PFSF. These operating controls and limits are addressed in the vendors' storage cask system Safety Analysis Reports (SARs) licensed under 10 CFR 72 (References 1 and 2) and shipping cask system SARs licensed under 10 CFR 71 (References 3 and 4).

10.2 DEVELOPMENT OF OPERATING CONTROLS AND LIMITS

10.2.1 Functional and Operating Limits, Monitoring Instruments, and Limiting Control Settings

This section provides requirements for the controls or limits, which apply to operating variables classified as Important to Safety that are observable and measurable. The operating variables required for the safe operation of the PFSF are:

- Fuel characteristics
- Canisters authorized for use at the PFSF
- Maximum concrete storage cask lift height
- Temperature restrictions for lifting transfer casks
- Placement of concrete storage casks on the storage pads

The specifications for these operating variables are shown on the following pages.

10.2.1.1 Fuel Characteristics

Specification: The spent fuel selected for storage must comply with the specifications below and be independently verified and documented at the originating power plant prior to shipping. Documentation will be maintained at the PFSF identifying each cask, canister, and the fuel assemblies stored in the canister.

Type/Condition: PWR or BWR fuel assemblies listed in tables 3.1-1 and 3.1-2, including MOX fuel and failed fuel confined in approved containers within the canisters, as specified by the storage cask vendors. See section 12.3.1.4 of reference 1 for specifications for damaged BWR fuel.

Fuel Cladding: HI-STORM 100 System:

Zircaloy or Stainless Steel

TranStor Storage System:

Zircaloy or Stainless Steel

Initial Enrichment: HI-STORM 100 System:

PWR: See HI-STORM SAR Table 2.1.3 (Zircaloy) or Table 2.1.8 (stainless steel) (Reference 1).

BWR: $\leq$ 4.2 wt% U-235, and bounded by HI-STORM SAR Figure 2.1.2 for fabricated Boron-10 loading.

TranStor Storage System:

PWR: See TranStor SAR Tables 12.2-2 and 12.2-3 (Reference 2).

BWR: See TranStor SAR Tables 12.2-4 and 12.2-5.

Burnup: HI-STORM 100 System:

See HI-STORM SAR Figure 2.1.6 (Zircaloy) or Table 2.1.8 (stainless steel) for the allowable burnup based on assembly cooling time.

TranStor Storage System:

PWR: See TranStor SAR Tables 12.2-2 and 12.2-3.

BWR: See TranStor SAR Tables 12.2-4 and 12.2-5.

Cooling Time: (Post Irradiation) HI-STORM 100 System:

≥ 5 years - See HI-STORM SAR Figure 2.1.6 (Zircaloy) or Table 2.1.8 (stainless steel) for the assembly burnup versus cooling time.

TranStor Storage System:

≥ 5 years - Cooling times for a given fuel assembly must be adequate to assure the decay heat of the assembly does not exceed the limits specified in the following paragraph. Representative minimum cooling times (for specified burnups) are provided in TranStor SAR Chapter 5, Table 5.1-1.

Decay Heat: Decay heat is estimated based on DOE/RW-0184-R1, July, 1992 (or equivalent spent fuel database), using the characteristics of the spent fuel assemblies.

HI-STORM 100 System:

PWR: Zircaloy ≤ 1.177 kW per assembly
Stainless steel ≤ 0.662 kW per assembly

BWR: Zircaloy ≤ 0.3989 kW per assembly
Stainless steel ≤ 0.079 kW per assembly

TranStor Storage System:

PWR: ≤ 1.083 kW per assembly

BWR: ≤ 0.426 kW per assembly

Fuel Assembly Weights: (incl. Control components)	PWR: ≤ 1680 lb BWR: ≤ 700 lb

Applicability: All PWR and BWR spent fuel to be stored at the PFSF.

Objective: To ensure the maximum fuel cladding temperatures, cask dose rates, and criticality conditions are within the vendors' design values.

Action: Fuel not meeting this specification shall not be accepted at the PFSF.

Basis: This specification is based on the design criteria and associated SARs of the PFSF and storage systems (References 1 and 2). It assures that the design basis remain valid by defining the type/condition of the spent fuel and limits on maximum initial enrichment, irradiation history, minimum post irradiation cooling time, and maximum decay heat. These limits protect the integrity of the spent fuel and the storage systems by ensuring that the storage system thermal, shielding, and criticality analyses are valid for fuel stored at the PFSF.

The limits therefore assure that dose rates associated with the transfer and storage casks do not exceed those analyzed. The maximum fuel assembly weights ensure that structural condition assumptions in the vendors' SARs bound those of the actual fuel being stored.

10.2.1.2 Canisters Authorized for Use at the PFSF

Specification: Two types of canisters are authorized for use at the PFSF, the HI-STORM canister and the TranStor canister. Canister designs will accommodate PWR or BWR fuel, including MOX fuel and failed fuel. The canisters are sealed by welding, vacuum dried, backfilled and pressurized with helium, and leak tested at the originating nuclear power plant, prior to shipment to the PFSF. This limitation assures that only canisters that comply with requirements specified by the vendors of these two spent fuel storage systems are transported to the PFSF. Documentation certifying the canisters are in compliance with vendor specifications is prepared at the originating nuclear power plants during spent fuel loading operations and shall be reviewed at the PFSF upon receipt of the canisters to confirm canisters meet the requirements of this specification.

Canisters received at the PFSF shall meet the following requirements, as confirmed by documentation accompanying the canister shipments:

1. Only canisters designed and fabricated in compliance with vendor specifications for Holtec's HI-STORM spent fuel storage system and SNC's TranStor spent fuel storage system are authorized for use at the PFSF. The steel canisters shall be designed and fabricated in accordance with Section III of the ASME code, as specified by the vendors in their storage system SARs (References 1 and 2).

2. The internals (baskets) of canisters received at the PFSF shall be designed as described in the vendor storage system SARs. Baskets authorized for use at the PFSF are the HI-STORM MPC-24 (PWR spent fuel), the HI-STORM MPC-68 (BWR spent fuel), the TranStor PWR basket, which stores up to 24 PWR spent fuel assemblies, and the TranStor BWR basket, which stores up to 61 BWR spent fuel assemblies. These basket configurations are described in detail and are the subject of analyses in the vendor's storage system SARs. No other basket configurations are permitted to be used at the PFSF.
3. Fuel assembly loading configurations in the above canister baskets shall conform to requirements specified by the vendors in their storage system SARs.
4. Prior to receipt at the PFSF, all canisters shall be vacuum dried, then backfilled and pressurized with helium to the pressures specified by the vendors in their shipping cask SARs (References 3 and 4).
5. Prior to receipt at the PFSF, all canisters shall be sealed by welding with redundant closure welds as specified by the vendors in their shipping cask SARs.
6. Canisters shall be helium leak tested prior to shipment to the PFSF, and any leakage shall be less than the maximum allowable leak rates specified by the vendors in their shipping cask SARs.

Applicability: All loaded canisters received at the PFSF.

Objective: To assure that only those canisters that have been analyzed in the Vendors SAR are stored at the PFSF.

Action: If canisters are received at the PFSF that do not conform with the above requirements, arrangements shall be made for return of the canisters to the originating nuclear power plant. Nonconforming canisters shall not be removed from the shipping cask.

Basis: This specification assures that canisters received at the PFSF, which will be used in spent fuel storage and transfer operations, are in full compliance with vendor specifications and requirements. This assures that analyses associated with canisters in the PFSF SAR and vendor storage system SARs are valid for canister storage and transfer operations conducted at the PFSF. Storage of canisters not meeting these requirements is not allowed at the PFSF, as their use could be outside of previously analyzed conditions.

10.2.1.3 Maximum Concrete Storage Cask Lift Height

Specification: The concrete storage cask lift height shall not exceed:

- 10 inches for the HI-STORM 100 storage cask.
- 18 inches for the TranStor storage cask.

Applicability: All concrete storage casks loaded with spent fuel at the PFSF.

Objective: To ensure storage casks are not lifted above the vendors' analyzed safe handling height.

Action: If the specified height is exceeded, immediately lower the cask to within the specification value.

Basis: The HI-STORM 100 and TranStor concrete storage cask vendors have determined that no unacceptable damage would occur to the storage cask in the event of vertical drops from the above specified heights. The canisters would retain their leak-tight integrity and continue to provide the confinement boundary; damage would not prevent removal of fuel assemblies; a criticality accident would not occur; and the concrete cask would retain its structural integrity and continue to provide physical protection and shielding of the canister. The vendors' storage cask safe handling height restrictions are described in Chapter 12 of their respective SARs (References 1 and 2).

10.2.1.4 Minimum Temperature for Lifting the TranStor Transfer Cask

Specification: The transfer cask shall only be used to move the loaded canister if the transfer cask temperature is -3° F or above.

Applicability: All loaded canister transfer operations using a TranStor transfer cask.

Objective: To avoid the potential for brittle failure.

Action: Confirm that the transfer cask temperature is -3° F or above before using the transfer cask for transfer operations.

Basis: Nil ductility transition temperature / Charpy test requirements are addressed in Chapter 12 of the TranStor SAR (Reference 2).

10.2.1.5 Ambient Temperature Limits for Handling a Loaded HI-TRAC Transfer Cask

Specification: The loaded HI-TRAC cask shall not be handled in an environment where the ambient temperature is below 32° F, except as noted below under action item a, or above 100° F.

Applicability: All loaded HI-TRAC transfer casks.

Objective: To avoid the potential for brittle failure and damage resulting from freezing of the neutron water jacket and to avoid exceeding the upper ambient temperature used in the thermal analysis.

Action:

a. If the ambient temperature is above 0° F and below 32° F, the HI-TRAC may be handled if either a thermal analysis has been performed demonstrating that the MPC decay heat will prevent freezing of the water jacket or a 25 percent solution of ethylene glycol and demineralized water is used in the water jacket. Otherwise do not handle the loaded HI-TRAC.

b. In the event that the ambient temperature is expected to reach or exceed 100° F, do not handle the loaded HI-TRAC.

c. In the event that the ambient temperature is expected to reach or go below 0° F, do not handle the loaded HI-TRAC.

Basis: The HI-TRAC thermal analysis is based on an upper ambient temperature of 100°F. Operating the HI-TRAC at or below 32°F may lead to freezing and subsequent damage to the neutron shield jacket. Handling the HI-TRAC below an ambient temperature of 0°F may present a risk of brittle fracture.

10.2.1.6 Placement of Concrete Storage Casks on the Storage Pad

Specification: All loaded concrete storage casks shall be placed in a storage array with a minimum center-to-center spacing of 15 ft. The cask centerline shall also be a minimum of 7 ft, 6 in from any edge of the storage pad.

Applicability: All loaded concrete storage casks.

Objective:
a. To ensure the storage casks thermal margins are not exceeded.
b. To ensure the cask remains on the pad and does not collide with another cask during a seismic event.

Action: The center-to-center and distance to edge of pad spacing shall be measured upon initial storage cask placement. After a seismic event of magnitude greater than 5.0 Richter at the PFSF, as determined by the National Earthquake Information Center, Golden, CO., verify spacing specified above. If required, restore center-to-center and distance to edge of pad spacing.

Basis: The thermal analysis for the storage cask system utilizes the above specified spacing. Additionally, seismic analysis shows that this spacing will prevent any storage cask impact during a seismic event.

10.2.2 Limiting Conditions for Operation

This section provides requirements for the limiting conditions for operation, which specify the lowest acceptable levels of performance that are needed to assure that the PFSF can fulfill its safety functions. Limiting conditions for operation required for the safe operation of the PFSF are:

- Canister external surface contamination
- Storage cask surface dose rate
- Storage cask air outlet temperature-initial installation
- Storage cask air vent operability

The specifications for these limiting conditions for operation are shown on the following pages.

10.2.2.1 Canister External Surface Contamination

Specification: The removable surface contamination on the outer surface of the canister shall be less than 22,000 dpm/100 sq cm from beta and gamma and less than 2,200 dpm/100 sq cm from alpha emitting sources.

Applicability: All canisters containing spent fuel at the PFSF.

Objective: To assure contamination control prior to handling the canister and placing it in the storage cask.

Action: If the above specified limits are exceeded, return the canister and shipping cask to the originating nuclear power plant for decontamination.

Surveillance: A contamination survey for removable surface contamination shall be taken on the accessible external surfaces of the canister. The survey shall be conducted after the shipping cask lid is removed and before removing the canister from the shipping cask.

Basis: Specified values are consistent with the requirements of 49 CFR 173.443 for transporting spent fuel shipping containers.

10.2.2.2 Concrete Storage Cask External Dose Rate

Specification: HI-STORM 100 System:

The total contact dose rates on the loaded HI-STORM Storage Cask shall be:

1. Less than 45 mrem/hr when measured around the mid-height axial plane,
2. Less than 20 mrem/hr when measured at the top center of the lid, and
3. Less than 70 mrem/hr when measured at the center of the air inlet and outlet vent screens.

TranStor System:

The dose from all types of radiation one meter from the cask surface shall be:

1. Less than 15 mrem/hr on the side and 200 mrem/hr on the top for Zircaloy clad fuel, and
2. Less than 30 mrem/hr on the side and 200 mrem/hr on the top for stainless steel clad fuel.

Applicability: All storage casks loaded with spent fuel at the PFSF.

Objective: To verify acceptable fuel has been loaded into the canister and to maintain dose rates ALARA.

Action: If the measured dose rate exceeds the specification, the fuel loading records shall be reviewed and verified to be in compliance with section 10.2.1.1. If the fuel loading is correct, an analysis shall be utilized to determine acceptable onsite and offsite dose levels to demonstrate compliance with 10 CFR 20 and 10 CFR 72 radiation protection requirements. Portable shielding may be used to lower the dose rates within acceptable limits.

Surveillance: Measure the dose rates prior to moving the loaded storage cask to the PFSF storage area.

Basis: The specified dose limits are consistent with the dose levels shown in HI-STORM SAR Section 12.3.16 and TranStor SAR section 12.2.1.3 and establish maximum levels that are deemed ALARA and acceptable for PFSF personnel.

10.2.2.3 Concrete Storage Cask Air Outlet Temperature-Initial Installation

Specification: The equilibrium air temperature, after initial installation, at the outlet of a loaded storage cask shall not exceed ambient by more than 125° F for the HI-STORM storage cask and 100° F for the TranStor storage cask.

Applicability: All storage casks loaded with spent fuel at the PFSF.

Objective: To ensure that the air vents are operable and that the temperatures of the fuel cladding, canisters, and the storage cask concrete do not exceed their specified limits.

Action: If the cask outlet temperature is greater than the values specified above, the first action is to check all inlet and outlet ducts for airflow blockage. If environmental factors are ruled out as the cause of the excessive air temperature, and if the correct fuel loading has been verified, then this condition is not addressed in the SAR and will require additional temperature measurements and/or analysis to justify acceptability of the actual cask performance.

Surveillance: TranStor Storage System:
To verify proper operation of the loaded storage cask, temperature measurements shall be conducted at intervals not to exceed 26 hours until the cask has reached equilibrium. The air temperature shall be the average of measurements at all four outlets.

HI-STORM System 100:

Temperature measurements shall be conducted at the intervals listed below:

1. Immediately following the installation of the lid.
2. 24 hours after the installation of the lid.
3. 7 days after installation of the lid.

Basis: The vendors' storage cask thermal analyses are described in their respective storage cask system SARs (References 1 and 2). These analyses ensure that the fuel cladding, canister, and cask concrete will be maintained at temperatures below material degradation levels.

10.2.2.4 Concrete Storage Cask Air Vent Operability

Specification: Air vent operability shall be determined by monitoring cask temperatures as indicated below:

1. The HI-STORM lid bottom plate temperature (as measured by an installed thermocouple) shall not exceed 368° F above ambient temperature.
2. The TranStor storage cask inner liner temperature (as measured by an installed thermocouple) shall not exceed 126° F above ambient temperature.

Applicability: All storage casks loaded with spent fuel at the PFSF.

Objective: To ensure that the air vents are operable and that the long-term temperature of the fuel cladding and the storage cask concrete do not exceed their specified limits.

Action: In the event the above specified temperatures are exceeded, the following actions shall be taken within 24 hours:

1. Inspect all air inlets and outlets on the subject storage cask for obstructions. Remove any obstructions. Repair or replace damaged screens as applicable.
2. Check, repair, or replace faulty thermocouples.
3. Recommence surveillance.

Surveillance: Storage cask temperatures shall be monitored continuously utilizing permanently installed thermocouples.

Basis: The vendors' storage cask thermal analyses are described in their respective storage cask system SARs (References 1 and 2). These analyses ensure that the fuel cladding and cask concrete will be maintained at temperatures below material degradation levels.

10.2.3 Surveillance Requirements

This section addresses the surveillance specifications that are placed on those Structures, Systems, and Components classified as Important to Safety during operation or necessary to prevent or mitigate the consequences of accidents in accordance with 10 CFR 72.128(a). The surveillance specifications required for the HI-STORM 100 and TranStor storage systems and the PFSF are:

- Fuel Parameters - The parameters of the fuel in each canister are verified upon arrival of the shipping cask at the PFSF by review of the loading and shipping documentation from the originating nuclear power plant.

- Canister External Surface Contamination - Contamination of the accessible external surfaces of each loaded storage canister is assessed prior to removing the canister from the shipping container.

- Concrete Storage Cask External Dose Rate - Each loaded storage cask external dose rate is measured prior to moving the cask to the storage area to assure that only acceptable fuels have been loaded and that the radiation dose levels are within the specified limits.

- Concrete Storage Cask Thermal Performance - The ambient temperature and specified temperatures for all storage casks on the storage pads are continuously measured from the time of cask placement on the storage pad to the time the cask is removed from the storage pad.

- PFSF Safety Status - A visual surveillance of the storage casks and surrounding area is performed on a quarterly basis to determine that no significant damage or deterioration of the exterior of the storage casks has occurred. Surveillance shall also include observation to determine that no significant accumulation of debris on cask or storage pad surfaces has occurred.

- Visual Inspection of Air Inlets and Outlets - Visual inspection of the concrete storage cask air inlets and outlets is performed on a quarterly basis to assure that no blockage exists that could prevent air movement and thereby inhibit cooling of the canister.

10.2.4 Design Features

Operating controls and limits cover the design characteristics of special importance to each of the physical barriers and to the maintenance of safety margins in the design. Design features of the HI-STORM 100 and the TranStor storage systems that are important to safe operation are addressed in their respective SARs (References 1 and 2).

10.2.5 Administrative Controls

The operation of the PFSF is managed by the Private Fuel Storage L.L.C. (PFSLLC). Day-to-day surveillance activities and security are performed by a full-time PFSF staff. The PFSLLC and the PFSF staff are responsible for monitoring and providing safe control of the facility under normal, off-normal, and accident conditions as required by 10 CFR 72.122(j). The administrative controls shall be in accordance with the requirements of the PFSLLC procedures and PFSF Technical Specifications.

10.3 REFERENCES

1. Topical Safety Analysis Report for the Holtec International Storage and Transfer Operation Reinforced Module Cask System (HI-STORM 100 Cask System) Holtec Report HI-951312, Revision 1, Docket 72-1014, January, 1997.

2. Safety Analysis Report for the TranStor Storage Cask System, SNC-96-72SAR, Sierra Nuclear Corporation, Revision B, Docket 72-1023, March, 1997.

3. Topical Safety Analysis Report for the Holtec International Storage, Transport, and Repository Cask System (HI-STAR 100 Cask System), Holtec Report HI-951251, Revision 4, Docket 71-9261, September 1996.

4. Safety Analysis Report for the TranStor Shipping Cask System, SNC-95-71SAR, Sierra Nuclear Corporation, Revision 1, Docket 71-9268, September 1996.

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

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CHAPTER 11

QUALITY ASSURANCE

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CHAPTER 11

QUALITY ASSURANCE

11.1 QA PROGRAM DESCRIPTION

This section addresses the means by which PFSLLC meets the requirement to establish and implement a Quality Assurance (QA) program in accordance with 10 CFR 72, Subpart G. The PFSLLC QA Program (Reference 1) was approved by the NRC on November 3, 1996 for use under 10 CFR 71, Subpart H (Docket 71-0829). This PFSLLC QA Program is being used to satisfy the requirements of 10 CFR 72, Subpart G. Implementation of the QA program as described below ensures that quality standards are met per the requirements of 10 CFR 72.122(a).

11.1.1 Organization

Section 1.0 of the PFSLLC QA Program describes the PFSLLC organization responsible for the establishment and execution of the QA Program at the PFSF.

The PFSLLC organization during the pre-licensing and the licensing and construction stages of the PFSF is depicted on Figures 9.1-1 and 9.1-2 respectively, and is comprised of a Board of Managers, Project Manager, Architect Engineer, Storage Cask Vendors, Licensing and Regulatory Committee, Technology Committee, QA Committee, and other PFSLLC supporting organizations committees. The key QA responsibilities are:

- Board of Managers - Responsible for assuring the effective and efficient implementation of the QA Program.

- Architect/Engineer (A/E) - Responsible for the design of the PFSF. The A/E is responsible for performing design and design control activities in accordance with an approved QA program.

- QA Committee - Responsible for maintaining the QA Program, assessing the effectiveness of the program by performing independent assessments and audits, and qualifying subcontractors and suppliers. The QA Committee has the authority to "stop work" in cases where project activities are not in compliance with specifications, procedures, codes, standards, or regulations or when the quality of Structures, Systems, and Components (SSCs) are indeterminate.

  The QA Committee is an independent organization reporting to the Board of Managers and shall not be responsible for day to day activities, costs, or schedules. The QA Committee has the organizational freedom and authority to identify quality problems; to stop unsatisfactory work and assure that proper processing, delivery, installation, or use is controlled until proper disposition of a nonconformance, deficiency, or unsatisfactory condition; to initiate, recommend, or provide solutions; and to verify implementation of solutions. The QA Committee shall have sufficient access to all work areas necessary to perform their duties.

  The QA Committee oversight activities will include contract/specification preparation, oversight during procurement and fabrication activities, and receipt inspection. Fabrication oversight will include surveillance, inspection, and audits to ensure fabricator compliance with all contract and licensing documents. On-site shop inspections will be a large element of the oversight plan. Typical oversight activities include (but are not limited to) review of procurement documents, drawings, specifications, personnel qualifications, test and NDE reports, non-conformance reports, and as-built drawings. Contract documents will ensure that PFS personnel have access to the fabrication facilities to perform the above functions.

The PFSLLC QA organization is responsible for providing sufficient QA staff to provide assurance that activities that are important to safety have been correctly performed.

The QA Committee, through continuing involvement, evaluations, assessment, surveillance's, and audits, is responsible for ensuring that the PFSLLC QA policies and objectives are met by the PFSLLC and its subcontractors.

The PFSLLC organization, during the operation stage of the PFSF, is depicted on Figure 9.1-3 and is comprised of a Board of Managers, General Manager, Lead QA Technician, and other manager positions. The key QA responsibilities are:

- Board of Managers - Responsible for assuring the effective and efficient implementation of the QA Program.
- Lead QA Technician - Responsible for implementing and maintaining the QA Program during operation of the PFSF.

11.1.2 QA Program

Section 2.0 of the PFSLLC QA Program describes the QA program that provides control over all activities affecting quality at the PFSF.

The QA Program is comprised of the QA Program Description and QA Procedures. The QA Program provides control of activities affecting quality in SSCs that are important to safety.

The QA classification of SSCs at the PFSF classified as "Important to Safety" are shown on Table 3.4-1 of this SAR.

The level of quality applied to those systems that are not designated as important to safety shall be specified within implementing procedures, specifications, plans, and drawings.

Individuals responsible for QA functions shall be trained as required with implementing procedures. When required by applicable codes and standards, personnel requiring qualification shall be appropriately certified in accordance with approved procedures.

The QA Program shall be reviewed at established intervals to assure that it is being effectively implemented and is adequate. All QA program requirements shall be required of subcontractors and suppliers and translated within procedures, instructions, purchase orders, contracts, specifications, plans, and drawings.

The QA Program shall be implemented by the member organizations of the PFSF through the implementation of the PFSLLC QA Program. Other organizations responsible for quality shall provide services in accordance with an approved QA program. The QA Program of the A/E, Stone and Webster Engineering Corporation (SWEC), (Reference 2), has been approved by the NRC as meeting the requirements of 10 CFR 50 Appendix B.

11.1.3 Design Control

Section 3.0 of the PFSLLC QA Program establishes the measures to implement a design control process through approved procedures to ensure that SSCs are designed in accordance with applicable regulatory requirements, codes, and standards.

Procedures shall describe and control the design and any changes from inception through final approval, release, distribution, and implementation; provide identification and control of design interfaces and coordination among participating design organizations; and provide for a design review by qualified personnel.

11.1.4 Procurement Document Control

Section 4.0 of the PFSLLC QA Program establishes the measures to assure that procurement documents covering material, equipment, and services specify appropriate quality requirements. The procurement documents shall specify or reference the applicable requirements, design bases, codes, and standards to assure quality.

Procedures shall delineate requirements for the preparation, review, approval, and control of procurement documentation for all procurement activities and shall provide the requirements for the qualification of suppliers, objective evidence of supplier quality, and the assignment of quality requirements to procurement documents.

To the extent necessary to assure quality, procurement documents shall require suppliers of material, equipment, and services to have a QA program complying with the pertinent provisions of 10 CFR 71, Subpart H or 10 CFR 72 Subpart G. The requirements of 10 CFR 21 shall be specified on procurement documents, as applicable.

11.1.5 Instructions, Procedures, and Drawings

Section 5.0 of the PFSLLC QA Program establishes the measures to assure that activities affecting quality are performed in accordance with instructions, procedures, and drawings that are developed, reviewed, approved, utilized, and controlled in accordance with approved procedures.

Procedures and instructions shall ensure that sufficient records are specified, reflect the quality of the work performed, and comply with appropriate codes, standards and regulatory requirements.

11.1.6 Document Control

Section 6.0 of the PFSLLC QA Program establishes the measures to control the issue, use, review, approval, distribution, and revision of quality related documents, which shall be prepared, reviewed, and approved by qualified personnel using document control procedures.

Procedures shall identify individuals and/or organizations responsible for the control, review, approval, and issuance of documents and shall specify the required reviews, approvals, and distribution of documents.

11.1.7 Control of Purchased Materials, Equipment, and Services

Section 7.0 of the PFSLLC QA Program establishes the measures to assure that purchased materials, equipment, and services conform to the procurement documents.

Procedures shall be used to evaluate and select suppliers and ensure that supplier designs and performance are under the control of an NRC-approved QA program.

11.1.8 Identification and Control of Materials, Parts, and Components

Section 8.0 of the PFSLLC QA Program establishes the measures for the identification and control of materials, parts, and components from their receipt through installation or use.

Procedures shall ensure the identification and control of materials, parts and components.

11.1.9 Control of Special Processes

Section 9.0 of the PFSLLC QA Program establishes the measures to assure that special processes are controlled and accomplished by qualified personnel using qualified procedures in accordance with applicable codes, standards, specifications, criteria, and other special requirements.

Procedures implemented for special processes shall specify the qualifications of personnel, the proper equipment to be used, and control of materials and supplies.

11.1.10 Inspection

Section 10.0 of the PFSLLC QA Program establishes the measures for the inspection of activities affecting quality to verify conformance with approved procedures, drawings, and specifications.

Procedures implemented for inspection activities shall delineate inspection methods, characteristics, and documentation and ensure that inspections are performed by qualified personnel.

11.1.11 Test Control

Section 11.0 of the PFSLLC QA Program establishes the measures for a test program, which demonstrates that SSCs will perform satisfactorily in service.

Procedures implemented for testing shall incorporate or reference requirements and acceptance criteria contained in applicable design documents and specifications and ensure that test results are documented and evaluated to ensure that test requirements have been satisfied.

11.1.12 Control of Measuring and Test Equipment

Section 12.0 of the PFSLLC QA Program establishes the measures to ensure that tools, gauges, instruments, and measuring and test equipment, utilized to verify conformance with established requirements, are controlled, calibrated, and periodically adjusted as required.

Inspection, test, and work procedures shall assure that tools, gauges, instruments, and other inspection, measuring, and test equipment and devices used in activities affecting quality are of the proper range, type, and accuracy to verify conformance to established requirements and test parameters.

11.1.13 Handling, Storage, and Shipping

Section 13.0 of the PFSLLC QA Program establishes the measures to control the handling, storage, shipping, cleaning, packaging, and preservation of materials and equipment to prevent damage, deterioration, or loss through shipment, installation, or use.

Procedures shall delineate the requirements for handling, storage, shipping, cleaning and preservation of materials and equipment; describe special equipment to be used, protective environments and coatings, or other protective measures; and specify required documentation.

11.1.14 Inspection, Test, and Operating Status

Section 14.0 of the PFSLLC QA Program establishes the measures, which indicate the inspection, test, and operating status of SSCs.

Procedures shall include measures to preclude the bypassing of inspections and tests, to prevent the operation of equipment or systems until authorized by qualified personnel, and to specify appropriate status indicators.

11.1.15 Nonconforming Materials, Parts, or Components

Section 15.0 of the PFSLLC QA Program establishes the measures to control materials, parts, or components that do not conform to specified requirements in order to prevent their inadvertent use or installation.

Procedures shall provide requirements for identifying, segregating, reporting discrepancies, and dispositioning of nonconforming items and reporting nonconforming items to the affected organizations. Procedures shall identify methods for reporting adverse quality conditions in accordance with 10 CFR 21 requirements.

11.1.16 Corrective Action

Section 16.0 of the PFSLLC QA Program establishes the measures to assure that conditions adverse to quality are promptly identified and corrected.

Procedures shall ensure that conditions adverse to quality are identified and reported to the appropriate personnel, and that the cause and corrective action necessary to prevent recurrence of significant conditions adverse to quality are identified, implemented, and followed-up to verify corrective action effectiveness.

11.1.17 QA Records

Section 17.0 of the PFSLLC QA Program establishes the measures for maintaining records of activities affecting quality.

Procedures shall provide controls for the identification, receipt, storage, preservation, safekeeping, traceability, retrieval, and dispositioning of records.

11.1.18 Audits

Section 18.0 of the PFSLLC QA Program establishes the measures for planned and documented audits to verify compliance and effectiveness with all aspects of the PFSLLC QA Program.

Procedures shall ensure that audits are performed by appropriately trained personnel; audit results are documented, reported, and reviewed by appropriate levels of supervision and management; and responsible persons in the area audited take necessary action to correct reported deficiencies.

11.2 REFERENCES

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

2. Stone & Webster Engineering Corporation Standard Nuclear Quality Assurance Program, Docket No. 99900509.

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