CONSOLIDATED

SAFETY EVALUATION REPORT

CONCERNING THE

PRIVATE FUEL STORAGE FACILITY

- MARCH 2002 -

Docket No. 72-22

PREFACE

On September 29, 2000, the Nuclear Regulatory Commission (NRC) staff issued its "Safety Evaluation Report Concerning the Private Fuel Storage Facility" (PFS SER). Subsequent to the issuance of the PFS SER, Private Fuel Storage, Limited Liability Company (PFS) submitted four license application (LA) amendments to the NRC staff. These included LA Amendment No. 20 on January 19, 2001; LA Amendment No. 21 on January 25, 2001; LA Amendment No. 22 on March 30, 2001; and LA Amendment No. 23 on November 21, 2001. The staff reviewed these LA Amendments and issued two SER Supplements. SER Supplement No. 1 was issued on November 13, 2001. That supplement included changes to Chapter 15 of the PFS SER to address the staff's evaluation of PFS's revised aircraft crash and cruise missile hazard analyses. SER Supplement No. 2 was issued on December 21, 2001. That supplement included changes to Chapters 2, 4, 5, 6, 7, 11, and 15 of the PFS SER to address the staff's evaluation of PFS's revised seismic analyses and proposed facility design. The chapters of the PFS SER that were revised following the NRC staff's receipt of the four LA amendments contain specific references to those LA amendments, as appropriate.

For the convenience of the parties to the PFS licensing proceeding before NRC's Atomic Safety and Licensing Board, the NRC staff has prepared a version of the PFS SER that consolidates the revisions from SER Supplements Nos. 1 and 2 with the PFS SER into a single document. In this consolidated SER (March 2002), the NRC staff has interfiled revised paragraphs and sections from SER Supplements Nos. 1 and 2, into the September 29, 2001, PFS SER, removing the original information where appropriate. Change bars are included in the right margin of pages to identify the revised paragraphs and sections. The table of contents has been revised to reflect new pagination of interfiled chapters. In addition, a typographical error that appeared in SER Supplement No. 2, Chapter 2, pages 42 and 47, has been corrected. No other changes have been made to the text of the original PFS SER. For example, the Executive Summary and Introduction are unchanged from the SER as published in September 2000.

It should also be noted that in December 2001, NUREG-1741, "Final Environmental Impact Statement for the Construction and Operation of an Independent Spent Fuel Storage Installation on the Reservation of the Skull Valley Band of Goshute Indians and the Related Transportation Facility in Tooele County, Utah," was issued by the NRC and the cooperating Federal agencies (the Department of the Interior's Bureau of Indian Affairs and Bureau of Land Management and the U.S. Surface Transportation Board).

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ACRONYMS

AAS	Alternate Alarm Station
ACM	Advanced Cruise Missile
ACRAM	Aircraft Crash Risk Analysis Methodology
AFI	Air Force Instruction
ALARA	As Low As Reasonably Achievable
ALCM	Air Launched Cruise Missile
ANS	American Nuclear Society
ANSI	American National Standards Institute
ASLB	Atomic Safety and Licensing Board
ASCE	American Society of Civil Engineers
ASME	American Society of Mechanical Engineers
BDU	Bomb Dummy Unit
BWR	Boiling Water Reactor
CALCM	Conventional Air Launched Cruise Missile
CPIA	Chemical Propulsion Information Agency
CPT	Cone Penetrometer Test
DOD	U. S. Department of Defense
DOE	U. S. Department of Energy
DOT	U. S. Department of Transportation
DSHA	Deterministic Seismic Hazard Analysis
EIS	Environmental Impact Statement
FAA	Federal Aviation Administration
FEMA	Federal Emergency Management Agency
FSAR	Final Safety Analysis Report
FTS	Flight Termination System
FY	Fiscal Year
HMR	Hydrometeorological Report
IM&TE	Instruments, Measuring, and Test Equipment
ISFSI	Independent Spent Fuel Storage Installation
LCO	Limiting Conditions for Operation
LES	Louisiana Energy Services
LLEA	Local Law Enforcement Agency
M	Magnitude
MOA	Military Operating Area
MPC	Multi-Purpose Canister
MTU	Metric Tons of Uranium
NRC	U. S. Nuclear Regulatory Commission
NFPA	National Fire Protection Association
O&M	Operating and Maintenance
PAS	Primary Alarm Station
PFS	Private Fuel Storage, Limited Liability Company
PFS Facility	Private Fuel Storage Facility
PMF	Probable Maximum Flood
PMP	Probable Maximum Precipitation
PSHA	Probabilistic Seismic Hazard Analysis
PWR	Pressurized Water Reactor
QA	Quality Assurance
RAI	Requests for Additional Information
RCRA	Resource Conservation Recovery Act

ACRONYMS (continued)

SAR	Safety Analysis Report
SER	Safety Evaluation Report
SFPE	Society of Fire Protection Engineers
SLCIA	Salt Lake City International Airport
SPT	Standard Penetration Test
SR	Surveillance Requirement
TLD	Thermoluminescent Dosimeters
TMI-2	Three Mile Island Unit-2
TNT	Trinitroluene
UAV	Uninhabited Aerial Vehicle
UBC	Uniform Building Code
UPS	Uninterruptible Power Supply
USDA	U.S. Department of Agriculture
UTTR	Utah Test and Training Range
WSEP	Weapons System Evaluation Program

EXECUTIVE SUMMARY

On June 20, 1997, Private Fuel Storage, L.L.C. (PFS), submitted an application to the U.S. Nuclear Regulatory Commission (NRC) for a license to operate a temporary storage facility for spent nuclear fuel on the Reservation of the Skull Valley Band of Goshute Indians (the Reservation). The Skull Valley Band is formally recognized as an Indian Tribe by the Federal Government. The application consists of several different documents:

1. A **License Application**, in which the applicant describes itself and provides some general and financial information;

2. A **Safety Analysis Report**, in which the applicant describes its plans for building, operating, maintaining, and funding the cleanup and decommissioning of the proposed Facility;

3. An **Emergency Plan**, in which the applicant describes its plan for resolving any emergencies that happen during the Facility's operation;

4. A **Safeguards and Physical Security Plan** (this document is not released to the public), in which the applicant describes its plans for ensuring that the Facility and nuclear material are appropriately protected; and

5. An **Environmental Report**, in which the applicant provides the information that the NRC staff uses in developing its Environmental Impact Statement (EIS) on the proposed Facility. (A draft EIS was published in June 2000, and a Final EIS is expected to be published in early 2001.)

The NRC staff documents its review and conclusions on the safety-related aspects of an application in a Safety Evaluation Report (SER). This SER documents the NRC staff's review and conclusions concerning the first four documents of the PFS license application. Although this Executive Summary provides the reader with some brief overview and summary of the SER, for a full discussion of the NRC staff's safety evaluation and conclusions about PFS's compliance with the applicable regulatory requirements of 10 CFR Part 72, please consult the SER.

The facility that PFS proposes to build (called the PFS Facility) would store spent fuel, that was used to generate power at commercial nuclear power plants in the United States, in large metal and concrete containers that are called storage casks. This method of storing spent fuel is called dry cask storage technology. This is to differentiate it from wet storage, which is a method of storing the spent fuel in a large pool of water.

PFS proposes to locate the PFS Facility on the Reservation of the Skull Valley Band of Goshute Indians. The Reservation is 27 miles west-southwest of Tooele City, Utah. The site for this Facility will cover 820 acres of the Reservation's 18,000 acres. The spent fuel storage casks will be stored on about 100 of these 820 acres. As a Federally-recognized Indian Tribe, the Skull Valley Band is recognized as a sovereign, sub-national political entity, and its Reservation is not considered to be part of the State of Utah. For purposes of geographic orientation, the Reservation is surrounded by Tooele County, Utah.

PFS has requested an initial 20-year license. Before the end of this first 20 years, PFS may submit an application to renew the license. In accordance with NRC's licensing requirements and with PFS's lease arrangements with the Skull Valley Band, all spent fuel would be transferred offsite and the Facility would be ready for decommissioning (that is, returning it in a clean and safe condition to the Skull Valley Band for any use that they choose) by the end of a second term.

While transportation of the spent fuel from the nuclear power plants to the proposed PFS Facility is not considered in this license application, it is obviously a topic of interest. Interstate Highway 80 and the Union Pacific Railroad main line are approximately 24 miles north of the proposed site. Shipping casks that have been approved by NRC will be used to transport the spent fuel to the Facility. Currently, the closest rail service goes only to an area north of the Skull Valley Indian Reservation. One of two approaches could be used to take the shipping casks to the proposed Facility. PFS proposes that the shipping casks will either be off-loaded at a new transfer facility to be built near Timpie, Utah, where they would be loaded onto heavy haul tractor trailers for transport to the PFS Facility, or PFS will build a new railroad line connecting the PFS Facility directly to the Union Pacific main line. The PFS Facility will be accessed by a new road from the Skull Valley Road as shown in Figure 1.1-1 of the Safety Analysis Report.

Description of the Storage Cask

The dry cask storage system that PFS proposes to use at the PFS Facility is Holtec International's HI-STORM 100 Cask System (the cask system). The cask system is a canister-based storage system that stores spent fuel in a vertical orientation (the cask and the fuel rods inside of them are, in effect, standing up). The HI-STORM 100 Cask System consists of three parts:

1. the multi-purpose canister (MPC),
2. the HI-TRAC transfer cask, and
3. the HI-STORM 100 storage overpack.

The MPC is called the confinement system for the spent fuel. It is the metal canister in which the fuel is sealed. The HI-TRAC transfer cask provides radiation shielding and structural protection of the MPC during transfer operations. When the spent fuel arrives at the PFS Facility, this MPC will be in an NRC-certified transportation cask. The HI-TRAC transfer cask will be used to move the MPC from the shipping cask into the HI-STORM storage overpack. The storage overpack provides radiation shielding and structural protection of the MPC during storage. The HI-STORM system can be used to store either pressurized water reactor (PWR) fuel assemblies or boiling water reactor (BWR) fuel assemblies. The HI-STORM 100 Cask System does not rely on any active cooling systems to remove spent fuel decay heat.

The HI-STORM 100 Cask System has been approved by the NRC for use under the general license provisions of 10 CFR Part 72, Subpart K. The HI-STORM 100 Cask System is approved under Certificate of Compliance No. 1014, effective date May 31, 2000, Docket No. 72-1014. The NRC staff evaluated the cask system for general use for dry storage. This evaluation is documented in the NRC's "Holtec International HI-STORM 100 Cask System Safety Evaluation Report", which was issued with the certificate of compliance (the regulatory

document by which NRC allows general use of any approved storage or transportation cask). To demonstrate that the HI-STORM 100 Cask System was acceptable for use at the PFS Facility, PFS evaluated the HI-STORM system against the parameters and conditions specific to the Facility. The NRC staff reviewed the PFS evaluation and, as discussed in this SER, the staff finds that the HI-STORM 100 Cask System is acceptable for use at the PFS Facility under the site-specific license provisions of 10 CFR Part 72.

SAFETY OF FACILITY

In its evaluation of the application, the NRC staff determined that PFS showed that its proposed Facility and the HI-STORM cask design are structurally sound and will ensure that the spent fuel will remain within the cask and maintain a sound structure during all phases of operation for both normal operating conditions and accidents. PFS included analyses of all natural and man-made phenomena, including an in-depth study of potential seismic activity at the PFS Facility. The applicant used a probabilistic seismic hazard analysis approach, rather than a deterministic method required by 10 CFR Part 72 regulations, to analyze potential seismic activity. However, the staff agreed during its review that an exemption to the requirement to use deterministic methods is acceptable because PFS's probabilistic approach considered a full range of seismic factors. The NRC staff performed confirmatory analyses of the PFS probabilistic approach. The confirmatory analyses gave the staff confidence that the approach was acceptable. The PFS probabilistic approach showed that the Facility will remain safe during any credible seismic activity. After reviewing the applicant's analyses and performing additional confirmatory calculations, the NRC staff concluded that the PFS Facility and HI-STORM design is structurally safe and will meet regulatory requirements.

The NRC staff also determined that PFS has shown that the spent nuclear fuel within the storage casks will remain subcritical (that is, unable to sustain a nuclear chain reaction) during all phases of operation for both normal and credible accident conditions. PFS provided radiation dose estimates for the surrounding public and the workers at the Facility. The HI-STORM storage canister will be welded closed to prevent leakage of radioactive material. The canister is surrounded by a thick wall of concrete and steel to shield the area outside of the cask from direct radiation during storage.

The amount of radiation to which a person is exposed is called a dose. PFS has estimated that members of the public near the proposed Facility would receive doses below NRC's regulatory requirements, which for normal conditions of operation is 25 mrem/yr and for credible accidents is 5 rem/yr. PFS also calculated radiation dose rates within the vicinity of individual casks to demonstrate that workers at the proposed Facility will not receive doses that exceed 5 rem/yr, NRC's annual regulatory limits for workers at nuclear facilities. These radiation dose limits have been established by the NRC to prevent any undue risk and to ensure the safety of all members of the public and workers at a nuclear facility. PFS also described its radiation protection program, which employs an As Low As Reasonably Achievable (ALARA) radiation protection principle. The operating PFS Facility would also monitor radiation doses received by the workers and dose rates within the vicinity of the storage pad to verify that radiation dose limits are not exceeded. The NRC staff reviewed PFS's analyses and performed additional confirmatory calculations and concluded that the PFS Facility and HI-STORM design are radiologically safe and will meet regulatory requirements.

PFS was required to demonstrate that all of the important parts of its proposed Facility would continue to perform their designed functions during normal conditions and during any of the accidents that might reasonably be expected to occur. The NRC staff concluded that, as required by 10 CFR Part 72, PFS has provided acceptable analyses of the design and performance of these "structures, systems, and components important to safety" under credible, off-normal and accident scenarios. Among the "off-normal accidents" analyzed by PFS were a cask drop from a height of less than ten inches (the maximum allowable lift height for the cask), partial blockage of the cask vents, and certain operational events. Applicable accident events analyzed by PFS included cask tipover, cask drop from the maximum lift height, flood, fire and explosion, lightning, earthquake, loss of shielding, adiabatic heatup of the cask, tornadoes and missiles generated by natural phenomena, accidents at nearby sites, building structural failure effects on structures, systems, and components, and an unlikely (or non-mechanistic) failure of the confinement boundary. Hazards from nearby sites that were considered included offsite explosions, aircraft crashes, and other potential hazards from nearby military facilities. Based on its evaluation of these events, the staff concluded that they do not pose a credible hazard to the Facility.

The staff further concluded that PFS's analyses of off-normal and accident events demonstrate that the proposed Facility will be sited, designed, constructed, and operated so that during all credible off-normal and accident events, public health and safety will be adequately protected and the capability to retrieve fuel from the Facility will be preserved.

Other Requirements

To demonstrate its financial qualifications, PFS identified anticipated sources of funds to construct its Facility, indicating that much of the total revenue will be required from its customers as prepayments before they ship fuel to the Facility. Appropriate license conditions have been developed and stated in this SER providing reasonable assurance of the applicant's financial qualifications.

The NRC staff also found PFS's emergency plan and safeguards and physical security plans to be acceptable. The emergency plan appropriately described PFS's program for responding to onsite emergencies. It also described plans for seeking offsite assistance, if needed. The safeguards and physical protection plan were also found to meet NRC requirements.

References

Holtec International. *Final Safety Analysis Report for the HI-STORM 100 Cask System.* Docket No. 72-1014. August 2000.

Nuclear Regulatory Commission. *Certificate of Compliance No. 1014, Amendment No. 0.* Docket No. 72-1014. Effective Date: May 31, 2000.

Nuclear Regulatory Commission. *Holtec International HI-STORM 100 Cask System Safety Evaluation Report.* Docket No. 72-1014. May 2000.

Nuclear Regulatory Commission. *Draft Environmental Impact Statement for the Construction and Operation of an Independent Spent Fuel Storage Installation on the Reservation of the Skull Valley Band of Goshute Indians and the Related Transportation Facility in Tooele County, Utah, NUREG-1714.* Docket No. 72-22. June 2000.

Private Fuel Storage Limited Liability Company. *License Application for the Private Fuel Storage Facility.* Docket Number 72-22. June 20, 1997, as amended May 22 and August 28, 1998; May 19, August 10, August 27, September 8, September 21, and December 16, 1999; and February 2, March 17, April 14, May 8, June 23, June 28, July 18, July 27, August 11, August 31, September 14, and September 25, 2000.

Private Fuel Storage Limited Liability Company. *Safety Analysis Report for the Private Fuel Storage Facility, Revision 18.* Docket Number 72-22. September 25, 2000.

Private Fuel Storage Limited Liability Company. *Emergency Plan for the Private Fuel Storage Facility, Revision 10.* Docket Number 72-22. August 31, 2000.

Private Fuel Storage Limited Liability Company. *Private Fuel Storage, L.L.C. Independent Spent Fuel Storage Installation Security Plan, Revision 2.* Docket Number 72-22. June 8, 1999.

Private Fuel Storage Limited Liability Company. *Private Fuel Storage, L.L.C. Independent Spent Fuel Storage Installation Safeguards Contingency Plan, Revision 1.* Docket Number 72-22. June 8, 1999.

Private Fuel Storage Limited Liability Company. *Private Fuel Storage, L.L.C. Independent Spent Fuel Storage Installation Security Training and Qualification Plan, Revision 1.* Docket Number 72-22. June 8, 1999.

Private Fuel Storage Limited Liability Company. *Environmental Report for the Private Fuel Storage Facility, Revision 12.* Docket Number 72-22. September 25, 2000.

Safety Evaluation Report
Concerning the
Private Fuel Storage Facility

INTRODUCTION

On June 20, 1997, Private Fuel Storage Limited Liability Company (PFS or the applicant) submitted an application for a 10 CFR Part 72 license to receive, possess, store, and transfer power reactor spent fuel, and other radioactive materials associated with spent fuel storage, at an independent spent fuel storage installation (ISFSI). The proposed ISFSI is known as the Private Fuel Storage Facility (PFS Facility or the Facility). The Facility will be located on the Reservation of the Skull Valley Band of Goshute Indians (the Reservation) which is geographically located in Tooele County, Utah. The siting of the Facility on the Reservation has been approved by the tribal government of the Skull Valley Band of Goshute Indians.

In support of its application, PFS submitted the following documents, which contain the information specified in 10 CFR Part 72, Subpart B, License Application, Form, and Contents:

(1) the License Application, which contains:
- the general and financial information required by 10 CFR 72.22;
- the proposed technical specifications required by 10 CFR 72.26;
- the applicant's technical qualifications required by 10 CFR 72.28; and
- the preliminary decommissioning plan required by 10 CFR 72.30.

(2) the Safety Analysis Report (SAR) for the Private Fuel Storage Facility required by 10 CFR 72.24;

(3) the Emergency Plan for the Private Fuel Storage Facility required by 10 CFR 72.32;

(4) the Environmental Report for the Private Fuel Storage Facility required by 10 CFR 72.34; and

(5) the Security Plan for the Private Fuel Storage Facility, which includes the safeguards contingency plan, as required by 10 CFR 72.180 and 72.184.

This safety evaluation report (SER) documents the staff's review of the design, operation, and other safety aspects of the Facility, as described in the above submittals except for the Environmental Report. The Environmental Report is the subject of a separate Environmental Impact Statement (EIS), a draft of which was published in June 2000. A Final EIS is expected to be published in early 2001.

The staff's assessment in this SER is based on whether the Facility meets the requirements of 10 CFR Part 72. In its review, the staff evaluated: (1) the characteristics of the site; (2) the Facility operations and operation systems; (3) the design and design criteria for the Facility and its structures, systems, and components important to safety; (4) the programs that support protection of worker and public health and safety; (5) the impact of potential off-normal and

accident events on structures, systems, and components important to safety; (6) the financial qualifications of the applicant; and (7) the proposed Technical Specifications.

The applicant has identified the HI-STORM 100 Cask System as the dry cask storage system that will be used at the Facility. The HI-STORM 100 Cask System has been reviewed and approved by the U.S. Nuclear Regulatory Commission (NRC) for use under the general license provisions of 10 CFR Part 72, Subpart K. The staff's evaluation and approval of the PFS Facility is based, in part, on the use of the HI-STORM 100 Cask System, as approved, evaluated, and described in Certificate of Compliance No. 1014, Amendment No. 0 (Docket No. 72-1014), the NRC's "Holtec International HI-STORM 100 Cask System Safety Evaluation Report" which was issued with the certificate of compliance, and Holtec International's Final Safety Analysis Report (FSAR) for the HI-STORM 100 Cask System. In evaluating the use of this cask at the Facility, the staff reviewed the HI-STORM 100 FSAR and the related NRC SER to determine whether or not the Facility site parameters are enveloped by the cask design parameters considered in those reports and whether the HI-STORM 100 Cask System is acceptable for use at the PFS Facility site. The staff also verified that the Facility cask storage pads and areas are designed to adequately support the static load of the stored cask and that the radiological limits of 10 CFR 72.104 are met.

The staff has reviewed the proposed Private Fuel Storage Facility as described herein and in the documents specified above. Based on the information provided by the applicant, the proposed Technical Specifications, and the proposed license conditions established in this SER, the staff has reasonable assurance that the Facility meets the requirements of 10 CFR Part 72. Therefore, the staff concludes that the Private Fuel Storage Facility can be safely operated.

References

Holtec International. *Final Safety Analysis Report for the HI-STORM 100 Cask System.* Docket No. 72-1014. August 2000.

Nuclear Regulatory Commission. *Certificate of Compliance No. 1014, Amendment No. 0.* Docket No. 72-1014. Effective Date: May 31, 2000.

Nuclear Regulatory Commission. *Holtec International HI-STORM 100 Cask System Safety Evaluation Report.* Docket No. 72-1014. May 2000.

Nuclear Regulatory Commission. *Draft Environmental Impact Statement for the Construction and Operation of an Independent Spent Fuel Storage Installation on the Reservation of the Skull Valley Band of Goshute Indians and the Related Transportation Facility in Tooele County, Utah, NUREG-1714.* Docket No. 72-22. June 2000.

Private Fuel Storage Limited Liability Company. *License Application for the Private Fuel Storage Facility.* Docket Number 72-22. June 20, 1997, as amended May 22 and August 28, 1998; May 19, August 10, August 27, September 8, September 21, and December 16, 1999; and February 2, March 17, April 14, May 8, June 23, June 28, July 18, July 27, August 11, August 31, September 14, and September 25, 2000.

Private Fuel Storage Limited Liability Company. *Safety Analysis Report for the Private Fuel Storage Facility, Revision 18.* Docket Number 72-22. September 25, 2000.

Private Fuel Storage Limited Liability Company. *Emergency Plan for the Private Fuel Storage Facility, Revision 10.* Docket Number 72-22. August 31, 2000.

Private Fuel Storage Limited Liability Company. *Private Fuel Storage, L.L.C. Independent Spent Fuel Storage Installation Security Plan, Revision 2.* Docket Number 72-22. June 8, 1999.

Private Fuel Storage Limited Liability Company. *Private Fuel Storage, L.L.C. Independent Spent Fuel Storage Installation Safeguards Contingency Plan, Revision 1.* Docket Number 72-22. June 8, 1999.

Private Fuel Storage Limited Liability Company. *Private Fuel Storage, L.L.C. Independent Spent Fuel Storage Installation Security Training and Qualification Plan, Revision 1.* Docket Number 72-22. June 8, 1999.

Private Fuel Storage Limited Liability Company. *Environmental Report for the Private Fuel Storage Facility, Revision 12.* Docket Number 72-22. September 25, 2000.

1 GENERAL DESCRIPTION

1.1 Conduct of Review

Chapter 1 of the Private Fuel Storage Facility (PFS Facility or the Facility) Safety Analysis Report (SAR) provides a general, nonproprietary description of the major components and operations of the Facility and of the site. The objective of this Chapter of the SAR is to familiarize the reader with the pertinent features of the installation.

1.1.1 Introduction

The Facility is an independent spent fuel storage installation (ISFSI) that uses dry cask storage technology. In accordance with 10 CFR 72.42, the Facility would be initially licensed for 20 years. Before the end of this license term, the applicant may submit an application to renew the license. If granted, all spent fuel will be transferred offsite and the Facility will be ready for decommissioning by the end of the second term.

The Facility will be located on the Reservation of the Skull Valley Band of Goshute Indians. The Reservation is geographically located in Tooele County, Utah, 27 miles west-southwest of Tooele City, Utah. No large towns are located within 10 miles of the proposed site. The Skull Valley Band of Goshute Indians' Village, which has about 30 residents, is 3.5 miles east-southeast of the site. The site will cover 820 acres of the Reservation's 18,000 acres.

Interstate Highway 80 and the Union Pacific Railroad main line are approximately 24 miles north of the site. Shipping casks approved under 10 CFR Part 71 will be used to transport the spent nuclear fuel to the Facility. The shipping casks will either be off-loaded at an intermodal transfer point near Timpie, Utah, and loaded onto a heavy haul tractor/trailer for transporting to the Facility, or transported via a new railroad line connecting the Facility directly to the Union Pacific main line. The shipping casks and their mode of transport to the Facility are not considered in this safety evaluation report (SER). The Facility will be accessed by a new road from the Skull Valley Road as shown in Figure 1.1-1 of the SAR.

The applicant proposes to begin commercial operation in June 2002.

1.1.2 General Description of the Private Fuel Storage Facility

The Facility is designed to store up to 40,000 metric tons of uranium (MTU) in the form of spent fuel from commercial nuclear power plants in sealed metal canisters. The spent fuel assemblies are placed in sealed canisters, which are then placed inside a steel and concrete storage cask. The ISFSI, consisting of approximately 4,000 storage casks, is passive and does not rely on active cooling systems.

The Facility's restricted area is approximately 99 acres surrounded by a chain link security fence and an outer chain link nuisance fence. An isolation zone and intrusion detection system are located between the two fences. The cask storage area that surrounds the concrete cask storage pads that support the storage casks is surfaced with compacted gravel that slopes slightly to allow for runoff of storm water. Each concrete pad supports up to eight storage casks in a 2 x 4 array. The Canister Transfer Building, where canisters are transferred from the

shipping cask to the storage cask, is located within the restricted area. An overhead bridge crane and a semi-gantry crane are located within the Canister Transfer Building to facilitate shipping cask loading/unloading operations and canister transfer operations.

The staff finds that the site and Facility descriptions have sufficient detail to allow familiarization with the site characteristics of the proposed ISFSI.

1.1.3 General Systems Description

The dry cask storage system that has been identified for use at the Facility is the HI-STORM 100 Cask System (the cask system). The cask system is a canister-based storage system that stores spent fuel in a vertical orientation. It consists of three discrete components: the multi-purpose canister (MPC), the HI-TRAC transfer cask, and the HI-STORM 100 storage overpack. The MPC is the confinement system for the stored fuel. The HI-TRAC transfer cask provides radiation shielding and structural protection of the MPC during transfer operations. The storage overpack provides radiation shielding and structural protection of the MPC during storage. The cask system stores up to 24 pressurized water reactor (PWR) fuel assemblies or 68 boiling water reactor (BWR) fuel assemblies. The HI-STORM 100 Cask System is passive and does not rely on any active cooling systems to remove spent fuel decay heat.

The spent fuel is loaded into the MPCs at the originating nuclear power plant. Before transport, the MPC's lid is welded in place and the canister is drained, vacuum dried, filled with an inert gas, sealed, and leak tested. Shipping casks that are approved under 10 CFR Part 71 (e.g., the HI-STAR 100) are used to transport the MPCs from the originating power plants to the Facility. At the Facility, the shipping cask is lifted off the transport vehicle and placed in a shielded area of the Canister Transfer Building, called a transfer cell. The MPC is transferred from the shipping cask to the transfer cask, then from the transfer cask into the storage cask. The storage cask, loaded with the MPC, is then closed, and moved to the storage area using a cask transporter and placed on a concrete pad in a vertical orientation.

A general description of the cask system and its operation is provided in the SAR. A detailed description of the cask system is given in the Final Safety Analysis Report (FSAR) for the HI-STORM 100 Cask System (Holtec International, 2000), which is referenced in the SAR. The staff finds that the description of the storage cask system to be used at the Facility is sufficiently detailed to allow familiarization with its design and use at the proposed ISFSI.

The HI-STORM 100 Cask System has been approved by the U.S. Nuclear Regulatory Commission (NRC) for use under the general license provisions of 10 CFR Part 72, Subpart K. The HI-STORM 100 Cask System is approved under Certificate of Compliance No. 1014, effective date May 31, 2000, Docket No. 72-1014 (Nuclear Regulatory Commission, 2000a). The staff's evaluation of the cask system for general use is documented in the NRC's "Holtec International HI-STORM 100 Cask System Safety Evaluation Report" (Nuclear Regulatory Commission, 2000b), which was issued with the certificate of compliance. For site-specific use at the Facility, the applicant evaluated the cask system against the parameters and conditions specific to the PFS Facility. Based on the applicant's evaluation, and the staff's evaluation as discussed in this SER, the staff finds that the HI-STORM 100 Cask System is acceptable for use at the Facility under the site-specific license provisions of 10 CFR Part 72.

1.1.4 Identification of Agents and Contractors

Section 1.5 of the SAR identifies the organizations responsible for providing the licensed spent fuel storage and transfer systems and engineering, design, licensing, and operation of the Facility. Holtec International is responsible for the design of the HI-STORM 100 Cask System. Stone & Webster Engineering Corporation is responsible for the design of the Facility. The applicant has overall responsibility for planning and design of the Facility using Stone & Webster Engineering Corporation as a contractor. The applicant is also responsible for the operation of the Facility and for providing quality assurance (QA) services.

The staff finds that Agents and contractors responsible for the design and operation of the installation have been identified.

1.1.5 Material Incorporated by Reference

Each chapter of the SAR includes a reference section that identifies documents referred to in that chapter.

The staff finds that material incorporated by reference, including topical reports and docketed material, has been appropriately identified in the SAR.

1.2 Evaluation Findings

The staff finds that the site and Facility descriptions presented in the SAR have sufficient detail to allow familiarization with the pertinent site-related features of the proposed ISFSI. All Open Items identified in the staff's previous SER for the Facility (Nuclear Regulatory Commission, 2000c) have been resolved. The staff finds that the SAR, in conjunction with the supporting documents referenced in the SAR, describes the ISFSI in sufficient detail to support the findings in 10 CFR 72.40. The staff also finds that the HI-STORM 100 Cask System, as described in the HI-STORM 100 FSAR, is acceptable for use at the Facility under the site-specific license provisions of 10 CFR Part 72.

1.3 References

Holtec International. 2000. *Final Safety Analysis Report for the Holtec International Storage and Transfer Operation Reinforced Module Cask System (HI-STORM 100 Cask System).* Volumes I and II. HI–2002444. Docket No. 72-1014. Marlton, NJ: Holtec International.

Nuclear Regulatory Commission. 2000a. 10 CFR Part 72 *Certificate of Compliance No. 1014, Amendment 0, for the HI-STORM 100 Cask System.* Effective Date May 31, 2000. Docket No. 72-1014.

Nuclear Regulatory Commission. 2000b. *Holtec International HI-STORM 100 Cask System Safety Evaluation Report.* May 2000. Docket No. 72-1014.

Nuclear Regulatory Commission. 2000c. *Safety Evaluation Report of the Site-Related Aspects of the Private Fuel Storage Facility Independent Spent Fuel Storage Installation.* December 15, 1999 (revised and reissued January 4, 2000). Docket No. 72-22.

Private Fuel Storage Limited Liability Company. 2000. *Safety Analysis Report for Private Fuel Storage Facility*. Revision 18. Docket No. 72-22. La Crosse, WI: Private Fuel Storage Limited Liability Company.

2 SITE CHARACTERISTICS

2.1 Conduct of Review

Chapter 2 of the SAR discusses the geographical location of the PFS Facility and meteorological, hydrological, seismological, geological, and volcanological characteristics of the site and the surrounding area. It describes the population distribution within and around the Reservation, land and water uses, and associated site activities. Chapter 2 of the SAR also evaluates site characteristics with regard to safety and identifies assumptions that need to be applied when evaluating safety, establishing installation design, and providing design bases in other evaluations in the SAR.

The staff evaluated site characteristics by reviewing Chapter 2 of the SAR, documents cited in the SAR, and other relevant literature. The staff also considered information and analyses, with respect to geotechnical and seismic considerations, that were submitted by the applicant (Private Fuel Storage, 2001) subsequent to the issuance of the staff's SER on September 29, 2000. The applicant requested an exemption to 10 CFR 72.102(f), which requires a deterministic seismic hazard analysis (DSHA) approach for determining the impact of earthquakes on the Facility. The applicant requested instead to apply a probabilistic seismic hazard analysis (PSHA) approach for analyzing potential seismic events. The staff reviewed this exemption request, as presented in Chapter 2 of the SAR, and conducted an independent evaluation of seismic ground motion hazard at the site based on a survey of existing literature, state of the knowledge in PSHAs and DSHAs, and consideration of existing NRC regulations and regulatory guidance documents (Stamatakos et al., 1999) regarding seismic analyses. As discussed in Section 2.1.6.2, the staff agrees that the use of the PSHA methodology with a 2,000-year return period is acceptable and there is a sufficient basis to grant an exemption to 10 CFR 72.102(f) at the time a license is issued for the Facility. The exemption will only be issued upon completion of the applicable regulatory process described in 10 CFR Part 72. As discussed in Chapters 4 and 5 of this SER, the Facility is designed to withstand a 2,000-year return period ground motion.

The information and analyses in SAR Chapter 2 were reviewed with respect to the applicable siting evaluation regulations in 10 CFR Part 72, Subpart E, and 10 CFR 72.122(b). Where appropriate, findings of regulatory compliance are made for the 10 CFR Part 72 requirements that are fully addressed in Chapter 2 of the SAR. Because compliance with some regulations can only be determined by the integrated review of several sections in Chapter 2 and/or other Chapters within the SAR, a finding of regulatory compliance is not made in each major section unless the specific regulatory requirement is fully addressed. However, findings of technical adequacy and acceptability are made for each section in Chapter 2, as it relates to the regulatory requirements. Upon full consideration of information presented in all Chapters of the SAR and applicable regulatory requirements, the staff concludes that the PFS Facility is in compliance with the requirements of 10 CFR Part 72.

2.1.1 Geography and Demography

This section contains the review of Section 2.1, Geography and Demography, of the SAR. Subsections discussed include (i) site location, (ii) site description, (iii) population distribution

and trends, and (iv) land and water uses. The staff reviewed the discussion on geography and demography with respect to the following regulatory requirements:

- 10 CFR 72.90(a) requires site characteristics that may directly affect the safety or environmental impact of the ISFSI to be investigated and assessed.

- 10 CFR 72.90(b) requires proposed sites for the ISFSI to be examined with respect to the frequency and severity of external natural and man-induced events that could affect the safe operation of the ISFSI.

- 10 CFR 72.90(c) requires design basis external events to be determined for each combination of proposed site and proposed ISFSI design.

- 10 CFR 72.90(d) requires that the proposed sites with design basis external events for which adequate protection cannot be provided through ISFSI design shall be deemed unsuitable for the location of the ISFSI.

- 10 CFR 72.90(e) requires that pursuant to Subpart A of Part 51 of Title 10 for each proposed site for an ISFSI, the potential for radiological and other environmental impacts on the region must be evaluated with due consideration of the characteristics of the population, including its distribution, and of the regional environs, including its historical and aesthetic values.

- 10 CFR 72.90(f) requires the facility to be sited so as to avoid to the extent possible the long-term and short-term adverse impacts associated with the occupancy and modification of floodplains.

- 10 CFR 72.98(a) requires that the regional extent of external phenomena, man-made or natural, that are used as a basis for the design of the ISFSI must be identified.

- 10 CFR 72.98(b) requires that the potential regional impact due to the construction, operation or decommissioning of the ISFSI must be identified. The extent of regional impacts must be determined on the basis of potential measurable effects on the population or the environment from ISFSI activities.

- 10 CFR 72.98(c) requires that those regions identified pursuant to paragraphs 10 CFR 72.98(a) and (b) of this section must be investigated as appropriate with respect to: (1) The present and future character and the distribution of population, (2) Consideration of present and projected future uses of land and water within the region, and (3) Any special characteristics that may influence the potential consequences of a release of radioactive material during the operational lifetime of the ISFSI.

- 10 CFR 72.100(a) requires that the proposed site must be evaluated with respect to the effects on populations in the region resulting from the release of radioactive materials under normal and accident conditions during operation and decommissioning of the ISFSI; in this evaluation both usual and unusual regional and site characteristics shall be taken into account.

- 10 CFR 72.100(b) requires that each site must be evaluated with respect to the effects on the regional environment resulting from construction, operation, and decommissioning for the ISFSI; in this evaluation both usual and unusual regional and site characteristics must be taken into account.

2.1.1.1 Site Location

Section 2.1.1 of the SAR, Site Location, and relevant literature cited in the SAR describes the site location. The Facility will be located within the boundaries of the Reservation of the Skull Valley Band of Goshute Indians. The Reservation is geographically located in Skull Valley, Tooele County, Utah, about 50 miles southwest of Salt Lake City, Utah, and 14 miles north of the entrance to the Dugway Proving Ground in Tooele County, Utah.

The staff reviewed the description of the site location and found it acceptable because it clearly describes the geographic location of the site, including its relationship to political boundaries and natural anthropogenic features. The maps provided in the SAR are acceptable because they provide sufficient detail, which is needed for review of the Facility. This information is acceptable for use in other sections of the SAR to develop the design bases of the Facility, perform additional safety analysis, and demonstrate compliance with the regulatory of requirements in 10 CFR 72.90(a), 72.90(e), and 72.98(a) with respect to this issue.

2.1.1.2 Site Description

Section 2.1.2 of the SAR, Site Description, and relevant literature cited in the SAR describe the site with maps to delineate the site boundary and controlled area. The proposed site is located on a typical valley floor of the local Basin and Range topography. The Stansbury Mountains lie to the east of the site and separate the site from Tooele City, Utah, about 27 miles to the northeast. The Cedar Mountains are approximately 14 miles to the west and separate the Facility from portions of the Utah Test and Training Range within the Great Salt Lake Desert. Skull Valley, Utah, has little population and limited agriculture, although a cattle ranch is located on the north border of the Facility. The site is located within the northern boundary of the Sevier B military operating area, utilized by military aircraft traveling to and from the Utah Test and Training Range and Hill Air Force Base.

Access to the controlled area will be restricted by typical range fencing, and ingress and egress of site personnel will be controlled. Skull Valley Road (Federal Aid Secondary Road 108) is about 1 mile east of the site and connects I-80 to the north with State Route 199 to the south. Traffic on this road is local, either to the Reservation or to the Dugway Proving Ground entrance about 14 miles south of the proposed site (the northern border of the Dugway Proving Ground is about 9 miles from the site). The orientation of the Facility structures with respect to nearby roads, railways, and waterways is shown on various maps and plots, and there is no obvious way in which traffic on adjacent transportation links can interfere with Facility operations.

The site (approximately 99 acres of restricted area for cask storage and a total controlled area of about 820 acres) has about a 15-foot elevation change across the facility with the south side higher than the north side. Local vegetation is sparse due to meager rainfall and extended

drought periods. A shallow dry wash is found to the west of the site, and the primary floodway has been identified to the east of the proposed pad site.

The staff reviewed the site description and relevant literature cited in the SAR. The staff finds that the site description is adequate because the descriptive information and maps clearly delineate the site boundary and controlled area. The maps have a sufficient level of detail and are of appropriate scale and legibility that is required for the review of the site and Facility. The information is also acceptable to determine distances between the Facility and nearby facilities and cities. This information is acceptable for use in other sections of the SAR to develop the design bases of the Facility, perform additional safety analysis, and demonstrate compliance with regulatory requirements in 10 CFR 72.90(a), 72.90(e), and 72.98(a) with respect to this issue.

2.1.1.3 Population Distribution and Trends

Section 2.1.3 of the SAR, Population Distribution and Trends, and relevant literature cited in the SAR describes the population distribution and trends. The population data used in the SAR were derived through local interrogation. Within 5 miles of the proposed site, there are two tribal homes approximately 2 miles southeast of the Facility, additional residences on the Reservation, about 3.5 miles east-southeast of the site, and two private farm residences located approximately 2.75 and 4.0 miles to the northeast of the site. The population within 5 miles of the site is about 36 people. Ten miles east-southeast of the proposed site is the small residential community of Terra with an estimated population of 120. The town of Dugway, with an estimated population of 1,700, is located about 12 miles south of the Facility. The permanent population of the immediate area during the period of operation of the Facility is not expected to grow. As described by the applicant, it is expected that the construction, operation, and decommissioning of the Facility will have a negligible effect on the overall population of the region. No transient or institutional populations are present within 5 miles of the Facility, and no public facilities are anticipated to be located in the vicinity. Based on this information, the applicant concludes that it is likely that the effect of the Facility on the population distribution and growth trends in Skull Valley, Utah, will be small, if any.

The staff reviewed the information presented in the SAR and has determined that the population distribution and trends in the region have been adequately described and assessed. The source of the population data used in the SAR is appropriate and the basis for population projections is reasonable. The staff found that 10 CFR 72.98(c)(1) is met because the region has been appropriately investigated with respect to the present and future character and distribution of the population. This information is also acceptable for use in other sections of the SAR to develop the design bases of the Facility, perform additional safety analysis, and demonstrate compliance with regulatory requirements in 10 CFR 72.90(e), 72.98(a), 72.98(b), 72.100(a), and 72.100(b) with respect to this issue.

2.1.1.4 Land and Water Uses

Section 2.1.4 of the SAR , Uses of Nearby Lands and Waters, and relevant literature cited in the SAR (Bureau of Land Management, 1985, 1986, 1988, 1992) describe land water uses. Land use within the Reservation boundary includes residential use by tribal members. Until 1999, it also included the leased operation (since 1975) of the Tekoi Rocket Engine Test

Facility on the south side of Hickman Knolls by Allied Techsystems. There is no current lessee of the Tekoi facility.[1] Within the 5-mile radius of the proposed site there are approximately 28,000 acres of property owned by the Bureau of Land Management, 13,000 acres of Indian-owned land, and 9,000 acres of privately owned land. The principal land use is for grazing of livestock. The grazing quality is considered to be fair to poor, and in decline. Sheep and cattle are grazed seasonally on much of the acreage within the 5-mile radius and are sequentially pastured within the total acreage. Fifty-five percent of the Bureau of Land Management property within the 5-mile radius of the Facility is within the Pony Express Resource Area and is open to off-highway vehicle use, dispersed camping, and hunting. There are no designated camping areas, off-highway vehicle trails, or roads other than the Skull Valley Road within a 5-mile radius of the proposed site.

Domestic water wells in Skull Valley, Utah, are almost exclusively in unconsolidated alluvial fan deposits along the east side of the valley. Some stock wells in the central part of Skull Valley, Utah, operate under artesian conditions (Arabasz et al., 1987). Water quality varies from good along the east side of the valley to poor in the central part due to the high total dissolved solids content. It is anticipated that water wells will be drilled within the Facility's controlled area to accommodate water needs during construction and operation of the Facility. The applicant will locate and develop the water wells in a manner that prevents any impact (e.g., groundwater drawdown) on adjacent wells (the nearest of which is 1.5 miles from the Facility). Estimated water pumpage from all sources in Skull Valley, Utah is about 5,000 acre-feet of water per year. The applicant estimates water needs at no more than 10,000 gallons per day during construction and on average 1,800 gallons per day during operation. Assuming a conservative 365 days in a year, the water usage is estimated to be 11.2 acre-feet per year during construction activities and 2.0 acre-feet per year for operational activities. Therefore, the projected amount of water used during construction activities is about 0.2 percent (11.2 acre-feet divided by 5,000 acre-feet used per year) of current total water production estimated in Skull Valley and 0.04 percent (2.0 acre-feet divided by 5,000 acre-feet used per year) for operations. These water-use amounts attributed to the Facility are very small when compared with the total ground water budget and should have no perceptible impact on current water use.

The applicant indicates that there will be little projected population growth near the Facility because it is unlikely that the permanent population within 5 miles of the proposed Facility would change significantly during the proposed license period and due to the remoteness and extreme low population density of the area (36 persons within a 5-mile radius), no facilities such as hospitals, prisons, and recreational areas are located and planned within the 5-mile study area. Based on preliminary testing of the onsite monitoring well, the applicant determined that operation of the Facility water well will have no measurable offsite effects on existing groundwater quality or levels (Stone & Webster Engineering Corporation, 1999a). Thus, future impacts on water use are also considered to be minimal.

The staff reviewed the description of the land and water use in the SAR and information (Bureau of Land Management, 1985, 1986, 1988, 1992) cited in the SAR for the region and found that it has been adequately described and assessed. The staff accepts the use of land

[1]The Tekoi Rocket Engine Test Facility was in operation when the NRC initiated its review of the PFS Facility application; therefore, the potential impact of operations in the Tekoi facility is considered in this SER.

and water information provided by the Bureau of Land Management. The region has been investigated as appropriate with respect to consideration of present and projected future uses of land and water within the region. This information is acceptable for use in other sections of the SAR to develop the design bases of the Facility, perform additional safety analysis, and demonstrate compliance with regulatory requirements in 10 CFR 72.98(a), 72.98(b), and 72.98(c) with respect to this issue.

2.1.2 Nearby Industrial, Transportation, and Military Facilities

Section 2.2 of the SAR, Nearby Industrial, Transportation, and Military Facilities, and relevant literature cited in the SAR (Donnell, 1999a,b,c) describes nearby industrial, transportation, and military facilities and identifies potential hazards from these facilities. This information is necessary to evaluate credible scenarios involving manmade facilities that may endanger the PFS Facility site. The staff reviewed nearby industrial, transportation, and military facilities with respect to the following regulatory requirements:

- 10 CFR 72.94(a) requires that the region must be examined for both past and present man-made facilities and activities that might endanger the proposed ISFSI. The important potential man-induced events that affect the ISFSI design must be identified.

- 10 CFR 72.94 (b) requires that information concerning the potential occurrence and severity of such events must be collected and evaluated for reliability, accuracy, and completeness.

- 10 CFR 72.94 (c) requires that appropriate methods must be adopted for evaluating the design basis external man-induced events, based on the current state of knowledge about such events.

- 10 CFR 72.98(a) requires that the regional extent of external phenomena, man-made or natural, that are used as a basis for the design of the ISFSI must be identified.

- 10 CFR 72.98(b) requires that the potential regional impact due to the construction, operation or decommissioning of the ISFSI must be identified. The extent of regional impacts must be determined on the basis of potential measurable effects on the population or the environment from ISFSI activities.

- 10 CFR 72.98(c) requires that those regions identified pursuant to paragraphs 10 CFR 72.98 (a) and (b) must be investigated as appropriate with respect to: (1) The present and future character and the distribution of population, (2) Consideration of present and projected future uses of land and water within the region, and (3) Any special characteristics that may influence the potential consequences of a release of radioactive material during the operational lifetime of the ISFSI.

- 10 CFR 72.100(a) requires that the proposed site must be evaluated with respect to the effects on populations in the region resulting from the release of

radioactive materials under normal and accident conditions during operation and decommissioning of the ISFSI; in this evaluation both usual and unusual regional and site characteristics shall be taken into account.

- 10 CFR 72.100(b) requires that each site must be evaluated with respect to the effects on the regional environment resulting from construction, operation, and decommissioning of the ISFSI; in this evaluation both usual and unusual regional and site characteristics must be taken into account.

Summary of Review

The identification of potential hazards includes identification of facilities and determination of credible scenarios that may endanger the PFS Facility. The facilities identified by the applicant include the Dugway Proving Ground, the Tekoi Rocket Engine Test Facility, the Utah Test and Training Range, the Tooele North Army Depot Area, and the Tooele South Army Depot Area.

The Dugway Proving Ground is a federal site that performs activities that include testing and disposing of chemical and biological agents. The site also contains the Michael Army Airfield, which is used by military aircraft and potentially for vehicle landings of the X-33 suborbital demonstrator. Its entrance is about 14 miles east-southeast of the PFS Facility. The Cedar Mountains (elevation greater than 5,300 feet) lie between this site and the PFS Facility. The Tekoi Rocket Engine Test Facility is a commercial facility that has performed high explosive and rocket motor testing. This facility is about 2.3 miles south-southeast of the PFS Facility on the south side of Hickmam Knolls (a rock formation with elevation greater than 4,600 feet). The Utah Test and Training Range is a federal site with activities that test air-to-ground and air-to-air munitions. The Tooele North Army Depot Area is a federal site located 17 miles east-northeast of the PFS Facility. The Tooele South Army Depot Area is also a federal site located about 22 miles east-southeast of the PFS Facility. It performs incineration of retired nerve agents. The Stansbury Mountains (elevation greater than 8,000 feet) lie between the Tooele site and the PFS Facility.

The applicant identified the potential crash of civilian or military aircraft onto the site as a potential hazard. Civilian aircraft with a potential to crash at the PFS Facility site include aircraft taking off and landing at Salt Lake City International Airport, aircraft flying along jet routes J-56 and V-257, and general aviation aircraft flying in the vicinity of the proposed site. Military aircraft that have a potential to crash at the PFS Facility site include aircraft taking off and landing at Michael Army Airfield at the Dugway Proving Ground, aircraft flying military route IR-420, aircraft flying to and from the Utah Test and Training Range and Hill Air Force Base, helicopters flying near the site, and the X-33 suborbital demonstrator vehicle landing at Michael Army Air Field.

The staff finds that all nearby military and industrial facilities that may present a hazard to the PFS Facility have been adequately identified. The potential hazards from these facilities, including from military aircraft flying through Skull Valley and helicopter flights over the Utah Test and Training Range and Skull Valley, are assessed in Chapter 15 of this SER.

2.1.3 Meteorology

The staff has reviewed the information presented in Section 2.3 of the SAR, Meteorology. Subsections discussed below include (i) regional climatology, (ii) local meteorology, and (iii) onsite meteorological measurement program. The staff reviewed the discussion on meteorology with respect to the following regulatory requirements:

- 10 CFR 72.90(a) requires site characteristics that may directly affect the safety or environmental impact of the ISFSI be investigated and assessed.
- 10 CFR 72.90(b) requires proposed sites for the ISFSI to be examined with respect to the frequency and severity of external natural and man-induced events that could affect the safe operation of the ISFSI.
- 10 CFR 72.90(c) requires design basis external events to be determined for each combination of proposed site and proposed ISFSI design.
- 10 CFR 72.90(d) requires the proposed sites with design basis external events for which adequate protection cannot be provided through ISFSI design shall be deemed unsuitable for the location of the ISFSI.
- 10 CFR 72.90(e) requires that, pursuant to Subpart A of Part 51 of Title 10, for each proposed site for an ISFSI, the potential for radiological and other environmental impacts on the region must be evaluated with due consideration of the characteristics of the population, including its distribution, and of the regional environs, including its historical and aesthetic values.
- 10 CFR 72.90(f) requires the facility to be sited so as to avoid to the extent possible the long-term and short-term adverse impacts associated with the occupancy and modification of floodplains.
- 10 CFR 72.92(a) requires that natural phenomena that may exist or that can occur in the region of a proposed site be identified and assessed according to their potential effects on the safe operation of the ISFSI. The important natural phenomena that affect the ISFSI design must be identified.
- 10 CFR 72.92(b) requires that records of the occurrence and severity of those important natural phenomena must be collected for the region and evaluated for reliability, accuracy, and completeness. The applicant shall retain these records until the license is issued.
- 10 CFR 72.92(c) requires that appropriate methods must be adopted for evaluating the design basis external natural events based on the characteristics of the region and the current state of knowledge about such events.
- 10 CFR 72.98(a) requires that the regional extent of external phenomena, man-made or natural, that are used as a basis for the design of the ISFSI must be identified.

- 10 CFR 72.98(c) requires that those regions identified pursuant to paragraphs 10 CFR 72.98(a) and (b) be investigated as appropriate with respect to: (1) The present and future character and the distribution of population, (2) Consideration of present and projected future uses of land and water within the region, and (3) Any special characteristics that may influence the potential consequences of a release of radioactive material during the operational lifetime of the ISFSI.

- 10 CFR 72.122(b) requires (1) structures, systems, and components important to safety must be designed to accommodate the effects of, and to be compatible with, site characteristics and environmental conditions associated with normal operation, maintenance, and testing of the ISFSI and to withstand postulated accidents. (2) structures, systems, and components important to safety must be designed to withstand the effects of natural phenomena such as earthquakes, tornadoes, lightning, hurricanes, floods, tsunami, and seiches, without impairing their capability to perform safety functions. The design bases for these structures, systems, and components must reflect: (i) Appropriate consideration of the most severe of the natural phenomena reported for the site and surrounding area, with appropriate margins to take into account the limitations of the data and the period of time in which the data have accumulated, and (ii) Appropriate combinations of the effects of normal and accident conditions and the effects of natural phenomena. The ISFSI should also be designed to prevent massive collapse of building structures or the dropping of heavy objects as a result of building structural failure on the spent fuel or high-level radioactive waste or on to structures, systems, and components important to safety. (3) Capability must be provided for determining the intensity of natural phenomena that may occur for comparison with design bases of structures, systems, and components important to safety. (4) If the ISFSI is located over an aquifer which is a major water resource, measures must be taken to preclude the transport of radioactive materials to the environment through this potential pathway.

2.1.3.1 Regional Climatology

Section 2.3.1 of the SAR, Regional Climatology, and relevant literature cited in the SAR (National Oceanic and Atmospheric Administration, 1960, 1992; Ashcroft et al., 1992) describe the regional climatology associated with the Facility site. The applicant used climatologic data collected at the Salt Lake City International Airport (approximately 50 miles north of the proposed site) and at the Dugway Proving Ground (within 14 miles of the proposed site) to characterize the climate in Skull Valley, Utah. Regional data have been augmented with data collected at a meteorologic station established in Skull Valley, Utah, especially for the purpose of verifying the regional climatic and meteorologic data. Long-term weather data and severe weather data from the National Weather Service (National Oceanic and Atmospheric Administration, 1975–1995; Ramsdell and Andrews, 1986; Grazulis, 1993) are discussed. The information presented includes (i) weather influence of terrain; (ii) regional temperature, precipitation, atmospheric moisture, and winds; (iii) severe weather including maximum and minimum temperatures, temperature ranges, freeze-thaw cycle, degree days, design temperature, subsoil temperatures, extreme winds, tornadoes, dust devils, hurricanes, and tropical storms, precipitation extremes, thunderstorms and lightning, snow storms and snow

accumulation, hail and ice storms, and other phenomena; (iv) station pressure; and (v) air density (National Oceanic and Atmospheric Administration, 1975–1995).

The staff reviewed the regional climate data and discussions presented in the SAR, and found it acceptable. It is acceptable because reliable data sources, such as the National Weather Service, were used. In addition, all relevant data including weather data from nearby regional and local meteorological stations, were appropriately summarized to define the expected climatology of the site region. The information on severe weather data (National Oceanic and Atmospheric Administration, 1975–1995; Ramsdell and Andrews, 1986; Grazulis, 1993) is an acceptable source of data for the development of structural design criteria in Chapter 3, Principal Design Criteria, of the SAR regarding strong wind and windborne missiles.

The staff has determined that this information is acceptable for use in other sections of the SAR to develop the design bases of the Facility, perform additional safety analysis, and demonstrate compliance with the regulatory requirements of 10 CFR 72.90(a), 72.90(b), and 72.122(b) with respect to this issue.

2.1.3.2 Local Meteorology

Section 2.3.2 of the SAR, Local Meteorology, and relevant literature cited in the SAR (Ashcroft et al., 1992) describes local meteorology of the site. The SAR provides the maximum temperature data for Salt Lake City, Utah, (about 50 miles northeast to the site at an elevation of approximately 4,220 feet above mean sea level), as a part of the regional climatology information, based on the long-term meteorological data collected by the National Weather Service at the Salt Lake City International Airport (National Oceanic and Atmospheric Administration, 1992). Because the Stansbury and Oquirrh Mountains, with elevations exceeding 10,000 feet above mean sea level, are located between Salt Lake City, Utah, and the site, meteorological data collected in Skull Valley, Utah, are also needed to characterize the local conditions. The applicant provided the temperature data recorded at Dugway (approximately 12 miles south of the site at an elevation of 4,340 feet above mean sea level) and Iosepa South Ranch (about 12 miles north of the site at an elevation of 4,415 feet above mean sea level). These data are based on Ashcroft et al. (1992). The recorded period at Dugway was from 1950 to 1992 and the recorded period at Iosepa South Ranch was from 1951 to 1958. The applicant also provided annual average and average daily maximum temperatures for the month of July recorded at the proposed site meteorological tower during 1997 and 1998. Temperatures recorded at different sites are summarized in Table 2-1 of this SER.

Table 2.3-4 of the SAR provides the recorded temperatures at the site and at nearby locations. The annual average temperature measured at Salt Lake City is about 52 °F and at Skull Valley is 49 °F. Maximum and minimum average monthly temperatures recorded at Skull Valley are about 75 ° and 23 °F, respectively. Average daily maximum temperature recorded at the site is 90 °F, and the average daily minimum is 10 °F. As shown in Table 2.3-4 of the SAR and summarized in Table 2-1 of this SER, temperatures recorded at the site through the onsite measurement program during the period December 1996 and December 1998 correlate well with temperatures recorded at nearby sites for significantly longer periods.

Maximum solar radiation recorded during the onsite meteorological monitoring program at the site is 685 W/m^2 for a 12-hour period. Maximum solar insolation recorded at Salt Lake City during a 12-hour period in the 30-year Solar and Meteorological Surface Observational Network is 730 W/m^2. Relative humidity at the site varied from 3.9 to 98.6 percent in 1997. The average humidity in 1997 was 58.7 percent. The wind at the proposed site is similar to that at Salt Lake City. The prevailing wind direction is southeast or south-southeast throughout the year (Private Fuel Storage Limited Liability Company, 2000). Highest and lightest monthly average speeds recorded at the site are 9.6 and 7.4 mph. The average wind speed recorded for the two-year periods at the site is 8.7 mph.

The applicant did not develop atmospheric diffusion estimates for the facility based on the measurement program. However, the applicant assumed design basis atmospheric diffusion characteristics based on Regulatory Guide 1.145 (Nuclear Regulatory Commission, 1983). These design basis characteristics are a wind speed of 1 m/sec, atmospheric stability class F, and no consideration of plume meander.

The staff reviewed the local meteorological data and discussions presented in the SAR and found them acceptable because reliable data sources such as the National Weather Service were used, and the data from December 1996 to March 1998 are appropriately summarized.

Table 2-1. Temperature at the Private Fuel Storage Facility site and nearby cities

Temperature	Salt Lake City	Dugway	Iosepa South Ranch	Site Meteorological Tower
Average Daily Maximum (°F)				
June	82.3†/83*	84.7†/85*	86*	78*
July	92.2†/93*	94.4†/94*	95*	90*
August	89.4†/90*	91.3†/91*	93*	90*
Average Daily Minimum (°F)				
June	55.4†/53*	53.2†/53*	45*	46*
July	63.7†/62*	61.9†/62*	52*	54*
August	61.8†/60*	59.3†/59*	53*	57*
Annual Average (°F)	52*	51*	50*	49*
Monthly Average (°F)				
June	69.1†/68*	69.0†/69*	66*	63*
July	77.9†/78*	78.2†/78*	74*	74*
August	75.6†/75*	75.3†/75*	73*	75*
Record High (°F)				
June	104†	107†	NA	93.4*
July	107†	109†	NA	99.3*
August	104†	108†	NA	96.6*
Record Low (°F)				
December	-15†	-27†	NA	-4.7*
January	-22†	-25†	NA	-7.0*
February	-14†	-29†	NA	4.7*

† Ashcroft et al., 1992
* Private Fuel Storage Limited Liability Company, 2000
NA Not Available

The staff reviewed the topographic maps to determine the affects of meteorology on erosion at the site. The maps indicate that there is approximately 15 feet of relief across the proposed site and the site slopes from south to north. Staff analysis of the slope and the expected meteorologic environment indicates that the slopes will be stable and the site will not experience significant erosion. The staff determined that the current information presented in the SAR is acceptable for use in other sections of the SAR to develop the design bases of the Facility, perform additional safety analysis, and demonstrate compliance with the regulatory requirements of 10 CFR 72.92(a), 72.98(a), 72.98(c)(3), and 72.122(b).

2.1.3.3 Onsite Meteorological Measurement Program

Section 2.3.3 of the SAR, Onsite Meteorological Measurement Program, describes the onsite meteorological measurement program. The applicant described the meteorologic instrumentation that was used, including detail on its emplacement and operation. Actual siting, types of sensors, recordings of sensor output, instrument surveillance plans, and data acquisition and reduction methods are included in the SAR. Examples of data collected from the instruments are provided. As discussed in Section 2.1.3.2 of this SER, the applicant did not develop atmospheric diffusion estimates for the facility based on the measurement program.

The staff reviewed the information on the onsite meteorological measurement program and found it acceptable because:

- The onsite meteorologic measurement program has been adequately described such that potential meteorological effects on the Facility can be identified and assessed.
- The onsite meteorologic measurement program has been adequately described such that the regional extent of external phenomena, manmade or natural, used as a basis for the design of the Facility, can be identified.
- The onsite meteorologic measurement program has been adequately described such that the regional impact on the population or the environment due to the construction, operation, or decommissioning of the Facility can be identified.

The staff has determined that the current information is acceptable for use in other sections of the SAR to develop the design bases of the Facility, perform additional safety analysis, and demonstrate compliance with the regulatory requirements of 10 CFR 72.92(a), 72.98(a), 72.98(c)(3), and 72.122(b) with respect to this issue.

2.1.4 Surface Hydrology

The staff has reviewed the information presented in Section 2.4 of the SAR, Surface Hydrology. Subsections discussed include (i) hydrologic description, (ii) floods, (iii) probable maximum flood on streams and rivers, (iv) potential dam failures, (v) probable maximum surge and seiche flooding, (vi) probable maximum tsunami flooding, (vii) ice flooding, (viii) flood protection requirements, and (ix) environmental acceptance of effluents. The staff reviewed the discussion on surface hydrology with respect to the following regulatory requirements:

- 10 CFR 72.90(a) requires site characteristics that may directly affect the safety or environmental impact of the ISFSI be investigated and assessed.
- 10 CFR 72.90(b) requires proposed sites for the ISFSI to be examined with respect to the frequency and severity of external natural and man-induced events that could affect the safe operation of the ISFSI.
- 10 CFR 72.90(c) requires design basis external events to be determined for each combination of proposed site and proposed ISFSI design.

- 10 CFR 72.90(d) requires the proposed sites with design basis external events for which adequate protection cannot be provided through ISFSI design be deemed unsuitable for the location of the ISFSI.

- 10 CFR 72.90(e) requires that, pursuant to Subpart A of Part 51 of Title 10, for each proposed site for an ISFSI, the potential for radiological and other environmental impacts on the region must be evaluated with due consideration of the characteristics of the population, including its distribution, and of the regional environs, including its historical and aesthetic values.

- 10 CFR 72.90(f) requires the facility to be sited so as to avoid to the extent possible the long-term and short-term adverse impacts associated with the occupancy and modification of floodplains.

- 10 CFR 72.92(a) requires that natural phenomena that may exist or that can occur in the region of a proposed site must be identified and assessed according to their potential effects on the safe operation of the ISFSI. The important natural phenomena that affect the ISFSI design must be identified.

- 10 CFR 72.92(b) requires that records of the occurrence and severity of those important natural phenomena must be collected for the region and evaluated for reliability, accuracy, and completeness. The applicant shall retain these records until the license is issued.

- 10 CFR 72.92(c) requires that appropriate methods be adopted for evaluating the design basis external natural events based on the characteristics of the region and the current state of knowledge about such events.

- 10 CFR 72.98(a) requires that the regional extent of external phenomena, man-made or natural, that are used as a basis for the design of the ISFSI be identified.

- 10 CFR 72.98(b) requires that the potential regional impact due to the construction, operation or decommissioning of the ISFSI be identified. The extent of regional impacts must be determined on the basis of potential measurable effects on the population or the environment from ISFSI activities.

- 10 CFR 72.98(c) requires that those regions identified pursuant to paragraphs 10 CFR 72.98(a) and (b) must be investigated as appropriate with respect to: (1) The present and future character and the distribution of population, (2) Consideration of present and projected future uses of land and water within the region, and (3) Any special characteristics that may influence the potential consequences of a release of radioactive material during the operational lifetime of the ISFSI.

- 10 CFR 72.122(b) requires (1) structures, systems, and components important to safety must be designed to accommodate the effects of, and to be compatible with, site characteristics and environmental conditions associated with normal operation, maintenance, and testing of the ISFSI and to withstand postulated

accidents. (2) Structures, systems, and components important to safety must be designed to withstand the effects of natural phenomena such as earthquakes, tornadoes, lightning, hurricanes, floods, tsunami, and seiches, without impairing their capability to perform safety functions. The design bases for these structures, systems, and components must reflect: (i) Appropriate consideration of the most severe of the natural phenomena reported for the site and surrounding area, with appropriate margins to take into account the limitations of the data and the period of time in which the data have accumulated, and (ii) Appropriate combinations of the effects of normal and accident conditions and the effects of natural phenomena. The ISFSI should also be designed to prevent massive collapse of building structures or the dropping of heavy objects as a result of building structural failure on the spent fuel or high-level radioactive waste or on to structures, systems, and components important to safety. (3) Capability must be provided for determining the intensity of natural phenomena that may occur for comparison with design bases of structures, systems, and components important to safety. (4) If the ISFSI is located over an aquifer which is a major water resource, measures must be taken to preclude the transport of radioactive materials to the environment through this potential pathway.

2.1.4.1 Hydrologic Description

The Facility will be located in Skull Valley, Utah, between the Stansbury and East Cedar Mountain ranges. The Skull Valley watershed is approximately 50 miles long and 22 miles wide at its widest point, sloping gently northward to the Great Salt Lake. The site is situated near the center of the valley approximately 24 miles south of Interstate Highway 80 and the Great Salt Lake. The watershed through the central valley is an alluvium comprised of poorly sorted coarse to fine grained deposits resulting in a relatively high permeability. A U.S. Department of Agriculture (USDA) soil Curve Number of 70 was determined appropriate for the alluvium. No perennial streams are observed in the Skull Valley watershed. Most runoff infiltrates into the alluvium and recharges the subsurface groundwater. Surface runoff drains through channels formed during wet years; however, a continuous system of drainage channels is not apparent in the area adjacent to the Facility.

The tributary watershed of the Facility location drains approximately 334 sq mi, which includes the west slope of the Stansbury Mountains, west slope of the Onaqui Mountains, north slope of Lookout Mountain, east slope and south tip of the lower Cedar Mountain Range, and the valley lowlands. The watershed was subdivided into Basin A (tributary to the southeasterly side of the Facility) and Basin B (tributary to the southwesterly side of the Facility) as shown in Figure 1 of Donnell [1999f, calculation 0599602–G(B)–17, Revision 1]. A slight ridge line extends from the Facility northerly to Hickman Knolls and then westerly toward the East Cedar Mountain range. The ridge naturally segments the watershed into drainage basins of approximately 270 square miles (Basin A) and 64 square miles (Basin B) for flood analysis purposes.

Structures

The Facility will be situated on approximately 99 acres located near the center of the valley, approximately 26 miles from the southerly end of the watershed. The casks will be placed on storage pads at an elevation 4,475 feet above mean sea level at the southwest corner, falling to

elevation 4,462 feet above mean sea level at the northeast corner. Four predominant structures will be integrated into the Facility that have flood impact potential as shown in Figure 2 of Donnell [1999f, calculation 0599602–G(B)–17, Revision 1]. A berm will be constructed along the upstream side of the Facility to divert potential flood waters around the cask storage area. A railroad embankment will be constructed extending from the west side of the valley and linking into the diversion berm. An access road will link the roadway located on the east side of the valley to the Facility. In conjunction with the access road, a diversion berm (road berm) will be constructed perpendicular to the road (immediately east of the Facility) to span a gap in the natural ridge, thereby isolating the flood waters from Basins A and B. The structures are to be designed as an additional measure to ensure that flood water surface elevations remain below cask storage pad elevations.

The staff reviewed the hydrologic description and found it acceptable because the basic information regarding surface hydrology of the site and the vicinity has been described in sufficient detail for review of the license application. The staff has determined that this information is acceptable for use in other sections of the SAR to develop the design bases of the Facility, perform additional safety analysis, and demonstrate compliance with regulatory requirements in 10 CFR 72.90, 72.92(a), 72.98(a), and 72.98(b) with respect to this issue.

2.1.4.2 Floods

A probable maximum flood (PMF) analysis was performed for the proposed site based on state of the art procedures and practices outlined by the U.S. Army Corps of Engineers (1997). The analysis comprised delineation of the tributary drainage basins, determination of the appropriate rainfall depths, simulation of the storm and routing of the runoff hydrographs, determination of the flood water surface elevations near and through the PFS Facility site, and an evaluation of how the proposed structures affect site safety. Based on this analysis, the site is a flood dry site (i.e., the cask storage pads are elevated out of the adjacent flood plain), although the site will be temporarily isolated during a major flood event.

Little information is available pertaining to historic flooding in the Skull Valley watershed. The lack of a definitive, continuous stream channel or drainage feature throughout the basin indicates that drainage does not occur during frequent, low-intensity precipitation events (i.e., 2-year frequency or less storm events). The presence of segmented drainage channels in areas adjacent to the site indicates that drainage and subsequent channelization occurs, probably derived from less frequent, high-intensity precipitation events (i.e., 10-year storm events).

The staff reviewed the PMF analysis and found it acceptable because the surface water flooding that may directly affect the safety or environmental impact has been sufficiently investigated and assessed. The staff has determined that this information is acceptable for use in other sections of the SAR to develop the design bases of the Facility, perform additional safety analysis, and demonstrate compliance with regulatory requirements in 10 CFR 72.90(c), 72.90(d), 72.90(f), and 72.122(b) with respect to this issue.

2.1.4.3 Probable Maximum Flood on Streams and Rivers

The site is situated in an area that will be isolated (with the assistance of embankments) from the major floods derived from both Basins A and B. The site was evaluated in the SAR for the PMF scenario to ensure that a flood dry condition prevailed. The staff performed an independent flood analysis for both Basins A and B.

Probable Maximum Precipitation

The PMF is derived from the probable maximum precipitation (PMP) (rainfall) that may occur in each drainage basin. Using Hydrometeorological Report No. 49 (HMR 49) (National Weather Service, 1977), PMP values were determined to be 12.2 inches for a 72-hour General Storm and 10.2 inches for a 6-hour Local (thunderstorm) Storm. These precipitation amounts incorporate appropriate regional and elevation reduction factors. The staff reviewed the PMP values by using HMR 49 in accordance with NUREG–1623 (Nuclear Regulatory Commission, 1999), and determined the applicant's PMP values to be acceptable.

Curve Number

The Skull Valley alluvium was determined by the applicant to have a USDA soil Curve Number of 70 as indicated in the SAR. This value is based on the assumption that the soil is in a natural and unsaturated condition, thereby allowing the rainfall and runoff to infiltrate into the soil. However, a PMF analysis stipulates that the antecedent soil condition is saturated preventing infiltration and maximizing potential runoff. Therefore, the SAR elevated the Curve Number to 96 to account for the saturated soil conditions. A Curve Number of 96 was applied for the PMF computation. The staff computed the saturated soil Curve Number to be 85. Therefore, the staff considers the use of the conservative Curve Number of 96 as an acceptable value to describe the alluvium as proposed in the SAR.

Time of Concentration

An important factor in determining the magnitude of the PMF is estimating the time of concentration. The time of concentration is the time (hours) required for runoff to flow overland from the most distant point in the basin to the point of interest (site). The Kirpich (1964) method was applied resulting in times of concentration of 11 hours in Basin A and 4.2 hours in Basin B. The staff performed independent calculations for the times of concentration for both basins—Basin A resulted in a time of concentration of approximately 12 hours and Basin B yielded a time of concentration of approximately 4 hours. In Basin A, the applicant's 11-hour estimate will yield a more conservative peak flood discharge (larger) than the 12-hour value derived by the staff. In Basin B, the difference between 4 hours and 4.2 hours is considered negligible by the staff. Therefore, the staff considers the times of concentration presented in the SAR as acceptable input values for estimating the flood peak discharge.

Flood Peak Discharge Determination

The time of concentration, Curve Number, drainage area, and PMP rainfall depth were input into the HEC-I Flood Hydrograph Package (U.S. Army Corps of Engineers, 1990) to determine the PMF peak discharges, for both the General Storm and Local Storm options, for both Basin

A and Basin B. It was determined that the PMF for Basin A (Local Storm) yielded a peak flood discharge of 85,000 ft^3/sec. The staff independently performed a similar analysis yielding a PMF peak discharge of 85,800 ft^3/sec. Therefore, the staff considers the PMF peak discharge of 85,000 ft^3/sec as an acceptable value. The PMF peak discharge for Basin B (Local Storm) was determined by the applicant to be 102,000 ft^3/sec. The staff independently calculated the peak discharge yielding a peak discharge of approximately 100,000 ft^3/sec. The applicant's PMF is larger than that estimated by the staff and is therefore considered more conservative and acceptable for the flood impact analysis.

Flood Impacts on Site Structures

The PMF peak discharges for Basins A and B were routed to and through the site using HEC-RAS (U.S. Army Corps of Engineers, 1997). HEC-RAS is a numerical hydraulic routing simulation program that translates the PMF hydrograph output from HEC-I into water surface elevations throughout the site as a function of the site contours and structures. The water surface elevations from HEC-RAS are then compared to the proposed elevations of the critical components of the storage area to determine whether the components are flood dry. Flood water surface elevations were identified at each of the four structural components presented in Section 2.1.4.1 and potential impacts to the cask storage pads were determined as discussed below.

The SAR analysis indicates that the PMF (Basin A) peak discharge water surface elevation at the upstream face of the roadway embankment is 4,506.5 feet above mean sea level. The earth berm with top elevation of 4,507.5 feet above mean sea level contains the flood flow. The PMF overtops the low point of the access road embankment (elevation 4,502 feet) by approximately 4.5 feet. The cask storage pads are located downstream from the embankment where the flood water surface elevation is approximately 4.5 feet lower than the pad elevation. The PMF water surface elevation adjacent to the northeast corner of the Facility is approximately 4,456.8 feet above mean sea level. Therefore, the cask storage area is approximately 5 feet above the PMF water surface elevation and will be flood dry.

The staff has independently computed the PMF water surface elevations for a flood discharge of 85,800 ft^3/sec (Basin A) through the east channel area adjacent to the site. The staff computations yielded water surface elevations of 4,444.3 feet downstream of the roadway embankment, 4456.7 feet adjacent to the roadway embankment, and 4,477.4 feet upstream of the roadway embankment above mean sea level. The SAR presented water surface elevations of 4,444.2 feet, 4,456.7 feet, and 4477.4 feet above sea level for the same locations, respectively. Based upon the staff computation, the staff has determined that the cask storage area will remain flood dry. Therefore, the staff found the applicant's analysis for Basin A to be acceptable.

The SAR analysis indicates that the PMF (Basin B) will overtop the railroad embankment (elevation 4,475 feet above mean sea level) by approximately 3.2 feet at an elevation of 4,478.2 feet above mean sea level. The berm constructed immediately upstream of the Facility will have a top elevation of 4,480 feet above mean sea level and extend above the PMF water surface elevation approximately 1.8 feet (freeboard). Flood waters do not impact the south face of the berm. The PMF water surface elevation at the northeast corner of the site is at an elevation of 4,458 feet above mean sea level. The cask storage pad elevation is 4,462 feet

above mean sea level, indicating the pad is approximately 4 feet above the PMF water surface elevation. The staff reviewed the applicant's analysis and agrees that the cask storage pad will remain flood dry during the Basin B PMF event. Therefore the staff found the applicant's analysis for Basin B to be acceptable.

As discussed, the staff reviewed the analysis for PMF on streams and rivers and found it acceptable because the applicant has demonstrated that the potential flood scenarios will not result in flooding of the cask storage pad. The staff has determined that this information is acceptable for use in other sections of the SAR to develop the design bases of the Facility, perform additional safety analysis, and demonstrate compliance with the regulatory requirements of 10 CFR 72.90(c), 72.90(d), 72.90(f), and 72.122(b) with respect to this issue.

2.1.4.4 Potential Dam Failures (Seismically Induced)

There are no water storage, flow control, or embankment structures in the watershed upstream of the site. Therefore, there are no potential impacts on the site from dam or control structure failure. The only potential embankment failures that may occur during a flooding event are the railroad embankment, the Facility berm, the access road embankment, and the road berm. Therefore, each scenario presents a different potential impact on the site as discussed below.

Failure of the railroad embankment will result in the flood waters concentrating through a breach and then resuming their northerly flow in the floodway. The water surface elevation over the embankment will be lowered as the cross sectional area of the flood flow increases. The flood flow water surface elevation will be a minimum of 4 feet below the cask pad top elevation. The railroad embankment failure will isolate the site until the embankment is repaired. Therefore, failure of the railroad embankment has no flood impact on the cask storage area.

Failure of the Facility upstream berm is highly unlikely since flood water will not contact the upstream face of the embankment. The only point of flood water contact with the Facility berm is at the interface of the railroad embankment and the west component of the Facility berm. Should the berm fail, the flood water surface elevation could potentially rise approximately 3 feet. However, there will be a minimum 1-foot differential between the water surface elevation and the cask pad top elevation, thereby keeping the casks dry. Therefore, failure of the Facility berm has no flood impact on the cask storage area.

In the event that the access road embankment fails, a breach will result, thereby funneling the flood flows through a concentrated area. The breach represents an increase in flood flow cross-sectional area, thereby reducing the water surface elevation in the vicinity of the breach. Flood flows will expand into the receiving flood plain downstream of the roadway resuming its course as outlined in the PMF analysis. The primary impact will be that the site will be isolated until road repairs can be performed. The flood water surface elevation will be a minimum of 4.5 feet below the cask pad top elevation. Therefore, the staff has determined that there will be no flooding impacts to the cask storage area.

The purpose of the road berm is to confine flows and maintain a separation of Basin A and Basin B flood waters. In the event that the road berm fails, flood water will reach the Facility berm and then be diverted back to the Basin A flood plain east of the Facility. In the event the road berm should fail, the water surface elevation will rise approximately 3 feet. However, at

least a 2-foot differential will exist from the water surface elevation to the cask pad top elevation. Therefore, failure of the road berm has no flood impact on the cask storage area.

Failure of any embankment or combination of embankments resulting from the PMF does not impact the safety of the cask storage because the pads remain dry in all scenarios. As a result, the cumulative effect of these embankments is not important to safety and, consequently, is not presented.

The staff reviewed the analysis for potential dam failures and found it acceptable because the applicant has demonstrated that the potential embankment failures will not result flooding of the cask storage pad. The staff has determined that this information is acceptable for use in other sections of the SAR to develop the design bases of the Facility, perform additional safety analysis, and demonstrate compliance with the regulatory requirements of 10 CFR 72.90(c), 72.90(d), 72.90(f), and 72.122(b) with respect to this issue.

2.1.4.5 Probable Maximum Surge and Seiche Flooding

The nearest body of water is the Great Salt Lake, located over 24 miles north of the site. The site elevation is approximately 460 feet above the lake. A wave over 400 feet high would have to be generated at the Great Salt Lake and travel 24 miles to reach the site. Therefore, surge or seiche flooding would not impact the site.

The staff reviewed the discussion on probable maximum surge and seiche flooding and found it acceptable because this phenomenon will not impact the site.

2.1.4.6 Probable Maximum Tsunami Flooding

The site is not located adjacent to a coastal area. Therefore, flooding attributed to seismically-induced ocean waves is not applicable to the site.

The staff reviewed the discussion on probable maximum tsunami flooding and found it acceptable because this phenomenon will not impact the site.

2.1.4.7 Ice Flooding

The Facility is 24 miles from the nearest body of water. Closer to the site, the pooling or ponding of water is prevented by the semiarid climate and geologic conditions present. Therefore, ice flooding will not impact the site.

The staff reviewed the discussion on probable maximum ice flooding and found it acceptable because this phenomenon will not impact the site.

2.1.4.8 Flood Protection Requirements

The proposed cask storage pad elevations remain flood dry during the PMF event.

2.1.4.9 Environmental Acceptance of Effluents

The applicant states the sanitary sewer is the only liquid release during site operations and will not contain radioactive effluents.

The staff reviewed the discussion on environmental acceptance of effluents and found it acceptable because there will be no radioactive effluents.

2.1.5 Subsurface Hydrology

The staff has reviewed the information presented in Section 2.5, Subsurface Hydrology, of the SAR. Subsections discussed include (i) regional characteristics, (ii) site characteristics, and (iii) contaminant transport analysis. The staff reviewed the discussion on subsurface hydrology with respect to the following regulatory requirements:

- 10 CFR 72.98(a) requires that the regional extent of external phenomena, man-made or natural, that are used as a basis for the design of the ISFSI be identified.

- 10 CFR 72.98(b) requires that the potential regional impacts due to the construction, operation or decommissioning of the ISFSI must be identified. The extent of regional impacts must be determined on the basis of potential measurable effects on the population or the environment from ISFSI activities.

- 10 CFR 72.98(c) requires that those regions identified pursuant to paragraphs 10 CFR 72.98 (a) and (b) must be investigated as appropriate with respect to: (1) The present and future character and the distribution of population, (2) Consideration of present and projected future uses of land and water within the region, and (3) Any special characteristics that may influence the potential consequences of a release of radioactive material during the operational lifetime of the ISFSI.

- 10 CFR 72.122(b) requires: (1) Structures, systems, and components important to safety must be designed to accommodate the effects of, and to be compatible with, site characteristics and environmental conditions associated with normal operation, maintenance, and testing of the ISFSI and to withstand postulated accidents. (2) Structures, systems, and components important to safety must be designed to withstand the effects of natural phenomena such as earthquakes, tornadoes, lightning, hurricanes, floods, tsunami, and seiches, without impairing their capability to perform safety functions. The design bases for these structures, systems, and components must reflect: (i) Appropriate consideration of the most severe of the natural phenomena reported for the site and surrounding area, with appropriate margins to take into account the limitations of the data and the period of time in which the data have accumulated, and (ii) Appropriate combinations of the effects of normal and accident conditions and the effects of natural phenomena. The ISFSI should also be designed to prevent massive collapse of building structures or the dropping of heavy objects as a result of building structural failure on the spent fuel or high-level radioactive

waste or on to structures, systems, and components important to safety. (3) Capability must be provided for determining the intensity of natural phenomena that may occur for comparison with design bases of structures, systems, and components important to safety. (4) If the ISFSI is located over an aquifer which is a major water resource, measures must be taken to preclude the transport of radioactive materials to the environment through this potential pathway.

2.1.5.1 Regional Characteristics

Skull Valley, the proposed location of the Facility, is a north-trending valley that extends from the Onaqui Mountains to the southwest shore of the Great Salt Lake. This valley is bordered by the Cedar Mountains to the west and the Stansbury Mountains to the east. Most of the precipitation that falls in the higher elevations runs off the steep hillsides as spring snowmelt, with little infiltration into the mountain blocks. Water enters the valley-fill aquifers through an extensive recharge area consisting mainly of alluvial fans at the base of the mountains. The long-term average annual runoff from the uplands is about 32,000 acre-feet. The average annual groundwater discharge and recharge is between 30,000 and 50,000 acre-feet, with evapotranspiration accounting for 80–90 percent of the discharge.

Valley-fill consists of inter-stratified colluvium, alluvium, lacustrine, and fluvial deposits with minor ash and some Eolian material. The coarser deposits are generally near the perimeter of the valley, grading into well-sorted sand and gravel and interlayered with lacustrine silt and clay toward the center of the valley. Thick beds of clay exist in some areas and may create local, confined aquifers where they interfinger with sand and gravel along the alluvial fans. The Salt Lake Group of the Tertiary age comprises most of the valley-fill with a thickness ranging from 2,000 to over 8,000 feet. The Tertiary and older Quaternary deposits are slightly to highly permeable. The deeper deposits contain some volcanic deposits and have reduced permeability due to greater consolidation. The Tertiary and Quaternary deposits contain most of the usable groundwater in the valley. The valley floor is underlain by Quaternary and Holocene sediments that generally have low permeability. Most of the surface runoff ponds in discontinuous drainage channels until it evaporates.

Groundwater flows northward to the Great Salt Lake. The annual volume of underflow out of the valley is estimated at 800 acre-feet with a transmissivity of 2,675 ft/day^2. Annual discharge from pumping is estimated at 5,000 acre-feet and is not believed to have changed significantly in the past 30 years. Most of the domestic wells are developed in the unconsolidated alluvial fan deposits along the east side of the Skull Valley. This area provides most of the local recharge and yields high quality groundwater. Groundwater is generally between 110–160 feet below ground in this area. Some irrigation and stock wells show artesian conditions in the valley due to confining layers of lake clays. Most wells are drilled to depths of 250–500 feet but maintain a static water depth of 100 feet or less.

The groundwater quality depends on well proximity to the bordering mountain ranges. The groundwater along the base of the Stansbury Mountains contains the lowest dissolved solids content in the valley, with concentrations of 100 to 800 mg/L (0.006 to 0.050 lb/ft^3). In the southernmost part of the valley, the dissolved solids content concentrations range from 700 to about 900 mg/L (0.044 to about 0.056 lb/ft^3); however, dissolved solids content concentrations as high as 2,500 mg/L (0.156 lb/ft^3) have been observed in a well south of the Reservation.

The north end of the valley generally has high dissolved solids content concentrations in the range of 1,600–7,900 mg/L (0.100 to 0.493 lb/ft^3). The main ions in the groundwater are sodium and chloride.

The staff has reviewed the discussion and information regarding regional subsurface hydrology characteristics and found it acceptable because regional characteristics have been adequately described for further assessment of external events and the impact of the facility on present and future groundwater use in the region is negligible. The staff has determined that this information is acceptable for use in other sections of the SAR to develop the design bases of the Facility, perform additional safety analysis, and demonstrate compliance with regulatory requirements in 10 CFR 72.98(c)(2) and 72.122(b) with respect to this issue.

2.1.5.2 Site Characteristics

The applicant has classified the subsurface material at the proposed site as a relatively compressible top layer, approximately 25 to 30 feet thick, that is underlain by much denser and stiffer material. The underlaying layer is classified as dense sand and silt. The onsite boreholes, when drilled to a depth of 100 feet, did not intercept the water table. The groundwater table is greater than 100 feet below grade at the site. Based on regional studies in Skull Valley, the groundwater flows from the south to the north, toward Great Salt Lake. The hydraulic gradient is estimated at 9.5×10^{-4}. The permeability of silt soil in the Skull Valley ranges from 0.2 to 0.6 in/hr. The average estimated groundwater velocity ranges from 5.08×10^{-4} to 1.136×10^{-3} ft/day. The precipitation at the site does not contribute to the groundwater flow due to low permeability of surficial deposits and high rates of evapotranspiration. The groundwater flow beneath the site is mainly derived from precipitation at the higher elevations of the Stansbury Mountains.

The staff has reviewed the discussion and information regarding the site characteristics and found it acceptable because the groundwater characteristics have been adequately described for further assessment of external events. The staff has determined that this information is acceptable for use in other sections of the SAR to develop the design bases of the Facility, perform additional safety analysis, and demonstrate compliance with regulatory requirements in 10 CFR 72.98(c)(2) and 72.122(b) with respect to this issue.

2.1.5.3 Contaminant Transport Analysis

A hydrologic transport analysis was not included in the SAR because release of effluents from the Facility is not expected. The facility and cask designs are expected to preclude release of effluents for normal, off-normal, and accident conditions.

The staff has reviewed the discussion on contaminant transport analysis and has determined it to be acceptable because release of effluents from the Facility is not expected. The staff has determined that this information is acceptable for use in other sections of the SAR to develop the design bases of the Facility, perform additional safety analysis, and demonstrate compliance with regulatory requirements in 10 CFR 72.98(c)(2) and 72.122(b) with respect to this issue.

2.1.6 Geology and Seismology

Section 2.6 of the SAR, Geology and Seismology, describes the geological and seismological setting of the proposed site, geographically located within Skull Valley, Utah. This review corresponds to the following sections of the SAR: 2.6.1, Basic Geologic and Seismic Information; 2.6.2, Vibratory Ground Motion; 2.6.3, Surface Faulting; 2.6.4, Stability of Subsurface Materials; and 2.6.5, Slope Stability. The review includes the applicant's SAR (Private Fuel Storage Limited Liability Company, 2001), responses to requests for additional information (Parkyn, 1999a, 2001; Donnell, 1999d, 2001), and additional supporting documents (Geomatrix Consultants, Inc., 1999a, 2001a, 2001b, 2001c, and 2001d; Bay Geophysical Associates, Inc., 1999; Northland Geophysical, L.L.C., 2001; Stone and Webster Engineering Corporation, 2001a, 2001b, 2001c, 2001d). The staff reviewed the geology and seismology of the site with respect to the following regulatory requirements:

- 10 CFR 72.90(a) requires site characteristics that may directly affect the safety or environmental impact of the ISFSI to be investigated and assessed.
- 10 CFR 72.90(b) requires proposed sites for the ISFSI to be examined with respect to the frequency and severity of external natural and man-induced events that could affect the safe operation of the ISFSI.
- 10 CFR 72.90(c) requires design basis external events to be determined for each combination of proposed site and proposed ISFSI design.
- 10 CFR 72.90(d) requires the proposed sites with design basis external events for which adequate protection cannot be provided through ISFSI design be deemed unsuitable for the location of the ISFSI.
- 10 CFR 72.92(a) requires that natural phenomena that may exist or that can occur in the region of a proposed site must be identified and assessed according to their potential effects on the safe operation of the ISFSI. The important natural phenomena that affect the ISFSI design must be identified.
- 10 CFR 72.92(b) requires that records of the occurrence and severity of those important natural phenomena must be collected for the region and evaluated for reliability, accuracy, and completeness. The applicant shall retain these records until the license is issued.
- 10 CFR 72.92(c) requires that appropriate methods must be adopted for evaluating the design basis external natural events based on the characteristics of the region and the current state of knowledge about such events.
- 10 CFR 72.98(a) requires that the regional extent of external phenomena, man-made or natural, that are used as a basis for the design of the ISFSI be identified.
- 10 CFR 72.98(b) requires that the potential regional impact due to the construction, operation or decommissioning of the ISFSI be identified. The extent

of regional impacts must be determined on the basis of potential measurable effects on the population or the environment from ISFSI activities.

- 10 CFR 72.98(c) requires that those regions identified pursuant to paragraphs 10 CFR 72.98 (a) and (b) must be investigated as appropriate with respect to: (1) The present and future character and the distribution of population, (2) Consideration of present and projected future uses of land and water within the region, and (3) Any special characteristics that may influence the potential consequences of a release of radioactive material during the operational lifetime of the ISFSI.

- 10 CFR 72.102 (b) requires that West of the Rocky Mountain Front (west of approximately 104 west longitude), and in other areas of known potential seismic activity, seismicity will be evaluated by the techniques of Appendix A of Part 100 of this chapter. Sites that lie within the range of strong near-field ground motion from historical earthquakes on large capable faults should be avoided.

- 10 CFR 72.102(c) requires that sites other than bedrock sites must be evaluated for their liquefaction potential or other soil instability due to vibratory ground motion.

- 10 CFR 72.102(d) requires that site-specific investigations and laboratory analyses must show that soil conditions are adequate for the proposed foundation loading.

- 10 CFR 72.102(e) requires that in an evaluation of alternative sites, those which require a minimum of engineered provisions to correct site deficiencies are preferred. Sites with unstable geologic characteristics should be avoided.

- 10 CFR 72.102(f) requires that the design earthquake for use in the design of structures must be determined as follows: (1) For sites that have been evaluated under the criteria of Appendix A of 10 CFR Part 100, the design earthquake must be equivalent to the safe shutdown earthquake for a nuclear power plant. (2) Regardless of the results of the investigations anywhere in the continental U.S., the design earthquake must have a value for the horizontal ground motion of no less than 0.10 g with the appropriate response spectrum.

- 10 CFR 72.122(b) requires (1) Structures, systems, and components important to safety must be designed to accommodate the effects of, and to be compatible with, site characteristics and environmental conditions associated with normal operation, maintenance, and testing of the ISFSI and to withstand postulated accidents. (2) Structures, systems, and components important to safety must b designed to withstand the effects of natural phenomena such as earthquakes, tornadoes, lightning, hurricanes, floods, tsunami, and seiches, without impairing their capability to perform safety functions. The design bases for these structures, systems, and components must reflect: (i) Appropriate consideration of the most severe of the natural phenomena reported for the site and surrounding area, with appropriate margins to take into account the limitations of the data and the period of time in which the data have accumulated, and (ii)

Appropriate combinations of the effects of normal and accident conditions and the effects of natural phenomena. The ISFSI should also be designed to prevent massive collapse of building structures or the dropping of heavy objects as a result of building structural failure on the spent fuel or high-level radioactive waste or on to structures, systems, and components important to safety. (3) Capability must be provided for determining the intensity of natural phenomena that may occur for comparison with design bases of structures, systems, and components important to safety. (4) If the ISFSI is located over an aquifer which is a major water resource, measures must be taken to preclude the transport of radioactive materials to the environment through this potential pathway.

Summary of Review

The staff reviewed information presented in Section 2.6 of the SAR, Geology and Seismology. The staff also reviewed relevant literature cited in the SAR and and supporting documents. In Revision 18 of the SAR (Private Fuel Storage Limited Liability Company, 2000), geologic and seismic information included (i) review of published and unpublished literature; (ii) reconnaissance geological mapping of the valley; (iii) the test boring program performed by Earthcore, Inc. under the supervision of Stone & Webster Engineering Corporation (Appendix 2A); (iv) P and S wave seismic reflections surveys performed by Geosphere Midwest under the supervision of Stone & Webster Engineering Corporation (Appendix 2B); (v) the DSHA performed by Geomatrix Consultants, Inc. (Appendix 2D); (vi) a consulting report on surface geomorphology features prepared by D. Curry of the University of Utah, at the request of Stone & Webster (Appendix 2C); (vii) consultant reports on the composition and age of volcanic ash layers found in the test borings prepared by W. Nash of the University of Utah (Appendix 2E); and (viii) additional seismic evaluation (Appendix 2G).

The applicant's response to Round 1 RAIs 2-5 and 2-7 (Parkyn, 1999a; Donnell, 1999d; Geomatrix Consultants, Inc., 1999a; Bay Geophysical Associates, Inc., 1999), provided an extensive 8-month geological and geophysical investigation of the site. The additional analyses are summarized by Geomatrix Consultants, Inc. (1999a) and Bay Geophysical Associates, Inc. (1999). Additional information provided in support of fault displacement and seismic hazard assessments includes (i) a 1:12,000-scale compilation map of geology and surface features; (ii) supplementary discussions with R.B. Smith, R. Bruhn, and W. Arabasz of the University of Utah, and J.M. Helm, all professional researchers with expert knowledge of local and regional geological and geophysical conditions; (iii) two regional cross sections showing possible relationships of faults to the depth of the seismogenic crust; (iv) photo-geologic interpretations of low-sun-angle photographs of geomorphic features; (v) reconnaissance field investigations of active faulting along southern segments of the Stansbury fault; (vi) existing proprietary gravity data of the valley, previously collected by EDCON in support of petroleum exploration; (vii) 3.8 miles of high-resolution S-wave seismic reflection data acquired by Bay Geophysical Associates, Inc. (1999); (viii) reprocessed industrial P-wave reflection seismic data; (ix) ground magnetic and electric conductivity data acquired to assess the feasibility of additional ground penetrating radar studies; (x) 30 new boreholes drilled across the site to provide additional control on subsurface stratigraphy and to support surface mapping and subsurface geophysical investigations; (xi) 25 test pits and 2 trenches excavated on the site to provide detailed profiles of near-subsurface faulting and stratigraphy; (xii) geochronologic age dating to determine radiometric ages of important stratigraphic horizons used to correlate paleo-lake deposits and

confirm ages of inferred Bonneville Lake cycle stratigraphy; (xiii) the applicant's PSHA; and (xiv) a probabilistic fault displacement assessment.

The applicant provided documentation (Donnell, 1999g) on formulation of ground-motion and fault-displacement hazards, including the methodology used to develop the probabilistic seismic and fault displacement hazard assessments and the applicability of methods and results for this analysis developed in conjunction with the U.S. Department of Energy (DOE) seismic hazard analyses at Yucca Mountain, Nevada. Discussion included how models and data generated by DOE expert elicitation for ground-motion and fault-displacement hazards at Yucca Mountain are applicable to the applicant's site in Skull Valley, Utah.

The applicant also provided (i) gravity data used to support geological interpretations of the site geology (specifically, the industry EDCON gravity data set and gravity profiles collected by J. Baer at Brigham Young University), (ii) assessments of near-field ground motions from earthquakes that could possibly occur on faults near the site, and (iii) updated deterministic ground-motion assessment for the site based on recent revisions to the site characterization for comparison to probabilistic assessment (Donnell, 1999h; Parkyn, 1999b).

The applicant completed additional site characterization work in January, 2001 to verify shear-wave velocity information in the upper strata of the soil column (approximately 35 feet) at the PFS Facility site. In particular, shear-wave velocity data were collected by Northland Geophysical, L.L.C (2001) in two boreholes located near the planned Waste Handling Facility CTB-5(OW) and CTB-5A. The applicant also performed additional site-response modeling to evaluate the effects of alternative interpretations of the soil shear-wave velocity, modulus reduction, and damping on the ground motion hazard.

In March 2001, Geomatrix Consultants, Inc., (2001a, and 2001d) revised the dynamic soil properties, proposing a 9-sediment-layer soil model as representative of the site for a depth of 700 feet. Six of these layers are located within the upper 35 feet. These changes in the dynamic soil properties also led the applicant to make two changes to the ground motion attenuation models.

Based on the revised site model, the applicant concluded in the SAR that the PFS site had properties closer to that of a generic western United States rock site (Geomatrix Consultants, Inc., 2001a, Appendix F) than the soil model it had used previously. In Revision 18 of the SAR (Private Fuel Storage Limited Liability Company, 2000), the applicant likened the Skull Valley soils to a generic western United States deep soil site (Geomatrix Consultants, Inc., 1999a, Appendix F). This information and re-characterization of the PFS site led to a change in the set of empirical ground motion attenuation models that were the starting point for the development of the PFS ground motion models.

A second revision to the applicant's methodology involved the site adjustment factors to account for the differences in the generic western United States site and the proposed PFS site in Skull Valley. In Revision 18 of the SAR (Private Fuel Storage Limited Liability Company, 2000), the applicant used a site response model to estimate site response adjustment factors (Geomatrix Consultants, Inc., 1999a, Appendix F). In the revised analysis presented in Revision 22 of the SAR (Fuel Storage Limited Liability Company, 2001; Geomatrix Consultants, Inc., 2001a, Appendix F), the applicant considered two alternative approaches to estimate the site response factors. The first approach was based on site response modeling and the new

site model. The second approach derived site adjustment factors from empirical strong motion data. The final site adjustment factor was determined from a weighting of the two approaches. The applicant assigned a 1/3 weight to the empirical strong motion approach and a 2/3 weighting to the site response modeling approach.

The staff evaluated the revised information submitted by the applicant in the revised SAR and all pertinent documents and analyses (revised and new). This new and revised information includes the following references: (Private Fuel Storage Limited Liability Company, 2001); (Geomatrix Consultants, Inc., 2001a, 2001b, 2001c, and 2001d); (Northland Geophysical, L.L.C., 2001; (Stone and Webster Engineering Corporation, 2001a,b,c,d); and (Donnell, 2001; Parkyn, 2001).

The staff has reviewed the information presented in Section 2.6, Geology and Seismology in the SAR regarding the site. The documentation is acceptable because the breadth and depth of geological and geophysical investigations, especially those reported in Geomatrix Consultants, Inc. (1999a; 2001a), represent a comprehensive technical foundation of geological knowledge from which the potential for seismic and faulting hazards at the site can be adequately deduced. The applicant has sufficiently documented these investigations in the SAR and supporting documents. The staff has determined that this information is acceptable for use in other sections of the SAR to develop the design bases of the Facility, perform additional safety analysis, and demonstrate compliance with regulatory requirements in 10 CFR 72.92(a), 72.92(b), and 72.102(e) with respect to this issue.

2.1.6.1 Basic Geologic and Seismic Information

Basic geologic and seismic characteristics of the site and vicinity are presented in Section 2.6.1 of the SAR, Basic Geological and Seismological Information. These include discussions of physiographic background and site geomorphology, regional and site geological history, structural geologic conditions, and engineering evaluation of geologic features. Detailed static and dynamic engineering properties of soil and rock underlying the site are presented in Section 2.6.4 of the SAR, Stability of Subsurface Materials.

Physiography and Site Geomorphology

As summarized in the SAR, the proposed site is located in the northeastern margin of the Basin and Range Province, a wide zone of active extension and distributed normal faulting that extends from the Wasatch Front in central Utah to the Sierra Nevada Mountains in western Nevada and eastern California. Topography within the Basin and Range Province reflects Miocene to recent, east-west extensional faulting, in which tilted and exhumed footwall blocks form subparallel north-south striking ranges separating elongated and internally drained basins. Ranges are up to several hundred kilometers long with elevations up to 6,500 feet above the basin floors. Much of the surface faulting took place at the base of the ranges along normal faults that dip moderately (~60°) beneath the adjacent basins (herein defined as range-front faults), although complex faulting within the basins is also common [i.e., the fault-rupture patterns of the 1954 Rainbow Mountain-Stillwater or 1959 Hebgen Lake earthquakes as summarized in dePolo et al. (1991)].

The proposed site in Skull Valley lies in one of the typical basins of the province, bounded on the east by the Stansbury Mountains and the Stansbury fault and on the west and south by the Cedar Mountains and the East Cedar Mountain fault. The basin is underlain by late Quaternary lacustrine deposits laid down from repeated flooding of the valley during transgressions of intermontane lakes, most notably Lake Bonneville, which flooded Skull Valley several times during the Pleistocene and Holocene (e.g., Currey and Oviatt, 1985). These deposits form the basis for paleoseismic evaluations of the Skull Valley site. Topography of the proposed site is relatively smooth, reflecting the origin of the valley floor as the bottom of Lake Bonneville. The site gently slopes to the north with a slope of less than 0.1°. Detailed topographic maps of the region and the site were provided in the SAR. This smooth valley floor contains small washes up to 4 feet deep and soil ridges up to 4 feet high.

The geomorphology of Skull Valley in the vicinity of the site is typical of a semiarid to arid desert setting. The adjacent mountain ranges are affected by mass-wasting processes and stream erosion that deliver sediment loads to a complex of alluvial fans (aprons) situated at the bases of the ranges. Runoff is conveyed down the ranges and over the alluvial fans through a series of small channels to the valley floor. Stream and spring flows are absorbed into the fan and the valley floor near the fan-floor interface, resulting in minimal surface runoff reaching the central valley near the site. There is no evidence of flash-flooding near the site nor are there deposits indicative of geologically recent [last 2 Ma (million years)] mudflows or landslides.

The valley floor near the site comprises beach ridges and shoreline deposits interrupted by bedrock outcrops, such as Hickman Knolls rising about 400 feet above the valley bottom. The valley bottom relief comprises a series of braided, northerly flowing dry washes. The washes are disrupted and convey runoff for only short distances before merging into other washes or open space. This network of shallow washes extends offsite to the north where it confluences with the central valley drainage system and from there flows to the Great Salt Lake. The only perennial surface water is located approximately 10 miles north of the site. The central valley in the vicinity of the Facility is unaffected by fluvial processes.

In the southern and eastern parts of the proposed site, numerous north-trending linear sand ridges interrupt the otherwise smooth valley floor. The ridges, which are typically 8 feet high and 100 feet wide, were originally mapped as possible fault traces by Sack (1993). In the SAR, a brief summary report (Appendix 2C) reviewed the available surficial information and concluded that these features constitute sandy beach ridges deposited by southward longshore transport within the Stansbury shoreline coastal zone of Lake Bonneville. The applicant provided technical information (Parkyn, 1999a) about the nature and origin of the ridges to substantiate the conclusions reached in Appendix 2C of the Revision 2 of the SAR. This information, especially Figure 1-3 and associated discussion in Section 5.2.1 of Geomatrix Consultants, Inc. (1999a), was sufficient to document the conclusions. In addition, discussion of the stratigraphic relationships in test pit T-11 (Geomatrix Consultants, Inc., Revision 10, 1999a, Volume II, Figure C-1) provided additional technical information in support of the conclusion that these ridges have a depositional not tectonic origin.

In a few locations, bedrock composed of Paleozoic carbonate rocks crop out of the smooth valley floor. The largest of these is a small group of hills 1.3 miles south of the proposed site known as Hickman Knolls. Rocks of this outcrop are medium to dark gray dolomite breccia. The origin and stratigraphic correlation of the Hickman Knolls carbonate rocks within the Paleozoic section is not well known. The preferred interpretation put forth by Geomatrix

Consultants, Inc. (1999a) is that they are rooted bedrock outcrops. The alternative interpretation based on independent modeling of gravity data by the staff (Stamatakos et al., 1999) is that they are landslide deposits, resting unconformably on the Tertiary sediments in the valley. Geomatrix Consultants, Inc. (1999a) correlated them with the Upper Ordovician Fish Haven Formation based on descriptions of the regional stratigraphy by Hintze (1988) and the geological bedrock maps of Teichert (1959) and Rigby (1958). The differences in these two interpretations lead to differences in the estimated seismic hazard. In the Geomatrix preferred interpretation, rooted bedrock requires a significant and seismogenic fault just west of Hickman Knolls. In the alternative interpretation, no such fault is necessary. Therefore, the Geomatrix preferred interpretation leads to a slightly more conservative seismic hazard (see Stamatakos et al., 1999, for complete discussion).

The applicant's surface mapping and related field investigations (Geomatrix Consultants, Inc. 1999a) are sufficient to show that Hickman Knolls shows no evidence of significant karst features (e.g., collapsed solution cavities). Karstification is also not widespread in carbonate bedrock of the surrounding ranges. Because similar rocks lie beneath the valley floor, the staff concludes that karst processes have not affected the site and are not a concern to site suitability.

Regional and Site Geologic History

The SAR discusses the geological history of the site and surrounding region. The discussion includes background information about the tectonic setting of the region in the Precambrian and Paleozoic that led to the deposition of the bedrock stratigraphy presently exposed in the Stansbury and Cedar Mountains. In brief, the structural framework of bedrock across the region reflects overprinting of several major periods of North American tectonic activity. These include contractional deformation structures such as thin- and thick-skinned thrusts and folds associated with the Devonian Antler, Jurassic to Cretaceous Sevier, and Cretaceous-Tertiary Laramide orogenies (e.g., Cowan and Bruhn, 1992) and extensional normal and detachment faults associated with the Eocene to the current Basin and Range extension (e.g., Wernicke, 1992; Axen et al., 1993).

The proposed site lies near the center of a typical Basin and Range valley, situated between roughly north-south and northwest-southeast elongated ranges of exhumed bedrock. Exhumation of the ranges was accomplished by extensional faulting along range-front normal faults. Faulting tilted the ridges to the east. The adjacent basins subsided concomitant with exhumation while they accumulated sediment shed from the eroding ranges. In Skull Valley, as in much of central and western Utah, the valleys are also flooded by transgressions of the intramontane saline lakes. Tertiary and Quaternary deposits in and around the site document numerous transgressions associated with Lake Bonneville and pre-Lake Bonneville lacustrian cycles. The Great Salt Lake is the present-day remnant of Lake Bonneville.

In the SAR, the structural framework of the site within the valley is based on interpretations presented in the available literature integrated with detailed site geological studies, including site stratigraphy, geologic mapping, cross-sectional construction, and geophysical investigations (Geomatrix Consultants, Inc.,1999a; Bay Geophysical Associates, Inc., 1999). Most important to the evaluations of seismic and faulting hazards was identification and

characterization of a detailed Quaternary stratigraphy, that provided critical constraints on faulting activity and local and regional active faults.

Valley fill sediments in Skull Valley consist of Tertiary age siltstones, claystones, and tuffaceous sediments overlain by Quaternary lacustrian deposits. Late Miocene to Pliocene deposits of the Salt Lake Formation were exposed in Trench T1 and in Boring C-5. Microprobe analyses of glass shards from vitric tuffs (ash fall deposits) within the sediments were used to correlate the tuffs with volcanic rocks of known age. The analyses indicate ages for the stratigraphic units between 16 and 6 Ma consistent with the known age of the Salt Lake Formation. Microprobe analyses, performed by M. Perkins at the University of Utah, are documented in Appendix D of Geomatrix Consultants, Inc. (1999a).

During the Quaternary (approximately the last 2 Ma), especially the last 700 ka (thousand years), sedimentation in Skull Valley was dominated by fluctuations associated with lacustrian cycles in the Bonneville Basin (e.g., Machette and Scott, 1988; Oviatt, 1997). The SAR provides a detailed analysis of these deposits from trenches, test pits, and borings, including two radiocarbon ages on ostracodes and charophytes. The radiocarbon ages were performed by Beta Analytic, Inc. under the direction of G. Hood and are documented in Appendix D of Geomatrix Consultants, Inc. (1999a).

The stratigraphy was also critical to interpretations of the reflection seismic profiles. Two prominent paleosols were developed during inter pluvial periods near the Tertiary-Quaternary boundary (~2 Ma) and between the Lake Bonneville and Little Valley cycles (130–28 Ka). These buried soils are characterized by relatively well-developed pedogenic carbonate, both in the soil matrix and as coatings on pebbles. As such, these paleosols form strong reflectors that are readily apparent on the seismic reflection profiles. These horizons were also correlated with cores from the borings drilled directly beneath the seismic profile lines. These detailed constraints on the Quaternary stratigraphy and the high quality seismic reflection profiles provided in Geomatrix Consultants, Inc. (1999a) are sufficient to document the Quaternary faulting record of the site and to provide a necessary stratigraphic framework for reliable paleoseismic analyses of active faults in and around Skull Valley.

Structural Geologic Conditions

Primary faults. Classical structural models for the Basin and Range envision a simple horst and graben framework in which range-front faults are planar and extend to the base of the transition between the brittle and ductile crust, 9–12.5 miles below the surface (e.g., Stewart, 1978). More recent work has shown that many normal faults are not planar but curved or listric, and they sole into detachments that may or may not coincide within the brittle-ductile transition in the crust (e.g., Wernicke and Burchfiel, 1982). In Skull Valley, the detachment model places the Stansbury fault as the master or controlling fault of a half graben. The other side of the half graben would include the antithetic East Cedar Mountain fault and a series of antithetic and synthetic faults within the basin, all of which would sole into the Stansbury fault 1–12.5 miles deep in the crust. Details of these two alternatives to fault geometry are discussed in Stamatakos et al. (1999).

In Geomatrix Consultants, Inc. (1999a), two regional cross sections were developed that depict the overall structural framework of Skull Valley and the surrounding ranges. These cross

sections were constructed from a compilation and analysis of existing geological map data, reprocessed and new seismic profiles across the valley, and interpretation of proprietary gravity data. The cross sections were based on acceptable structural geology procedures for cross-sectional restoration and interpretation of subsurface geometries (e.g., Woodward et al., 1989; Suppe, 1983). The cross sections depict a series of pre-Tertiary folds and thrusts related to the Sevier and older contraction deformation that have been cut by a series of Tertiary and Quaternary normal faults related to Basin and Range extension. The normal faults are considered moderately dipping (~60°) planar features following the horst and graben model described previously.

As discussed in Stamatakos et al. (1999), this horst and graben model is conservative for predicting a maximum earthquake potential for these faults. Faults that extend all the way to the base of the seismogenic crust define a larger area for earthquake rupture and thus greater maximum magnitude earthquakes than those that terminate into a detachment above the brittle-ductile transition. The added feature of a detachment beneath the valley does not contribute to the earthquake hazard because large earthquakes on detachment faults are exceedingly rare or nonexistent (Wernicke, 1995; Ofoegbu and Ferrill, 1997). The staff notes the horst and graben model does not consider the possibility of triggered ruptures (e.g., rupture of the master basin fault triggering subsequent co-seismic ruptures on the opposing antithetic or synthetic faults in the basin). This is acceptable because the faults act independently.

The cross sections show three first-order, west-dipping normal faults and one east-dipping fault (the East Cedar Mountain fault). The west-dipping faults are the Stansbury and two previously unknown faults in the basin informally named the East and West faults. These new faults were interpreted based mainly on analyses of the gravity and seismic reflection data and by analogy to other faults in the Basin and Range. Discovery of these new faults and related structures has important implications to both the seismic and fault displacement hazard assessments (see Sections 2.1.6.2 and 2.1.6.3).

A critical aspect of the interpretation of the East and West faults centers on the origin and nature of rocks exposed at Hickman Knolls, which are composed of monolithologic carbonate breccias. Two possibilities were presented in the SAR:

(1) The breccias are part of a detached landslide block of a bedrock dislodged from one of the nearby ranges by Tertiary or Quaternary earthquake activity along the range fronts.

(2) The breccias are rooted to the Paleozoic basement beneath the basin fill. (In this latter interpretation, brecciation and related features represent *in situ* deformation associated with early post-depositional processes.)

Alternative (1) was based on an interpretation of gravity data collected and analyzed by J. Baer of Brigham Young University. Indeed, many characteristics of the Hickman Knolls breccias are similar to mapped landslide deposits throughout the Basin and Range Province (e.g.,Yarnold, 1993; Bishop, 1997). Observations of chaotic and low-angle faulting and folding of the Tertiary deposits in Trench T-1 also suggest Tertiary landslide activity (Geomatrix Consultants, Inc., 1999a).

Alternative (2) was based on the Geomatrix Consultants, Inc. (1999a) interpretation of the proprietary industry gravity data and detailed mapping of the meso-scale structures at Hickman Knolls. Deformation features, especially low-angle and high-temperature ductile shears overprinted by minor low-temperature and brittle faults and fractures, suggest a protracted history of *in situ* deformation of rooted bedrock. In this interpretation, the deformation of the Tertiary sediments in Trench T-1 are considered to represent a local landslide that originated on the flanks of Hickman Knolls itself.

The difference between these alternatives is important to structural interpretations of Skull Valley. In alternative (1), the significant structural relief of the basin would lie east of Hickman Knolls along both the East and Stansbury faults. This interpretation would reduce cumulative displacement along the East fault and thereby reduce its contribution to the overall seismic hazard. This interpretation is represented in Geomatrix Consultants, Inc. (1999a) seismogenic fault rupture Model A. In alternative (2), major relief in the basin lies west of the Knolls with significant displacement along the West fault. In this alternative, the West fault becomes a significant contributor to the overall seismic hazard as represented in Geomatrix Consultants, Inc. (1999a) seismogenic fault rupture Model B. Alternative (2) is favored in Geomatrix Consultants, Inc. (1999a), although some credence is given to alternative (1). In building the logic tree for seismogenic sources in the PSHA, alternative (1) is given a weight of 0.3 and alternative (2) is given a weight of 0.7 (see discussion in Section 2.1.6.2).

Independent analysis of EDCON gravity data provided in the SAR (Stamatakos et al., 1999) favors alternative (2). The West fault appears to be a splay of the East fault and, therefore, not capable of independently triggering earthquakes. Given that Geomatrix Consultants, Inc. (1999a) included the West fault coupled with other conservative assumptions about seismicity, Stamatakos et al. (1999) concluded that the Geomatrix Consultants, Inc. assessment has led to a conservative hazard assessment, in terms of the seismic source characterization.

Secondary faults. Within the valley fill itself, the SAR documents several additional secondary faults designated as fault zones A to F. Each fault zone has a number of secondary splays that are designated with numeral subscripts (e.g., A1 to A7, B1 and B2, and so forth). These fault zones are all considered secondary faults related to deformation of the hanging wall above the larger East and West faults. They are too small to be independent seismic sources but large enough to be considered important in the fault displacement analysis. The largest of the secondary faults is F fault, which appears to be a splay of the East fault. The characteristics of these secondary faults and their contributions to the surface faulting hazard at the proposed site are discussed in detail in Section 2.1.6.3 of this SER.

Engineering Evaluation of Geologic Features

The static and dynamic engineering soil and rock properties of the various materials underlying the site are evaluated in Section 2.1.6.4 of this SER. The properties evaluated include grain size classification, Atterberg limits, water content, unit weight, shear strength, relative density, shear modulus, Poisson's ratio, bulk modulus, damping, consolidation characteristics, seismic wave velocities, density, porosity, strength characteristics, and strength under cyclic loading.

Staff Review

The staff reviewed the information in Section 2.6.1 of the SAR and found it acceptable because the basic geologic and seismic characteristics of the site and vicinity have been adequately described in detail to allow investigation of seismic characteristics of the Facility. The staff has determined that this information is acceptable for use in other sections of the SAR to develop the design bases of the Facility, perform additional safety analysis, and demonstrate compliance with regulatory requirements in 10 CFR 72.92(a), 72.92(b), 72.102(e), and 72.122(b) with respect to this issue.

2.1.6.2 Ground Vibration and Exemption Request

Earthquake ground motion is discussed in Section 2.6.2 of the SAR, Vibratory Ground Motion. In the SAR, vibratory ground motion is addressed through discussions of historical seismicity and procedures to determine the design earthquake, including identification of potential seismic sources and their characteristics, correlation of earthquake activity with geologic structures, maximum earthquake potential, and seismic wave transmission characteristics.

According to 10 CFR 72.122(b)(2), structures, systems, and components important to safety must be designed to withstand the effects of natural phenomena, including earthquakes, without impairing their capability to perform safety functions. For sites west of the Rocky Mountains, such as Skull Valley, 10 CFR Part 72 requires that seismicity be evaluated by techniques set forth in Appendix A of 10 CFR Part 100 for nuclear power plants. This appendix defines the safe shutdown earthquake as the earthquake that produces the maximum vibratory ground motion at the site, and requires that the structures, systems, and components be designed to withstand the ground motion produced by the safe shutdown earthquake. This seismic design method implies use of a DSHA approach because it considers only the most significant event, and the method is a time-independent statement (i.e., it does not take into consideration the planned operating period of the Facility or how frequent or rare the seismic events are that control the deterministic ground motion). Also, 10 CFR 72.102(f)(1) requires that analyses using the Appendix A methodology use a design peak horizontal acceleration equivalent to that of the safe shutdown earthquake for a nuclear power reactor.

A detailed geological survey conducted by Geomatrix Consultants, Inc.(1999a) identified additional faults in the vicinity of the site. Taking into account these newly discovered faults with the DSHA methodology, in revision 18 of the SAR (Private Fuel Storage Limited Liability Company, 2000), the applicant estimated the peak horizontal and vertical acceleration values from the seismic event to be 0.72 and 0.80g, respectively (Geomatrix Consultants, Inc., 1999b). In Revision 22 of the SAR (Private Fuel Storage Limited Liability Company, 2001), using the DSHA methodology, the applicant estimated the peak horizontal and vertical acceleration values from a seismic event to be 1.15g and 1.17g respectively (Geomatrix Consultants Inc., 2001d). These values exceed the SAR proposed design values.

To resolve the issue of seismic design, the applicant submitted to the NRC, a request for an exemption to the seismic design requirement of 10 CFR 72.102(f)(1) to use PSHA along with considerations of risk to establish the design earthquake ground motion levels at the Facility (Parkyn, 1999b). The exemption request also proposed to design the Facility to the ground motions produced by 1,000-year return period earthquakes. Based on information supporting

Revision 18 of the SAR (Private Fuel Storage Limited Liability Company, 2000), these design-ground motions were calculated to have a peak horizontal acceleration of 0.40g and a peak vertical acceleration of 0.39g, resulting from a recent site-specific PSHA conducted by the Geomatrix Consultants, Inc. (1999a). These values were subsequently updated in Revision 22 of the SAR (Private Fuel Storage Limited Liability Company, 2001), as discussed below.

As part of the evaluation of PFS's exemption request, the staff conducted an independent technical review of seismic hazard investigations at the proposed site (Stamatakos et al. 1999). The objectives of this seismic investigation were to (i) conduct an independent review of existing seismic hazard studies at Skull Valley, in particular, to identify seismic and faulting issues important to siting the Facility; (ii) evaluate the adequacy and acceptability of PFS's seismic design approach; and (iii) determine an appropriate design basis return period for the PFS-proposed seismic design approach. The staff conducted its evaluation by reviewing information provided by the applicant, surveying other state-of-the-art literature, analyzing the bases of current NRC regulations, and performing independent analyses of geophysical data and sensitivity studies of model alternatives and consideration of uncertainties. This section of the SER summarizes information presented in the Revision 18 of the SAR (Private Fuel Storage Limited Liability Company, 2000), the result of the staff's independent investigation, and staff's review of new information presented in Revision 22 of the SAR (Private Fuel Storage Limited Liability Company, 2001). A summary is included at the end of this section pertaining to the staff's evaluation of the adequacy of the PFS-proposed seismic design for the Facility.

Geological and Seismotectonic Setting

Seismicity in the Basin and Range is generally concentrated along the Wasatch Front, Sierra Nevada and a medial zone called the Central Nevada Seismic Belt (dePolo et al., 1991). Within the region surrounding the proposed site are four seismotectonic provinces: (i) the Basin and Range, (ii) Wasatch Front as part of the Intermountain Seismic Belt, (iii) the Snake River Plain, and (iv) the Colorado Plateau. Of these four seismotectonic provinces, the Wasatch Front is the only one with levels of seismic activity that could affect the proposed site (see Stamatakos et al. (1999) for a more thorough discussion of the seismotectonic provinces).

The Skull Valley site is approximately 50 miles west of the Wasatch Front. The seismotectonic setting of the proposed site was discussed (Private Fuel Storage Limited Liability Company, 2000, Appendix 2D) within the larger context of the tectonic evolution and historic seismicity of the western Cordillera. This discussion included a brief discourse of regional crustal stresses and the driving forces of the Basin and Range extension. The SAR concluded that gravitationally derived buoyancy forces drive extension (Jones et al., 1996; England and Jackson, 1989), although recent global positioning system data used to assess present strain rates across the Basin and Range seem to suggest that external forces from motion of the Pacific and Sierra Nevada tectonic plates also play a role in driving deformation (Thatcher et al., 1999). As concluded in the Revision 2 of the SAR (Private Fuel Storage Limited Liability Company, 2000, Appendix 2D), the site in Skull Valley is presently affected by active tectonic extensional strain and, therefore, will be subjected to future seismicity and deformation.

Historical Seismicity

Geomatrix Consultants, Inc. (1999a) used the earthquake catalog compiled by the University of Utah, which includes historical earthquakes from about 1850 to 1962 and instrument recorded earthquakes from the University of Utah network of 26 statewide stations from 1962 to 1996. The compiled catalog was filtered by Arabasz et al. (1989) to remove duplicates and manmade events such as quarry and mining blasts. All magnitudes were also converted by Arabasz et al. (1989) to a common magnitude scale. Foreshocks and aftershocks were removed following the methodology of Youngs et al., (1987). The largest earthquake in the catalog is the 1909 M 6.0 event. Seismicity is generally concentrated along the Wasatch Front east of the site and in the Central Nevada Belt west of the site.

Because the reporting techniques improved through time, the catalog was incomplete; small magnitude events below about M 5.0 are absent from the record until primitive instruments became available in the early 1930s. As instrumentation improved, the record of smaller and smaller earthquakes became more complete. Completeness of the catalog for different magnitude scales was assessed using the methodology recommended by Stepp (1972) and reported in Youngs et al. (1987). The maximum likelihood technique (Weichert, 1980) was used by Geomatrix Consultants, Inc. (1999a) to derive recurrence parameters.

The staff reviewed the information provided by the applicant and evaluated the applicant's analyses of historical seismicity. The staff found no evidence of historic seismicity in the vicinity of the site. The staff believes that the analyses and information in the SAR provide reasonable assurance that an adequate set of data was used in developing seismic recurrence relationships and determining the maximum earthquake potential in the hazard analyses.

Potential Seismic Sources and Their Characteristics

The seismic source characterization of the Facility was developed from examination of the available literature integrated with detailed site geological studies, including site stratigraphy, geologic mapping, cross-sectional construction, and geophysical investigations (Geomatrix Consultants, Inc.,1999a; Bay Geophysical Associates, Inc., 1999). The most important aspects for the evaluations of seismic hazards were identification and characterization of active faults derived from paleoseismic and geophysical investigations. Identification of a detailed Quaternary stratigraphy was also essential because it provided critical constraints on faulting activity. Based on detailed site investigations and review of the seismotectonic setting, Geomatrix Consultants, Inc. (1999a) identified 29 fault sources and 4 areal sources. A logic tree approach was used to combine alternative models of source geometry, activity, and seismicity to formulate the PSHA.

The staff reviewed the seismic source characterization and found it acceptable because it is thorough, complete, and conservative. Models used by the applicant for the hazard assessment were appropriate. For example, Geomatrix Consultants, Inc. (1999a) conservatively considered all faults to be planar and to extend through the thickness of the brittle crust rather than considering the possibility that the primary faults could be listric and sole into a seismic detachment above the base of the seismogenic crust. Uncertainties in other aspects of fault geometry and seismic activity were incorporated into the probabilistic assessment. Upper ranges of those parameters that describe fault geometry or seismic activity were constructed to adequately bound geologic and geophysical observations. The historic seismic record was appropriately used to develop b-values for recurrence relationships and to develop the background areal source zone.

One aspect of the staff review included the interpretations of fault geometries for newly discovered East and West faults in Skull Valley based on reflection seismic data and forward modeling of gravity data in Geomatrix Consultants, Inc. (1999a). Staff review of the alternative models shows that the Geomatrix Consultants, Inc. assessment may have led to an overly conservative hazard result. Reanalysis (Stamatakos et al., 1999) of the proprietary industry gravity data does not support the interpretation that the West fault is an independent seismic source. Rather, the staff interprets the West fault as a splay of the East fault, incapable of independently generating large magnitude earthquakes. Therefore, the staff found the probabilistic assessment provided by Geomatrix Consultants, Inc. (1999a) to be acceptable, albeit conservative because the Geomatrix Consultants, Inc. (1999a) model considers the West fault as an active seismic source.

The conservative nature of the applicant's source characterization and PSHA results presented in the SAR is evident when the results are compared to PSHA results for other sites in Utah, especially those in and around Salt Lake City. Such a comparison shows that the seismic hazard in Skull Valley was calculated by the applicant to be higher than seismic hazard assessments that have been performed for sites at, or near, Salt Lake City, despite the fact that fault sources near Salt Lake City are larger and more active than fault sources near the PFS site. For example, the results of the applicant's PSHA for Skull Valley (Geomatrix Consultants, Inc., 2001a) suggest that it is 1.5 times more likely that a ground motion of 0.5g horizontal peak ground acceleration or greater will be exceeded at the PFS site (assuming hard rock site conditions), than at Salt Lake City, based on the USGS National Earthquake Hazard Reduction Program (Frankel et al., 1997). Similarly, the 2000-yr horizontal peak ground acceleration for Skull Valley (soil hazard) as estimated by the applicant, is higher than the 2500-yr ground motions for the nine sites along the Wasatch Front that were evaluated as part of the Utah Department of Transportation I-15 Reconstruction Project (Dames & Moore, Inc., 1996). The ground motions estimated by the applicant in Skull Valley are higher than those for the I-15 corridor, despite the close proximity of Salt Lake City to the Wasatch fault, which has a slip rate nearly ten times larger than the Stansbury or East Faults (cf., Martinez et al., 1998; Geomatrix Consultants, Inc., 1999a) and is capable of producing significantly larger magnitude earthquakes than the faults near the PFS Facility site in Skull Valley (cf., Machette et al., 1991; Geomatrix Consultants, Inc., 1999a).

Slip Tendency

Another aspect of the seismic source characterization that appears to be conservative, is the site-to-source models used in the ground motion attenuation relationships and the development of distributions of maximum earthquake magnitude based on the dimensions of fault rupture. This conclusion of additional conservatism is derived from a slip tendency analysis of the Skull Valley fault systems performed by the staff.

A slip tendency analysis (Morris et al., 1996) was completed using an interactive stress analysis program (3DStress™) that assesses potential fault activity relative to crustal stress. For Skull Valley, the stress tensor is defined with a vertical maximum principal stress (σ_1), a horizontal intermediate principal stress (σ_2) with azimuth of 355°, and a horizontal minimum principal stress (σ_3) with an azimuth of 085°. The stress magnitude ratios are $\sigma_1/\sigma_3 = 3.50$ and $\sigma_1/\sigma_2 = 1.56$. This orientation for the principal stresses was based on recent global positioning satellite information (Martinez, et al., 1998a). The slip tendency analysis assumed a normal-faulting regime, with rock density equal to 2.7 g/cc, fault dip equal to 60°, water table at a depth of 40 m, and a hydrostatic fluid pressure gradient.

In slip tendency analysis, the underlying assumption is that the regional stress state controls slip tendency and that there are no significant deviations due to local perturbations of the stress conditions. This assumption is supported by a similar slip tendency analysis of the Wasatch fault, which shows highest slip tendency values for the segments of the fault considered to be most active (Machette et al., 1991).

The slip tendency analysis shows that segments of the East fault and the East Cedar Mountain fault nearest the PFS site have relatively low slip tendency values compared to segments farther north in Skull Valley. As discussed in the following sections on site-to-source distances and maximum magnitudes, these results indicate that the seismic source characterization of the PSHA study conducted by Geomatrix Consultants, Inc. (1999a, and 2001a) is conservative. Three areas of conservatism are the distribution of site-to-source distance, maximum magnitude earthquakes, and potential of the West fault as a seismogenic source (discussed in Stamatakos et al., 1999).

Distributions of Site-to-Source Distances

Results of the slip tendency analysis indicate that fault segments with approximately North-South strikes (azimuth = 175°) are optimally oriented for future fault slip. Faults with north northeast-south southwest strikes have high slip tendency values. In contrast, fault segments with northwest-southeast strikes, such as the East fault near the PFS Facility site and the southern segments of the East Cedar Mountain fault also near the PFS Facility site, have relatively low slip tendency values. Therefore, these fault segments are less likely to slip in the future than fault segments further from the site. Fault rupture close to the site greatly influence the seismic hazard. The closer the earthquake is to the site, the larger the resulting ground motions compared to an equal magnitude earthquake on a fault segment farther away from the site.

In the site-to-source distributions used in the ground motion attenuation equations, Geomatrix Consultants, Inc. (1999a) assumed uniform distributions of earthquake ruptures along active fault segments. Given the slip tendency analysis described above, this assumption by Geomatrix Consultants, Inc. (1999a) is conservative. The staff concludes that seismic source models that incorporate slip tendency would result in a lower ground motion hazard than the one developed by the applicant.

Maximum Magnitude

The slip tendency results suggest that Geomatrix Consultants, Inc. (1999a) may have overestimated the maximum magnitude of the East and East Cedar Mountain faults near the PSFS site. In the SAR, the applicant first developed conceptual models of the physical dimensions of fault rupture—either rupture area or trace length of surface fault rupture—based on the geologic record (Geomatrix Consultants, Inc.,1999a). Second, the applicant developed distributions of maximum magnitudes for each active fault using empirical scaling relationships developed from the magnitudes and associated rupture dimensions of historical earthquakes (e.g., Wells and Coppersmith,1994). In developing the fault segment models, the applicant conservatively assumed that the entire mapped length of the surface trace length represents active fault segments. Thus, these maximum fault dimensions produce conservative estimates of maximum magnitude.

The slip tendency analysis indicates that parts of the East and East Cedar Mountain faults near the PFS Facility site have relatively low slip tendency values. Thus, these faults may be smaller

than in the fault models used by the applicant to estimate maximum magnitude. Fault rupture models developed using slip tendency analysis would therefore lead to fault segment models with smaller rupture dimensions (length or area) than those used by Geomatrix Consultants, Inc. (1999a). Because distributions of maximum magnitude for each active fault are derived from empirical scaling relationships of rupture area or rupture length (e.g., Wells and Coppersmith,1994), application of the slip tendency analysis would thereby result in smaller predicted maximum magnitudes than those developed by the applicant. Smaller maximum magnitudes would reduce the overall ground motion hazard.

In summary, the staff found that the applicant's considerations of seismic source characteristics and associated uncertainties provide reasonable assurance that all significant sources of future seismic activity have been identified and their characteristics and associated uncertainties are adequately or conservatively described and appropriately included in the evaluation of the seismic ground motion hazard. Stamatakos et al. (1999) provides more details of PFS's seismic source characterization and the staff's independent sensitivity analyses.

Further, the staff concludes that the seismic source characterization performed by the applicant is conservative (perhaps by as much 50% or more based on a comparison to Salt Lake City PSHA results). The staff does not attempt here to explicitly quantify the degree of conservatism in the seismic source characterization. Quantitative estimates of the degree of conservatism would require the staff to essentially recalculate the PFS PSHA, which is not necessary under the NRC Standard Review Plan (1997a). Nevertheless, this qualitative assessment of potential conservatism provides additional confidence that the applicant's seismic source characterization is acceptable. Because the applicant's seismic source characterization is conservative, it provides reasonable assurance that the seismic hazard has been adequately determined and is sufficient to assess safety of the PFS Facility. Therefore, the staff concludes that the information presented in Section 2.6.1.1 of the SAR is acceptable because the basic geologic and seismic characteristics of the site and vicinity have been adequately (albeit conservatively) described in detail to allow investigation of seismic characteristics of the proposed Facility site. The staff has determined that this information is acceptable for use in other sections of the SAR to develop the design bases of the Facility, perform additional safety analysis, and demonstrate compliance with regulatory requirements in 10 CFR 72.92(a), 72.92(b), 72.102(e), and 72.122(b) with respect to this subject.

Estimate of Ground Motion Attenuation

Yucca Mountain Approach

For purposes of estimating earthquake ground motions that may occur at the proposed site, the applicant utilized results of the PSHA conducted for the proposed high-level waste repository site at Yucca Mountain (Civilian Radioactive Waste Management System Management and Operating Contractor, 1998). The Yucca Mountain study developed and implemented a methodology for evaluating earthquake ground motions in the Basin and Range that includes the results of scientific evaluations and expert elicitations from seven ground motion experts. The staff found that the use of the Yucca Mountain methodology for the Facility PSHA ground motion analysis is appropriate, in general, because (i) it represents the state-of-the-art knowledge and (ii) both the PFS Facility site and site of the proposed geologic repository at Yucca Mountain have seismotectonic characteristics of the Basin and Range.

Geomatrix Consultants, Inc. (1999a) selected the published median ground motion attenuation models and weighted them according to the Yucca Mountain Seismic Hazard Study (Civilian Radioactive Waste Management System Management and Operating Contractor, 1998).

The Yucca Mountain PSHA used a sophisticated methodology for modeling and quantifying the epistemic uncertainty in ground motions. The Yucca Mountain analysis attempted to quantify all of the sources of uncertainty involved in the estimation of strong ground motion. As part of the Facility PSHA, Geomatrix Consultants, Inc. (1999a) elected to consider only that part of the epistemic uncertainty associated with the choice of different median ground motion models and not the uncertainty in the models themselves. As a consequence, sources of epistemic uncertainty that were quantified in the Yucca Mountain PSHA were not considered in the PFS Facility analysis. This leads to an underestimate of the total epistemic uncertainty and, therefore, an underestimate of the mean seismic hazard at the site. The staff performed sensitivity calculations and determined that the mean frequency of exceedance of ground motions changes by less than a factor of two. Therefore, the staff concludes this effect to be insignificant.

Revisions to the Ground Motion Modeling in 2001

In March 2001, Geomatrix Consultants, Inc. (2001a) published the revised probabilistic seismic hazard analysis result for the PFS Facility. The revision was motivated by the analysis of site-specific soils and velocity data obtained subsequent to the submittal of the initial PSHA results (Geomatrix Consultants, Inc., 1999a). In particular, the applicant provided additional shear wave velocity measurements of the upper 106.5 ft of strata in the soil column at the PFS Facility site in the SAR. The additional data were acquired from downhole geophysical measurements in two borings (Northland Geophysical Limited Liability Company, 2001) and 16 test pits excavated at the site. The applicant used the results to derive alternative interpretations of the shear wave velocity profiles that were used to develop site response models Calculation G(PO18)-2 of Parkyn, 2001.

The applicant provided revised dynamic properties of the soil strata above 106.5 feet in the SAR (Private Fuel Storage Limited Liability Company, 2001). Parkyn (2001) documents several changes in dynamic soil properties compared to those reported in the former revision of the SAR (Private Fuel Storage Limited Liability Company, 2000). These changes include:

(1) Small adjustment of the depths of the boundaries of several layers including the two prominent soil horizons.

(2) Incorporation of the downhole shear-wave velocity measurements from two boreholes, CTB-5(OW) and CTB-5A (Northland Geophysical. L.L.C., 2001).

(3) Alternative multi-step methodology to develop statistical models of shear wave velocity profiles from the 16 cone penetrometer tests and the CTB-5(OW) and CTB-5A borehole data.

(4) Direct measurement of shear wave velocities in the upper layers of the Tertiary Salt Lake Group strata, which lies just below the Quaternary-Tertiary unconformity.

(5) Revision of site response to include lower damping and lower levels of modulus reduction based on results of the resonant column tests leading to a more linear modulus reduction and damping relationship.

These revisions led to development of a nine-layer shear-wave soil profile used to calculate the site response. This change in the shear-wave profile and site response model led to a significant increase in estimated ground motions at the PFS site. As shown in Table 2-2, these changes significantly affect higher frequencies, but have much less effect on lower frequencies of ground motion (Appendix F of Geomatrix Consultants, Inc., 2001a).

Based on the new site velocity data, Geomatrix Consultants, Inc. (2001a) made several revisions to its assessment of the ground motions at the PFS site. These revisions included modifications to the site velocity model, the ground motion attenuation relationships adopted from the Yucca Mountain study, and the approach used in the site response analysis. In the aggregate, these changes resulted in an increase in the ground motion hazards estimated at the PFS site. Table 2-2 compares the estimated 2000-year PSHA accelerations as estimated in Revision 18 of the SAR (Private Fuel Storage Limited Liability Company, 2000) and the updated 2000-year PSHA accelerations in Revision 22 of the SAR, for horizontal and vertical ground motions at selected periods.

Table 2-2. Comparison of PSHA for 2,000-Year Return Period Spectral Acceleration (with 5% Damping)

Period (sec)	Horizontal Ground Motion (g)		Vertical Ground Motion (g)	
	SAR Revision 22	SAR Revision 18 (former design)	SAR Revision 22	SAR Revision 18 (former design)
PGA	0.711	0.528	0.695	0.533
0.1	1.541	1.046	1.752	1.369
0.5	1.045	1.166	0.509	0.476
2.0	0.164	0.272	0.088	0.088

The process used to estimate the ground motion at the PFS site in the original PSHA (SAR Revision 18, Private Fuel Storage Limited Liability Company, 2000), as well as in the revised analysis, consisted of the following elements.

- Median Ground Motion Attenuation Models – the ground motion models used in the Yucca Mountain study (Civilian Radioactive Waste Management System Management and Operating Contractor, 1998) were adopted in the PFS analysis to define the median ground motion and the epistemic uncertainty in the median, as a function of earthquake magnitude and distance. These are empirical models derived from ground motions recorded principally in California.

- Faulting Type – an adjustment factor was used to account for differences in the type of faulting between faults in California and Skull Valley. This adjustment factor was used to scale the California median ground motion attenuation models.

- Regional Attenuation – an anelastic attenuation model was used to remove the effects of regional attenuation of seismic waves in the crust in California and to account for the regional attenuation as it would be expected to occur in Utah.

- Site-Specific Response – the effects of California surficial materials were removed and the response of the PFS soils were computed and incorporated in the analysis to model the response of the near surface geologic deposits on ground motion at the PFS site.

- Near-Source Effects – adjustment factors were used to account for the near-source effects of faulting kinematics on ground motions at the PFS site. While the elements of the process of developing site-specific ground motion estimates for the PFS site were the same in the original PSHA and in the revised analysis, there are differences in the implementation of two of the four elements. Table 2-3 tabulates the elements of the ground motion model and how they were implemented in Revisions 18 and 22 of the SAR.

The revised PSHA used the same adjustment factors for the effects of faulting type, regional attenuation, and near-source effects. However, the median ground motion attenuation models and the evaluation of site response changed in the revision of the PSHA.

In the original PSHA, Geomatrix Consultants, Inc. (1999a) used a set of California empirical ground motion models applicable to soil sites. These models were the companion empirical models to the rock attenuation models selected by the Yucca Mountain study experts. The choice to use soil ground motion attenuation models as a starting point was based on the original observation that the PFS velocity profile compared favorably with California soil sites (Geomatrix Consultants, Inc., Appendix F, 1999a). Following revision of the soil profile data, Geomatrix Consultants, Inc. (2001a) concluded that the PFS velocity profiles now compared more favorably with California rock sites rather than California soil sites. On this basis, the California empirical rock ground motion attenuation models selected by the Yucca Mountain study experts were chosen. A total of 20 rock horizontal attenuation models with associated probability weights were used in the PFS analysis. For the vertical motions, 11 models were used. The model weights were derived from the weights assigned by the ground motion experts that participated in the Yucca Mountain study. Based on a review of the current site data, the staff agrees that the PFS site conditions compare more favorably with the California rock site conditions. Further, the staff notes that the process used in the Yucca Mountain study and in the PFS analysis is designed to remove the California regional and site-specific effects that are inherent in empirical ground motion attenuation models and to incorporate appropriate regional and site-specific effects for the site in question (in this case, Utah and Skull Valley).

By virtue of this modeling approach, the issue as to whether rock or soil median ground motion attenuation models should be used is not significant. The staff agrees, however, based on the current PFS site-specific information, that the use of the empirical rock attenuation models for the PFS site is reasonable.

Table 2-3. Comparison of Ground Motion Modeling and Soil Velocity Profiles

			SAR Revision 22	SAR Revision 18 (former design)
Median Ground Motion Attenuation Model			California rock models	California soil models
Faulting-Type Effect (Strike-slip to normal faulting)			Yucca Mountain scaling factors (re-normalized weights for rock models)	Yucca Mountain scaling factors (re-normalized weights for rock models)
Regional Attenuation (Crustal Path Effect) (California motion to Utah motion)			Yucca Mountain technique	Yucca Mountain technique
Near-Source Effects			Conservative application of Sommerville et al. (1997) factors	Conservative application of Sommerville et al. (1997) factors
Site Effect	Empirical Approach		New	-
	Modeling Approach	Input Motion	Rock recordings	Rock recordings
		Soil Velocity Profile	New 9-layer model	3-layer average velocity model 1-layer average velocity model
		Deconvolution	To a depth of 5 km	To a depth of 3 km
		Response Analyses	PFS multilayer profiles Western US generic rock profiles	PFS average profiles Westurn US generic soil profiles

Site Response Effects

A final step in the assessment of site-specific ground motions for the PFS site requires that the response of near-surface geologic deposits be considered. The effects of site response are included in the estimates of ground motion by means of frequency (or period) dependent site-response factors. In the revision of the PFS PSHA, two approaches were used to derive the site-adjustment factors. The first approach is empirical and the second is based on site-response calculations for the PFS site soils. Geomatrix Consultants, Inc. (2001a, b) assigned probability weights to each approach, based on their interpretation of the credibility in each method.

The empirical approach, used by Geomatrix Consultants, Inc. (2001a, b), was assigned 1/3 weight in PSHA calculation. The empirical approach is based on two assumptions: (i) the PFS site can be classified as a shallow soil site, and (ii) PFS soil velocity characteristics are similar to those of western United States shallow soil sites. In the empirical approach, a set of strong motion recordings obtained at shallow soil sites were selected. The selected ground motion

recordings were scaled to the desired ground motion levels at the PFS site. A set of empirical site response factors was determined from the distribution of spectral ratios that were determined from the set of shallow site recordings and the selected empirical hard-rock ground-motion models.

The second approach used by Geomatrix Consultants, Inc. (2001a) in the revision to the PSHA involved the calculation of site response factors using the SHAKE model and the PFS site data. The same approach was used in the original analysis. Based on the results of the site soils and velocity data obtained subsequent to the original submission of the PSHA, significant modifications were made to the site model. Geomatrix Consultants, Inc. (2001a, b, c) abandoned the 3-layer average velocity model used in the original study (Geomatrix Consultants, Inc., 1999a) and developed a new 9-layer soil velocity model above the Tertiary strata. The Tertiary strata in Skull Valley are part of the Salt Lake group, which is a ~500-700 ft thick sequence of semi-consolidated siltstones, claystones, and sandstones of Middle to Late Miocene Age (5.3 to 16.6 Ma). In the 2001 revisions, Geomatrix Consultants, Inc, (2001a) used both a constant velocity model and an increasing velocity model for these Tertiary strata, whereas a 1-layer average velocity model was used in the Geomatrix Consultants, Inc. (1999a) study.

The differences between the results of the empirical and site response analyses are considerable for periods less than about 0.3 s (see Fig. F-17 in Appendix F, Geomatrix Consultants, Inc., 2001a). At these periods, the site response analysis predicts higher scaling factors. However, at periods greater than about 1.0 s, the empirical factors are higher. In its revised PSHA report (Appendix F, Geomatrix Consultants, Inc., 2001a) Geomatrix Consultants, Inc. also concludes that the use of the empirical site response scaling factors is appropriate because they are based on actual strong motion recordings at shallow soil sites. At the same time, they recognize these factors are not site-specific and thus assign a lower weight to this approach.

Staff Review of Ground Motion Attenuation Models

The staff reviewed the characterization of strong ground motion in the Facility seismic hazard analysis and the approach taken to model the epistemic uncertainty, and found them acceptable. The approach to modeling strong ground motion provides reasonable assurance that the site hazard is adequately (albeit conservatively) estimated.

The staff agrees with the applicant that revision of the dynamic soil properties presented in the SAR was necessary because of the acquisition of new velocity data (Northland Geophysical, L.L.C, 2001), which was collected by the applicant after publication of the original SER. The revision of the original 3-layer shear-wave velocity profile to the current 9-layer model led to a large increase in the peak ground accelerations. However, the revised data are well within the uncertainty bands provided in the original 3-layer model. The staff considers the overall shear wave profile results, as revised, to be acceptable and conservative. In this regard, the staff notes that incorporation of the new shear-wave velocity data from the boreholes (Northland Geophysical Limited Liability Company, 2001) into the existing shear wave velocity profiles (Private Fuel Storage Limited Liability Company, 2000) or equal weighting of the original and new statistical methodologies would lead to a site response model with lower ground motions than the model presented in Revision 22 of the SAR.

The staff also agrees with the applicant's approach to estimate regional and site-specific ground motions based on the site response calculations for PFS Soils. There are sufficient

technical bases for ground motion modeling based on this approach for use in development of the site specific PSHA and ultimately in development of the design basis earthquake. In contrast, the staff finds that PFS did not provide sufficient technical basis for use of empirical site response factors. These factors are based on strong-motion recordings obtained at California sites for which no information is provided that supports a comparison to the PFS site, other than a general shallow soil site characterization. However, sensitivity results provided by PFS (Parkyn, 2001) show that inclusion of the empirical site response factors approach has a small effect on the PGA values (~12%), and an even smaller effect on the predicted ground motions at lower frequencies. The small increase in ground motions that would occur if the applicant did not use the empirical site response approach is more than compensated for by other conservatisms in the PSHA results, including the noted conservatism in the seismic source characterization.

In summary, the staff concludes that there is sufficient information on shear wave velocity profiles in the soil strata and ground motion attenuation modeling for use in other sections of the SAR to develop the design bases of the proposed Facility, perform additional safety analysis, and demonstrate compliance with the regulatory requirements of 10 CFR 72.90(b-d), 72.92(a-c), 72.98(b), 72.98(c)(3), and 72.122(b) with respect to this issue.

Probabilistic Seismic Ground Motion Hazard

The Geomatrix Consultants, Inc. (1999a) PSHA uses a well-established methodology and basic equations (e.g., Cornell, 1968, 1971; McGuire, 1976, 1978; Civilian Radioactive Waste Management System Management and Operating Contractor, 1998). Calculation of probabilistic seismic ground motion hazard requires specification of three basic inputs: (i) geometric characteristics of potential sources, (ii) earthquake recurrence characteristics for each potential source, and (iii) ground motion attenuation estimates. Details of these inputs to the PSHA at Skull Valley have been evaluated in Stamatakos et al. (1999) and summarized in previous sections of this SER. PSHA calculations include the seismic hazard from each individual source and the total hazard from all potential sources. Such calculations establish hazard curves that depict the relationship between levels of ground motion and probabilities (frequencies) at which the levels of ground motion are exceeded. In Geomatrix Consultants, Inc. (1999a) computations, fault sources were modeled as segmented planar surfaces. Areal sources were modeled as a set of closely spaced parallel fault planes occupying the source regions. The distance density functions were computed assuming that a rectangular rupture area for a given size earthquake is uniformly distributed along the length of the fault plane and located at a random point on the fault plane. Depth distribution for earthquakes was based on depth distribution of recorded historical earthquakes along the Wasatch Front. The rupture size (mean rupture area) of an event was estimated based on the empirical relation of Wells and Coppersmith (1994). The basis for using the mean rupture area is the study of Bender (1984) that shows nearly equal hazard results using the mean estimates of rupture size and considering statistical uncertainty in rupture size. The minimum earthquake magnitude considered in the Geomatrix PSHA was M 5 (Geomatrix Consultants, Inc., 1999a).

Mean and percentile (95, 85, 50, 15, and 5th) peak ground motion and 1-Hz spectral (5-percent damped) acceleration hazard curves were calculated and presented in Geomatrix Consultants, Inc. (1999a) for horizontal and vertical motions. In Revision 18 of the SAR (Private Fuel Storage Limited Liability Company, 2000), the mean peak horizontal accelerations were 0.40g and 0.53g and the mean peak vertical accelerations were 0.39g and 0.53g for 1,000- and 2,000-year return periods, respectively. Equal-hazard response spectra for return periods of 1,000 and 2,000 year (mean annual probabilities of exceedance of 1×10^{-3} and 5×10^{-4},

respectively) were calculated and presented in Geomatrix Consultants, Inc. (1999c). In Revision 22 of the SAR (Private Fuel Storage Limited Liability Company, 2001), mean peak horizontal and vertical accelerations for the 2000-yr return period were calculated to be 0.711g and 0.695g, respectively.

Contributions of individual seismic sources were calculated and the results show that the dominating sources are the Stansbury, East-Springline, and East Cedar Mountain faults for peak ground acceleration for return periods greater than 1,000 years and for 1-Hz spectral acceleration for a return period greater than 2,000 years. Deaggregation results show that the total hazard is dominated by ground motions from nearby M 6 to 7 events. Sensitivity results indicate that the choice of attenuation relationship is a major contributor to uncertainty in the hazard calculation. Geomatrix Consultants, Inc. (1999a) sensitivity results also indicate (i) alternative models for the geometry and extent of the West fault have little effect on the total hazard because the East fault dominates the hazard from the Skull Valley faults as a result of its higher estimated slip rate, and the alternative models for the West fault have only minor effects on the parameters of the East fault, (ii) the West fault, considered as an independent source or as a secondary feature, has a minimal influence on the hazard, and (iii) the East and Springline faults, combined as a single source, produces slightly higher hazard at low probabilities of exceedance and for longer period motions than separating them as individual fault sources. The Geomatrix Consultants, Inc. (1999a) summary of contributions to the uncertainty in the total hazard at the proposed Skull Valley site for a return period of 2,000 years shows that the major contributors to the total uncertainty in the hazard are the selection of attenuation relationships, assessment of maximum magnitude, recurrence rate, and magnitude distribution.

Deterministic Seismic Ground Motion Hazard

Site-specific deterministic ground motion hazard for the Facility was assessed by Geomatrix Consultants, Inc. (1997), in which two potentially capable fault sources were identified to be within 7 miles of the site—the East Cedar Mountain and Stansbury faults. Their closest distances to the site were estimated to be about 6 miles to the Stansbury fault and 5.5 miles to the East Cedar Mountain fault. The potential for a random nearby earthquake was considered by including an areal source within 16 miles of the site. Maximum earthquake magnitudes for the two fault sources were estimated using empirical relationships of Wells and Coppersmith (1994) and Anderson et al. (1996) based on estimated maximum rupture dimensions (rupture length and rupture area). The resulting mean estimates of maximum magnitudes are M 7.0 for the Stansbury fault and M 6.8 for the East Cedar Mountain fault. The maximum magnitude for the areal source was estimated to range from M 5.5 to 6.5, with a mean value of 6, based on the Wells and Coppersmith (1993) study on the relationship between earthquake magnitude and the occurrence of associated surface faulting and the assumption that these random earthquakes do not produce significant surface faulting. A mixture of attenuation relationships for strike-slip faults in California and for extensional stress regimes were used to account for uncertainties. These include Abrahamson and Silva (1997), Campbell (1997), Sadigh et al. (1993, 1997), Idriss (1991), and Spudich et al. (1997). In the Geomatrix DSHA, uncertainties were included for maximum magnitude, minimum source-to-site distance, and the selection of attenuation relationships. The recommended 84th-percentile peak ground accelerations were calculated to be 0.67g in the horizontal direction and 0.69g in the vertical direction. These accelerations envelop the calculated accelerations for a rock site and a deep soil site.

The Geomatrix Consultants, Inc. (1999b) DSHA considers the two new faults (i.e., the East and West faults) near the proposed site and in-depth characterization of other capable faults.

The detailed characteristics of the two new faults as well as other fault sources are reviewed in Stamatakos et al. (1999). In its updated DSHA, Geomatrix Consultants, Inc. (1999b) considered four nearby fault sources—the Stansbury, East, West, and East Cedar Mountains faults. The mean maximum magnitudes of these fault sources were estimated to be M 7.0, 6.5, 6.4, and 6.5, respectively, based on distributions for maximum magnitude of each source developed in Geomatrix Consultants, Inc. (1999a). The closest distances to the Canister Transfer Building from the surface traces of these faults were estimated to be 9, 0.9, 2.0, and 9 km, respectively. The ground motion models used in the updated DSHA were the set of 17 horizontal and 7 vertical attenuation relationships used in the PSHA (Geomatrix Consultants, Inc., 1999a). These relationships were reviewed and discussed in Stamatakos et al. (1999). The ground motion attenuation relationships were adjusted for near-source effects using the empirical model developed by Somerville et al. (1997). The updated DSHA results in 2000 showed that the ground motion from the East fault generally envelops those from the other sources. In Revision 18 of the SAR (Private Fuel Storage Limited Liability Company, 2000), the 84th-percentile peak ground accelerations for the East fault were calculated to be 0.72g in the horizontal direction and 0.80g in the vertical direction. When compared with the PSHA results in Revision 18 of the SAR, the controlling deterministic spectra generally were between the 5,000- and 10,000-year return period equal-hazard response spectra. In revision 22 of the SAR (Private Fuel Storage Limited Liability Company, 2001), the 84th percentile peak ground accelerations for the East fault were calculated to be 1.15g in the horizontal direction and 1.17g in the vertical direction (Geomatrix Consultants Inc., 2001d). As in revision 18 of the SAR (Private Fuel Storage Limited Liability Company, 2000), the revised controlling deterministic spectra (Geomatrix Consultants Inc., 2001d) in revision 22 of the SAR generally fall between the 5,000-yr and 10,000-yr return period equal-hazard response spectra.

Design-Basis Ground Motion

The design ground motion response spectra for the proposed Skull Valley site were developed by Geomatrix Consultants, Inc. (2001b) based on its site-specific PSHA results as reviewed in this SER and Stamatakos et al. (1999) and documented in detail in Geomatrix Consultants, Inc. (1999a, 2001a). The Geomatrix Consultants, Inc. development of design spectra is based on the procedures outlined in Regulatory Guide 1.165 (Nuclear Regulatory Commission, 1997c) and incorporates near-source effects.

The assessment of design ground motions for the Facility is described in Geomatrix Consultants, Inc. (2001b). The design ground motions were determined using the procedure described in Regulatory Guide 1.165 (Nuclear Regulatory Commission, 1997c). However, prior to implementing the Regulatory Guide 1.165 procedure, the site seismic hazard results were modified to account for the near-source effects of rupture directivity and the polarization of ground motions. Adjustments to the PSHA results that account for these effects were made using empirical models developed by Somerville et al. (1997). Based on its review, the staff determined that the deterministic approach of shifting the seismic hazard results to account for rupture directivity and ground motion directional effects is conservative for the frequencies to which these adjustments were applied. Based on the results of Somerville et al. (1997), adjustments were not made for the peak ground acceleration seismic hazard results or for spectral accelerations greater than 1.0 Hz. There is empirical evidence that suggests peak ground accelerations and high frequency ground motions may also be influenced by rupture directivity and source radiation. In addition, there is limited empirical evidence to verify the Somerville et al. (1997) model and to predict, in an absolute sense, the systematic effect of rupture directivity on strong ground motion. However, as discussed in Stamatakos et al. (1999) and Geomatrix Consultants, Inc. (1999c), the random effects of rupture directivity are

accounted for as part of the aleatory variability in ground motion. Therefore, it is an effect that is accounted for in the PSHA. In fact, for frequencies less than 1.0 Hz, these effects are double counted in the Facility estimate of design motions.

The Regulatory Guide 1.165 process for determining design basis ground motion spectra involves computing the contributions to the total hazard at the specified design return period (or reference probability) from events in discrete magnitude and distance bins. In the Geomatrix Consultants, Inc. (1999c) calculation, a magnitude bin size of 0.25 was selected. The distance bin size increases gradually from 3 to 32 miles as the source-to-site distance increases from 0 to 150 km. From these contributions and the average magnitude and distance for each bin, a weighted average magnitude, $\overline{M}$, and log average distance, $\overline{D}$, of the events contributing to the design level hazard were determined for spectral frequency ranges of 5–10 Hz and 1–2.5 Hz. Free-field ground surface response spectral shapes were developed using the 84th-percentile peak acceleration and the 84th-percentile response spectra for each of the $\overline{M}$ and $\overline{D}$ pairs using a weighted combination of the same ground motion attenuation relationships used for the PSHA (Geomatrix Consultants, Inc., 1999a). These response spectral shapes were scaled to the appropriate equal hazard spectra. Design ground motion response spectra were defined to be the envelope of the scaled spectra and equal hazard spectra. This envelope was further scaled by the adjustment factors for near-fault effect as described in Stamatakos et al. (1999). The final response spectra can be found in Geomatrix Consultants, Inc. (2001b). In Revision 18 of the SAR (Private Fuel Storage Limited Liability Company, 2000),these studies resulted in the following design ground motion accelerations: (1) for a 1,000-year return period earthquake, a peak horizontal acceleration of 0.40 g and a peak vertical acceleration of 0.39 g; and (2) for a 2,000-year return period earthquake, a peak horizontal acceleration of 0.53 g and a peak vertical acceleration of 0.53 g for a 2,000-year return period. In Revision 22 of the SAR (Private Fuel Storage Limited Liability Company, 2001), mean peak horizontal and vertical accelerations for the 2000-yr return period were calculated to be 0.711 g and 0.695 g respectively.

The applicant's exemption request specified a 1,000-year return period to calculate design basis ground motions with the PSHA methodology. The applicant (Parkyn, 1999b) stated (i) a 1,000-year return period is the same as that selected by the U.S. Department of Energy (1997) for preclosure seismic design of important to safety structures, systems, and components for NRC Frequency Category 1 design basis events at the proposed Yucca Mountain high-level waste geologic repository, and (ii) the consequences of a major seismic event at the Facility can be bounded using the HI-STORM 100 system technology and are limited to a storage cask-tipover event, which would result in a dose below regulatory limits. A Frequency Category 1 design basis ground motion refers to a mean recurrence interval of 1,000 years and a Frequency Category 2 design basis ground motion refers to a mean recurrence interval of 10,000 years. As discussed below, the staff has determined that a 2000-year return period is the appropriate value for the PFS Facility site. .

Staff Review of Ground Vibration and Request for Exemption to 10 CFR 72.102(f)(1)

The staff found the applicant's seismic hazard results to be conservative, based on the review of geological and seismotectonic setting, historical seismicity, potential seismic sources and its characteristics, estimate of ground attenuation, estimates of probabilistic and deterministic ground motion hazards, development of design basis ground motion, and independent staff analyses. The staff also found that in the application:

- Seismic events that could potentially affect the site were identified and the potential effects on safety and design were adequately assessed.
- Records of the occurrence and severity of historical and paleoseismic earthquakes were collected for the region and evaluated for reliability, accuracy, and completeness.
- Appropriate methods were adopted for evaluations of the design basis vibratory ground motion from earthquakes based on site characteristics and current state of knowledge.
- Seismicity was evaluated by techniques of 10 CFR Part 100, Appendix A. Seismic hazard, however, was evaluated using a probabilistic approach as stated in the Request for an Exemption to 10 CFR 72.102(f)(1).
- Liquefaction potential or other soil instability from vibratory ground motions was appropriately evaluated.
- The design earthquake has a value for the horizontal ground motion greater than 0.10g with the appropriate response spectrum.
- The applicant's considerations with respect to the approach taken to model the epistemic uncertainty in ground motions and near-source effects are adequate.
- As discussed in Stamatakos et al. (1999), the applicant adequately applied adjustment factors for the near-fault effect using the state-of-the-art techniques and applied procedures described in Regulatory Guide 1.165 (Nuclear Regulatory Commission, 1997c) for developing design-basis ground motion. The associated response spectra and design basis motion levels are adequate.

The staff reviewed the applicant's exemption request to use the PSHA methodology with a 1,000-year return period value by evaluating the technical basis of the PSHA methodology and its use in other Title 10 regulations regarding nuclear facilities and materials. Although 10 CFR Part 72 requires a deterministic approach for the seismic design of an ISFSI site west of the Rocky Mountain Front, a probabilistic approach for seismic design is acceptable by the 1997 amendments to 10 CFR Parts 50 and 100 that apply to new nuclear power plants, and 10 CFR Part 60 that applies to the disposal of high-level waste in geologic repositories. Also, the NRC issued Regulatory Guide 1.165 to provide guidance on PSHA methodology (Nuclear Regulatory Commission, 1997c). In addition, NRC has reviewed and approved the Request for Exemption to 10 CFR 72.102(f)(1) seismic design requirements to allow seismic design using PSHA results of 2,000-year return period earthquakes for the Three Mile Island Unit-2 (TMI-2) ISFSI (Nuclear Regulatory Commission, 1998b; Chen and Chowdhury, 1998). DSHA considers only the most significant earthquake sources and events with a fixed site-to-source distance. PSHA, on the other hand, considers contributions from all potential seismic sources and integrates across a range of source-to-site distances and magnitudes. Furthermore, DSHA is a time-independent statement, whereas PSHA estimates the likelihood of earthquake ground motion occurring at the location of interest within the time frame of interest. The staff concludes that there are sufficient regulatory and technical bases to accept the PSHA methodology for seismic design of the Facility.

The design basis ground motion for a particular structure, system, and component depends on the importance of that particular structure, system, and component to safety. As described in the NRC rulemaking plan for 10 CFR Part 72 (Nuclear Regulatory Commission, 1998a), an individual structure, system, and component may be designed to withstand only Frequency Category 1 events (1,000-year return period) if the applicant's analysis provides reasonable assurance that the failure of the structure, system, and component will not cause the Facility to exceed the radiological requirements of 10 CFR 72.104(a). If the applicant's analysis cannot support this conclusion, then the designated structures, systems, and component should have a higher importance to safety, and the structures, systems, and component should be designed such that the Facility can withstand Frequency Category 2 events (10,000-year return period).

The staff reviewed the applicant's request and supporting analysis to use the 1,000-year return period value and does not find this value acceptable because of the following reasons: (i) the DOE classification of Yucca Mountain proposed high-level waste geologic repository structures, systems, and components to design for Frequency Category 1 and Frequency Category 2 events as it applies to the proposed Yucca Mountain repository has not been reviewed or accepted by the NRC staff; (ii) the applicant has provided no technical basis for classifying all the important to safety structures, systems, and components for the Facility as those that could be designed for NRC Frequency Category 1 design basis events; and (iii) the consequence analysis using the HI-STORM 100 systems technologies includes only a single accident scenario (i.e., cask tipover) that is independent of ground motion level. The applicant did not demonstrate that the cask-tipover event envelops other unanalyzed conditions such as the effect of collapse of the Canister Transfer Building on canisters or the effects of sliding and bearing failures of the foundation and concrete pad on storage casks.

However, the staff has determined that a 2,000-year return value with the PSHA methodology can be acceptable for the following reasons:

- The radiological hazard posed by a dry cask storage facility is inherently lower and the Facility is less vulnerable to earthquake-induced accidents than operating commercial nuclear power plants (Hossain et al., 1997). In its Statement of Consideration accompanying the rulemaking for 10 CFR Part 72, the NRC recognized the reduced radiological hazard associated with dry cask storage facilities and stated that the seismic design basis ground motions for these facilities need not be as high as for commercial nuclear power plants (45 FR 74697, 11/12/80; SECY-98-071; SECY-98-126).

- Seismic design for commercial nuclear power plants is based on a determination of the Safe Shutdown Earthquake ground motion. This ground motion is determined with respect to a reference probability level of 10^{-5} (median annual probability of exceedance) as estimated in a probabilistic seismic hazard analysis (Reference Reg Guide 1.165). The reference probability, which is defined in terms of the median probability of exceedance, corresponds to a mean annual probability of exceedance of 10^{-4} (Murphy et al., 1997). That is, the same design ground motion (which has a median reference probability of 10^{-5}) has a mean annual probability of exceedance of 10^{-4} . Further, analyses of nuclear power plants in the western United States show that the estimated average mean annual probability of exceeding the safe shutdown earthquake is 2.0×10^{-4} (U.S. Department of Energy, 1997).

- On the basis of the foregoing, the mean annual probability of exceedance for the PFS Facility may be defined as greater than 10^{-4} per year.

- The DOE standard, DOE-TD-1020-94 (U.S. Department of Energy, 1996), defines four performance categories for structures, systems, and components important to safety. The DOE standard requires that performance Category-3 facilities be designed for the ground motion that has a mean recurrence interval of 2000 yrs (equal to a mean annual probability of exceedance of 5×10^{-4}). Category-3 facilities in the DOE standard have a potential accident consequence similar to a dry spent fuel storage facility.

- The NRC has accepted a design seismic value that envelopes the 2000-yr return period probabilistic ground motion value for the TMI-2 ISFSI license (Nuclear Regulatory Commission, 1998b; Chen and Chowdhury, 1998). The TMI-2 ISFSI was designed to store spent nuclear fuel in dry storage casks similar to the PFS Facility.

In summary, the staff agrees that the use of the PSHA methodology is acceptable. A 2,000-year return period is acceptable for the seismic design of the PFS Facility. As discussed in the subsequent chapters of this SER, the design analyses use a spectrum that envelops the 2,000-year return period uniform hazard spectra.

Additional Information on the East Great Salt Lake Fault

The staff reviewed additional information and analyses provided in Appendix 2G of the SAR (Private Fuel Storage Limited Liability Company, 2000) regarding reported fault characterization data for the East Great Salt Lake fault. Recent high-resolution seismic data collected from the Great Salt Lake and reported in Dinter and Pechmann (1999a,b) indicate a Holocene vertical slip rate for the East Great Salt Lake fault of 1 mm/yr (average recurrence period of 3000–6000 years). The applicant assessed the possibility of the East Great Salt Lake fault being linked with the Oquirrh fault and also with the Topliff-Hill and Mercur faults, which collectively could form a Wasatch-scale fault zone.

The applicant showed in Appendix 2G, that the information about slip in the East Great Salt Lake fault does not significantly change the existing PSHA given in Geomatrix Consultants, Inc. (1999a). The applicant reiterated that the possibility of a linked East Great Salt Lake-Oquirrh fault was already accounted for in the existing PSHA analyses. In the existing PSHA model, the mean slip rate for the East Great Salt Lake fault was 0.38 mm/yr. The data of Dinter and Pechmann (1999a,b) indicate a higher slip rate of 1 mm/yr. The applicant stated that this increase will have little effect on the PSHA because the East Great Salt Lake and the Oquirrh faults are located too far from the site to generate significant ground motion. The applicant concluded that compared to all seismic sources, the East Great Salt Lake fault contributes only a small fraction to the total hazard, including an assumption of a 1 mm/yr slip rate.

The staff agrees the applicant's analyses are acceptable. The contribution of the East Great Salt Lake fault to the PFS seismic hazard is not significant, including the possible connection with the Oquirrh fault.

Co-Seismic Rupture of Stansbury and East Faults

The staff reviewed information and analyses provided in the SAR (Appendix 2G) regarding possible co-seismic rupture of the Stansbury and East faults or East/West fault and the potential impact of co-seismic rupture on ground motion hazard at the proposed PFS Facility. The staff agrees that co-seismic rupture of the East/West faults with the Stansbury fault is not supported by historic earthquakes, nor is it supported by recent geomorphic or geologic observations. Consequently, co-seismic rupture of these faults during the license period are unlikely. Thus, co-seismic rupture scenario would likely be given a very low weight in fault tree analysis and its contribution to the total hazard would be negligible.

The applicant estimated the potential effect of co-seismic rupture of the Stansbury and East faults on ground motion hazard at the proposed Facility based on scaling factors similar to those proposed for co-seismic rupture at Yucca Mountain, Nevada [developed by the expert elicitation for the Yucca Mountain PSHA (Civilian Radioactive Waste Management System Management and Operating Contractor, 1998)]. In its assessment, the applicant stated that because both Yucca Mountain and the proposed Facility are within the same tectonic setting (extension in the basin and range), the effects of coseismic rupturing on the characteristics of ground motion attenuation is similar. The staff agreed and found using Yucca Mountain scaling factors for the Facility to be acceptable. This finding, however, is specific to the proposed Facility because it is based on specific site conditions and regulatory requirements for the proposed Facility. It is not necessarily applicable to evaluations of co-seismic rupture at other spent nuclear fuel-related facilities.

The effects of simultaneous multiple-fault ruptures on ground motions at Yucca Mountain were estimated as an increase in the median ground motion and an increase in the standard error (Civilian Radioactive Waste Management System, Management and Operating Contractor, 1998). The increase in the median ground motion is expressed as a multiple of the median. The increase in the standard error is expressed as either a multiple of the standard error or as an additional error incorporated using the square root of the sum of the squares. These scaling and additional factors for peak ground acceleration obtained by seven ground motion teams are summarized in tabular format in Appendix 2G. From this table, PFS computed the geometric means of the scale factors from all seven ground motion teams (Civilian Radioactive Waste Management System Management and Operating Contractor, 1998) for both the median ground motion and standard error and used these mean factors to estimate changes in the contributions of maximum magnitude earthquakes on Stansbury and East faults to the total hazard at the proposed PFS Facility. The calculations show that, without co-seismic rupture, a M 6.5 earthquake on East fault and a M 7.0 earthquake on Stansbury fault (the maximum expected magnitudes on these faults, respectively, Geomatrix Consultants, Inc., 1999a) have probabilities of approximately 0.35 and 0.32, respectively, of producing a peak ground acceleration in excess of 0.53g. The 0.53g is the 2000-year return period peak ground motion (Geometrix Consultants, Inc., 1999a). Considering that events of M 6.5 and larger on each fault have expected frequencies of occurrence of approximately 3×10^{-4} per year (Geomatrix Consultants, Inc., 1999a), these two earthquakes would contribute $0.35 \times (3\times10^{-4}) + 0.32 \times (3\times10^{-4}) = 2.0\times10^{-4}$ events per year to the annual frequency of exceeding 0.53 g. With co-seismic rupture of the East and Stansbury faults (i.e., assuming instead that the maximum earthquakes on the two faults occur as a single M 7.05 co-seismic rupture, M 7.05 was obtained using the combined moment for a M 6.5 and a M 7.0 earthquake), scaling the median ground motion level and the standard error produced by this earthquake by the mean factors results in a probability of approximately 0.62 of exceeding a peak ground acceleration of 0.53 g. Considering the frequency of the combined event remains to be 3×10^{-4}, the event would

contribute $0.62 \times (3 \times 10^{-4}) = 1.8 \times 10^{-4}$ event per year to the annual frequency of exceeding 0.53g. This contribution does not exceed the contribution by two independent earthquakes.

The staff concludes that a co-seismic rupture for the Stansbury and the East faults is unlikely and will not impact the existing PSHA results. Therefore, a design earthquake analyses based on the 2000-year return period ground motion is acceptable.

2.1.6.3 Surface Faulting

Geomatrix Consultants, Inc. (1999a) documented several small faults in and around the site. These faults are all considered secondary faults related to deformation of the hanging wall above the larger East and West faults. These faults are too small to be independent seismic sources but large enough to be considered in the fault displacement analysis.

Similar to the seismic hazard evaluation, Geomatrix Consultants, Inc. (1999a) developed a probabilistic fault displacement hazard. The fault displacement hazard analysis was built on two methodologies developed for the Yucca Mountain PSHA (Civilian Radioactive Waste Management System Management and Operating Contractor, 1998). These methodologies, termed the earthquake approach and displacement approach, use Basin and Range empirical relationships with site-specific data to generate fault displacement hazard curves similar to seismic hazard curves.

Probabilistic fault displacement hazard results were calculated for three potential secondary faults that are under or near the site. These faults—informally named the C, D, and F faults—were identified from detailed seismic reflection profiles and confirmed by boreholes. The seismic profiles document offset of the unconformity between Promontory soil, deposited between 130–28 Ka, and Bonneville lacustrian deposits, deposited between 28–12 Ka. Vertical separation across the largest strands of the F fault (F-1 and F-4) is approximately 5 feet in the last 60 Ka and 2 feet in the last 20 Ka. A critical observation is that these faults show evidence of repeated fault slip. This is important because it suggests that future faulting events will likely occur along these same faults and not on new faults under the site. In addition, these observations of repeated slip events allowed Geomatrix Consultants, Inc. (1999a) to constrain the average displacement per event for each fault.

Faulting recurrence rates and displacement per event were quantified based on vertical separation of the Quaternary marker horizons. The results show that based on the 95th-percentile curve, significant displacements, above 0.04 inch, are expected to occur only with an annual frequency of less than 3×10^{-4}, or once in 3,333.3 years. Significant displacements of 4 inches or more are expected to occur only with an annual frequency of less than 2×10^{-4}, or once in 5,000 years. For a 2,000-year return period (annual frequency of 5×10^{-4}), displacements due to faulting are smaller than 0.04 inch, which is less than the settlement allowance for concrete foundations.

Geomatrix Consultants, Inc. (1999a) also considered other possible distributed faulting between the mapped faults. These displacements were small. For example Geomatrix Consultants, Inc. (1999a) measured only 2 inches of cumulative displacement across 88 m of exposure in Trench T-2, with a fracture spacing between 3 and 5 feet. This suggests vertical displacement of less than 1 m accumulated across the entire width of the proposed site (approximately 5,000 feet) during the last several million years.

Based on its revisions to the soil velocity models presented in the SAR, the applicant examined whether the new shear wave velocity data or re-interpreted velocity profiles would alter existing conclusions regarding the shallow seismic surveys (Bay Geophysical Associates, Inc., 1999). The shallow seismic surveys were used in part by Geomatrix Consultants, Inc.(1999a) in its probabilistic fault displacement hazard assessment. The applicant stated (Parkyn 2001) that Bay Geophysical Associates, Inc. reviewed the Northland Geophysical, L.L.C. (2001) report and found the shear wave velocities were consistent with those used in the shallow seismic surveys. The average shear wave velocities from the borehole geophysical measurements (Northland Geophysical, L.L.C.,2001) are compatible with the values used by Bay Geophysical Associates, Inc. (1999) in processing the shear wave seismic data.

The staff reviewed the discussion and analysis and found the displacement approach is representative of site conditions, and that these results are acceptable for use in assessing the faulting hazard at the proposed site. The staff found the applicant's faulting hazard results conservative and representative of the best estimates. Using a 2,000-year return period to calculate fault displacement is appropriate and consistent with the return period for estimating seismic ground motion hazard and for seismic design. The investigations and materials presented by the applicant provide reasonable assurance that the displacements due to faulting are smaller than 0.04 inch for a 2,000-year return period (annual frequency of 5×10^{-4}), which is less than the settlement allowance for concrete foundations. Therefore, the facility is not required to be designed for a potential surface faulting hazard.

In sum, the staff reviewed the applicant's discussion on surface faulting and found it acceptable because:

- Surface geological structures at the proposed site were adequately described such that the safety of the site can be assessed and the design basis for surface faulting developed.
- Potential surface faulting that directly affects site conditions and the likely environmental impacts of activities at the site were sufficiently investigated and assessed.
- Surface faulting near or at the site will be too small to affect site safety. Therefore, no specific designs or mitigation actions with respect to surface faulting are required.
- Surface faulting will not directly influence potential consequences of a release of radioactive material during the operational lifetime of the Facility.
- No specific design is necessary for structures, systems, and components to withstand the effects of surface faulting.

This information is also acceptable for use in other sections of the SAR to develop the design bases of the Facility, perform additional safety analysis, and demonstrate compliance with the regulatory requirements of 10 CFR 72.90(b-d), 72.92(a-c), 72.98(b), 72.98(c)(3), and 72.122(b) with respect to this issue.

2.1.6.4 Stability of Subsurface Materials

The staff has reviewed information presented in Section 2.6.4, Stability of Subsurface Materials, of the SAR, which refers to the following sections of the SAR for details: 2.6.1.5, Facility Plot Plan and Geologic Investigations; 2.6.1.6, Relationship of Major Foundations to Subsurface Materials; 2.6.1.7, Excavations and Backfill; 2.6.1.11, Static and Dynamic Soil and Rock Properties at the Site; 2.6.1.12, Stability of Foundations for Structures and Embankments; and 2.6.2.1, Engineering Properties of Materials for Seismic Wave Propagation and Soil-Structure Interaction Analyses (Private Fuel Storage Limited Liability Company, 2001). The staff also reviewed information presented in Appendix 2A, Geotechnical Data Report, of the SAR (Private Fuel Storage Limited Liability Company, 2001) and other data and analyses provided by the applicant (Stone and Webster Engineering Corporation, 1998, 2001a,b,c,d; ConeTec, Inc., 1999).

Geotechnical Site Characterization

Geotechnical characterization of the site was performed through a combination of field and laboratory testing. The site investigation included 32 borings for sampling and standard penetration testing (20 in the pad emplacement area, 10 in the canister transfer building area, and 2 along the access road). The boring locations are described in Figures 2.6-2 and 2.6-18 of the SAR. Also, 39 cone penetrometer tests (CPTs) and 16 dilatometer tests were performed at locations described in Figures 2.6-18 and 2.6-19 of the SAR, Revision 13. The CPTs gave continuous profiles of tip resistance and sleeve friction, which were interpreted to obtain profiles of relative soil strength and compressibility (ConeTec, Inc., 1999). Sixteen of the CPTs included down-hole compressional and shear wave velocity measurements. The borings were used mainly for conducting standard penetration tests (SPTs). In addition, several split-spoon samples were obtained along with the SPT. The split-spoon samples were used for laboratory index testing, such as Atterberg limits and percentage of fine fraction. Undisturbed (Shelby-tube) samples were also obtained and used for laboratory triaxial, direct shear, and odometer testing to obtain strength and compressibility data. Laboratory specimens and the test results are listed in Tables 2-6 of Stone and Webster Engineering Corporation (2000a). Sixteen test pits were excavated in the proposed pad emplacement area in January, 2001 for sampling and *in-situ* examination of the near-surface soil layers (Parkyn, 2001).

The water-table depth was estimated to be approximately 125 feet below the ground surface (i.e., at about elevation 4,350 feet above mean sea level), based on data from an observation well. A depth to groundwater of about this value is also implied by P-wave velocities from a seismic refraction survey that change from about 2,780 ft/sec to about 5,525 ft/sec at a depth of 90–131 feet.

Soil classification was performed using information from three sources: (i) visual field classification of drill cuttings and split-spoon samples following ASTM D2488–93 (American Society for Testing and Materials, 1999), (ii) Atterberg limits and percentage of fine fraction from laboratory testing of split-spoon samples, and (iii) interpretation of CPT logs. Based on information from these sources, the subsurface materials at the site were classified by the applicant as consisting of a relatively compressible top layer (layer 1) that is approximately 25–30 feet thick. Layer 1 is underlain by much denser and stiffer material (layer 2) classified as dense sand and silt. The strength and stiffness of layer-2 soil, interpreted from SPT values that exceed 100, indicate that the soil is not a likely source of instability for the proposed structures. Therefore, geotechnical site investigation was focused on determining the engineering characteristics of layer-1 soil, a mixture of clayey silt, silt, and sandy silt with occasional silty

clay and silty sand. A detailed description of layer-1 soil is provided through 17 cross sections in the SAR Figures 2.6-5 (Sheets 1–14) and 2.6-21 through 2.6-23. Fourteen of the cross sections were developed along lines that cross the proposed storage-pad area and consist of six east-west lines, six north-south lines, and two diagonal lines (Figure 2.6-19 of the SAR). The other three cross sections were developed along east-west lines that cross the proposed Canister Transfer Building area (Figure 2.6-18 of the SAR). Based on these cross sections, layer-1 soil was subdivided into four sublayers, (in top-down order): layer 1A, classified as eolian silt, is typically about 3–5 feet thick; layer 1B, a silty clay/clayey silt mixture that varies in thickness from about 5 to 10 feet; layer 1C, a mixture of clayey silt, silt, and sandy silt, with thickness of about 7.5–12 feet; and layer 1D, a silty clay/clayey silt mixture with maximum thickness of about 5 feet. Information provided in the SAR indicates that the eolian silt (layer 1A soil) will be excavated, mixed with sufficient Portland cement and water, and re-compacted to form a soil-cement subgrade in the proposed pad-emplacement and canister transfer building areas.

Profiles of cone tip resistance from the CPT [Figure 2.6-5 (Sheet 1–14) of SAR; ConeTec, Inc., 1999, Appendix A] indicate that the strength of the silty clay/clayey silt layers (layers 1B and 1D) is smaller than the strength of layer 1C (clayey silt, silt, and sandy silt). The value of tip resistance in layers 1B and 1D is typically about 0.5 and 0.75, respectively, of layer 1C tip resistance. Information from the CPT tip resistance profiles, which indicate the variation of relative strength with depth, was combined with laboratory compression test results from layer-1B specimens to obtain values of undrained shear strength for layer 1B, layer 1C, and layer 1D soils.

Soil compressibility was determined using a combination of laboratory compressibility data for layer 1B soils and CPT data. Cone tip resistance profiles (from the CPT) show the relative compressibility of the soil layers, with layer 1B being the most compressible and layer 1C the least. This variation of relative compressibility indicates that values of settlement calculated using the compressibility data for layer 1B soil represent the upper bound for the entire soil profile. Settlement of the entire soil profile can also be calculated directly from the cone tip resistance values using an empirical approach developed by Schmertmann (1970, 1978). The approach is described in detail in Lunne et al. (1997).

The potential for significant additional settlement owing to collapsible soils was explored by the applicant. The occurrence of collapsible soils at the site is suggested by the high values of void ratio reported for several specimens in the SAR. Collapsible soils may undergo a relatively large decrease in volume when wetted or subjected to dynamic loading. Therefore, the occurrence of significant quantities of such soils under the foundation of a structure requires analysis on the potential for relatively high settlements if the foundation soil is wetted or subjected to dynamic loading. The following information presented in Section 2.6.1.11.4 of the SAR, demonstrates that the risk of significant additional settlement owing to soil collapse is negligible. First, results of laboratory testing on five specimens with high-void ratio (1.95–2.51) indicate the additional vertical strain that resulted from inundating the specimens with water is only about 0.001 (i.e., an additional settlement of about 0.12 inch for a 10-foot thick soil layer). Second, the top 5–7 feet soil layer at the pad emplacement area will be replaced with a low-permeability soil/cement mixture. Furthermore, the ground surface in the pad area will be graded to promote run-off toward the north. This arrangement is expected to make water influx into the pad foundation soil unlikely. Also, the pad emplacement area is at an elevation of at least 4 feet above the probable maximum flood level. Third, there is no known record of excess settlement resulting from collapsible soils occurring in the Skull Valley area. The only known occurrence of collapsible soil in Utah is in Cedar City, which is far from the site. Any

occurrence of excess settlement in the Skull Valley area would likely have been mentioned in the County Soil Report (a USDA unpublished report), which deals with the suitability of the various soil types for septic-systems construction.

The staff reviewed the geotechnical site characterization information provided in the SAR and concluded that:

- The depth and thicknesses of soil layers and the water-table depth at the site are described in sufficient detail to support engineering analyses of the proposed structures.
- The index properties and strength and compressibility of the soil layers were determined using an appropriate combination of field and laboratory testing. The information presented is sufficient to support appropriate engineering analyses of the proposed structures.
- The potential for instability resulting from possible occurrence of collapsible soils at the proposed site was investigated in sufficient detail. Results of the investigation indicate that the potential for such instability is negligible.

The staff concludes that the geotechnical site characterization information presented in the SAR is adequate for use in other sections of the SAR to develop the design bases for the Facility and perform additional safety analysis, and demonstrate compliance with regulatory requirements in 10 CFR 72.102(c, d) and 72.122(b).

Stability of Cask-Storage-Pad Foundation

The cask storage pads (each 30 ft wide, 67 ft long, and 3 ft thick) will be laid out in two clusters: a north cluster separated from a south cluster by a 90-foot wide space (Private Fuel Storage Limited Liability Company, 2001, Figure 1.2-1). Each cluster consists of 25 north-south columns of storage pads. Each pad column is separated from the adjacent column by a 35-foot wide space (in the east-west direction) and consists of ten pads arranged end-to-end in the north-south direction with a 5-foot separation between the adjacent pads. Each pad cluster, therefore, consists of 250 pads, giving a total of 500 pads in the two clusters. An east-west vertical section through a typical storage pad (Private Fuel Storage Limited Liability Company, 2001, Figure 4.2-7) indicates that the pad would be embedded in soil cement of a maximum thickness of 4 feet and 4 inches. The soil cement consists of two layers: an upper layer with the minimum unconfined compressive strength of 250 psi underlain by a lower layer with the maximum thickness of 2 ft, the minimum unconfined compressive strength of 40 psi, and the maximum elastic modulus of 75,000 psi. The base of the upper soil-cement layer is flush with the base of the pad. The soil cement is overlain by an 8-inch thick layer of compacted aggregate, the top surface of which is flush with the top surface of the pad.

Each storage pad will be loaded to a static bearing pressure of 1.87 ksf, considering the dead load plus long-term live load for a 30 feet × 67 feet × 3 feet concrete pad loaded with eight casks. The 35-foot width-wise (east-west) separation between pad columns is considered large enough that the zones of influence of the static foundation loading from adjacent pad columns can be assumed to be independent to a depth of 30 ft below the base of the pads. Potential dynamic loading of the pads was characterized by horizontal and vertical ground accelerations of 0.711g and 0.695g, respectively, calculated based on consideration of a 2,000-year return-period earthquake. The stability of the pads was evaluated with respect to the potential for

bearing-capacity failure or excessive settlement under static loading, and the potential for base sliding or bearing-capacity failure under dynamic loading. These aspects of the stability evaluation are reviewed in the following sections.

Stability Against Bearing-Capacity Failure Under Static Loading

Stability of the storage pads under static loading was determined through the allowable bearing pressure calculated using a factor of safety of 3.0. This is a standard procedure for the design of shallow foundations (e.g., Terzaghi et al., 1996). Two calculations of the allowable bearing pressure under static loading were provided (Stone and Webster Engineering Corporation, 2001b): one based on undrained analysis, using an undrained shear strength (c_u) of 2.2 ksf; and another based on drained analysis using a friction angle of 30° (with zero cohesion). The c_u value of 2.2 ksf was obtained from compression tests on specimens of layer 1B soil (silty clay/clayey silt), which, as described earlier, is the weakest soil layer, with a CPT tip resistance typically about 0.5 times the tip resistance of layer 1C soil. The friction angle of 30° is a lower bound estimate from the CPT data. Values of friction angle from the CPT data are generally greater than 35°. Therefore, either of these strength-parameter values (i.e., c_u value of 2.2 ksf, or friction angle of 30° with zero cohesion) is accepted as representing the average strength of layer 1 soil for the purpose of determining the allowable bearing pressure for the specified dimensions and embedment depth of the cask storage pad. The allowable bearing pressure was determined to be 4.36 ksf based on the undrained analysis, or 9.73 ksf based on the drained analysis (Table 2.6-6 of the SAR). Both values of allowable bearing pressure exceed the actual bearing pressure of 1.87 ksf, based on consideration of the foundation dead load plus long-term live load.

The staff reviewed the applicant's evaluation of the stability of the cask storage pads with respect to the potential for bearing-capacity failure under static loading. The evaluation was performed using appropriate techniques, foundation loading, and material properties; and an acceptable safety factor was demonstrated. Independent calculations were performed by the staff using a procedure suggested by Meyerhof (1956, 1965) to determine the SPT values (N) or CPT tip resistance values (Q_t) that are required to satisfy a safety factor of 3.0 against bearing failure under the cask-pad bearing pressure of 1.94 ksf, which bounds the bearing pressure of 1.87 ksf. The calculations gave the required values as $N = 0.9$ and $Q_t = 7.05$ ksf, which are much smaller than the measured N and Q_t values [Appendix 2A of the SAR, Revision 13, and Appendix A of ConeTec, Inc. (1999)]. Therefore, the staff concludes that the proposed cask-pad design is acceptable considering the potential for bearing-capacity failure under static loading, and the information provided in the SAR regarding the static bearing capacity of the storage pads is adequate for use in other sections of the SAR to perform additional safety analysis and demonstrate compliance with regulatory requirements in 10 CFR 72.102(c, d) and 72.122(b).

Stability Against Excessive Settlement Under Static and Dynamic Loading

The settlement of the cask storage pad under the bearing pressure of 1.94 ksf is given in the SAR as 3.3 inches, which is considered an upper bound estimate, having been calculated using laboratory compressibility data for layer 1B soil and a bearing pressure larger than the static foundation bearing pressure of 1.87 ksf. The estimated settlement of 3.3 inches can be accepted as the upper bound considering the Q_t profiles for the site (discussed under *Geotechnical Site Characterization*), which indicate that layer 1B is the most compressible soil layer. An alternative estimate of the storage-pad settlement made by PFS using the Q_t data and a procedure developed by Schmertmann (1970, 1978) gave values of settlement smaller

than 1.0 in. for the storage pads. Based on these calculations, the storage pads would be expected to undergo post construction settlement of not more than about 3 inches. The storage pads will be constructed such that their top surface is flush with the top surface of the compacted-aggregates layer.

The applicant estimated the potential settlement owing to dynamic compaction of the subsurface materials using the empirical procedure of Tokimatsu and Seed (1987) for the evaluation of settlements in sand from earthquake shaking (Stone and Webster Engineering Corporation, 2001d). The primary inputs to the analysis are the SPT blow count (corrected for the effects of overburden pressure), the earthquake magnitude, and the soil thickness that may undergo vibratory compaction. The applicant estimated a dynamic settlement of 0.15 in. using a corrected blow count of 28, earthquake magnitude of 7, and soil thickness of 15 ft. The values for blow count and soil thickness may be difficult to justify, but an examination of the calculation by the staff indicates that the settlement would increase to about 1.2 in. if the corrected blow count was decreased to 10 and the soil thickness increased to the maximum of about 30 ft for layer 1 soil.

The applicant indicated that changes caused by potential settlement of the pad would be corrected by scraping the aggregates from between the pads to maintain the top surface of the aggregates at the same elevation as the top surface of the pads (SAR p. 2.6-51; Enclosure 2 of Parkyn, 2001).

The staff reviewed the applicant's evaluation of the stability of the cask storage pads with respect to the potential for excessive settlement under static and dynamic loadings. The evaluation was performed using appropriate techniques, foundation loading, and material properties. The staff considers the analysis of the stability of the cask storage pads acceptable. In addition, the applicant committed to perform maintenance repair of the pad-emplacement area as necessary to correct any changes caused by settlement of the pad, such as by scraping aggregates from between the pads to maintain the top surface of the aggregate layer at the same elevation as the top surface of the pads. The staff concludes that the information provided in the SAR regarding potential settlement of the storage pads is adequate for use in other sections of the SAR to perform additional safety analysis and demonstrate compliance with regulatory requirements in 10 CFR 72.102(c, d) and 72.122(b).

Stability Against Sliding Under Dynamic Loading

As shown in Figure 4.2-7 of the SAR, each storage pad would be surrounded by soil cement that consists of two layers as follows: an upper layer with the minimum unconfined compressive strength of 250 psi, and a lower layer having the minimum unconfined compressive strength of 40 psi, and the maximum elastic modulus of 75000 psi. The base of the upper soil-cement layer is flush with the base of the pad. The applicant provided sliding stability analyses that rely on the shear strength of the natural soil underlying the lower layer of soil cement to resist sliding of the pads (Stone and Webster Engineering Corporation, 2001b, p. 18–35). For these analyses, the applicant assumed that a sufficient bond would develop at the interfaces between the upper and lower soil cement layers, the concrete pad and lower soil cement layer, and the lower soil cement layer and the underlying natural soil. Therefore, the applicant concluded that failure of the natural soil would be more likely than failure of any of the interfaces. The applicant has also committed to perform laboratory tests during the design of the soil cement to demonstrate that the required shear strengths can be achieved at the various interfaces, and to perform field tests during construction to demonstrate that the required shear strengths at these interfaces have been achieved (PFS - SAR, section 2.6.1.12.1).

The applicant also presented analyses to assess stability against sliding on two deep-seated failure surfaces: one at the base of the soil-cement subgrade, and another located within layer 1C soil (SAR, pp. 2.6-63 through 2.6-72). Such analyses require an examination of several potential failure surfaces and an assessment of stability using the conditions on the most critical failure surface. This approach was not followed in the analyses presented by the applicant for stability against sliding on deep-seated surfaces. The NRC staff concluded that an explicit analysis of deep-seated sliding is not necessary for the proposed facility because of the following reasons: (i) subsurface investigations conducted at the site do not indicate the occurrence of any deep-seated and relatively weak soil layer in which sliding may be localized; (ii) the assessment of stability against bearing-capacity failure (evaluated next) is based on the bearing capacity theory (e.g., Terzaghi et al., 1996, pp. 258–261), which considers sliding on a series of failure surfaces that may develop in a thick soil deposit of uniform shear strength; and (iii) the subsurface conditions at the site, (i.e., shear strength increasing with depth) satisfy the assumptions used to develop the bearing capacity theory.

The applicant also provided a set of analyses that rely on the frictional resistance of the interfaces and the passive resistance of the natural soil at the north or south boundaries of the soil-cement layers to resist sliding of the pads (Stone and Webster Engineering Corporation, 2001b, p. 36–42). Each of the interfaces was assigned a friction coefficient of 0.306 (friction angle of 17°) in the analyses. This value of friction coefficient is consistent with the values recommended in the literature for interfaces between concrete and fine-grained soils. For example, Terzaghi et al. (1996, p. 328) suggest a maximum value of about 0.364 (friction angle of 20°) for such interfaces. The values of safety factor obtained from the analyses indicate that ground motion from the design-basis earthquake could cause sliding of the pads (or pad-foundation system). The applicant determined that the magnitude of sliding displacement would not exceed about 6 inches (Stone and Webster Engineering Corporation, 2001b, p. 43–45) and stated that such sliding displacement would not constitute a safety hazard because there are no external safety-related connections to either the pads or the casks. This statement was supported by additional analyses provided by the applicant (Holtec International, 2001, Attachment 1), which also indicate that sliding of the pads would reduce the tendency for sliding or tipping over of the casks.

The staff agrees with the applicant's conclusion that sliding of the pads would not constitute a safety hazard because pad sliding tends to increase the stability of the casks (against sliding or tip over) and there are no safety-related external connections to the pads or casks that may rupture or be misaligned as a result of pad sliding. Therefore, the staff concludes that the proposed cask-pad design is acceptable considering the potential for instability resulting from sliding of the pads under dynamic loading, and the information provided in the SAR regarding potential sliding of the pads is adequate for use in other sections of the SAR to perform additional safety analysis and demonstrate compliance with regulatory requirements in 10 CFR 72.102(c, d) and 72.122(b).

Stability Against Bearing Capacity Failure Under Dynamic Loading

The assessment of stability against bearing capacity failure of the storage pads under dynamic loading was based on bearing-capacity analyses for the load cases shown in Table 2-4. In each load case, the static load (dead load plus long-term live load for a 30 feet × 67 feet × 3 feet concrete pad loaded with eight casks) was combined with dynamic-load components determined using the load factors shown in the table. The dynamic load applied in a given direction is equal to the product of the load factor and the design basis earthquake load for that direction. A negative load factor for vertical force indicates that the vertical force is applied

upward. The combinations of dynamic-load factors shown in the table satisfy NRC requirements given in Newmark and Hall (1978). The table shows values of the calculated and allowable bearing pressures for each load case. The allowable bearing pressure was determined using a factor of safety of 1.1 and a value of undrained shear strength (c_u) of 2.2 ksf. This value of c_u is the minimum for layer-1 soil and, consequently, is accepted as an average value along potential failure surfaces that may develop in this soil layer. Values of the calculated and allowable bearing pressures vary because of changes in the effective bearing area of the pads caused by the eccentricity of the resultant applied loading for each load case. The magnitude of dynamic horizontal force transmitted from the casks to the pad was calculated using a value of 0.8 for the cask-on-pad friction coefficient. As Table 2-4 shows, the calculated bearing pressure for each load case is smaller than the allowable bearing pressure.

PFS also presented stability analyses for partially loaded pads under dynamic loading. Analyses were presented for pads loaded with two or four casks (instead of the full load of eight casks) and subjected to 100 percent dynamic loading (load factor of 1.0) in every direction. The dynamic loadings were obtained from finite element analyses of a pad loaded with two or four casks and subjected to vertical and horizontal acceleration time histories representative of the design earthquake. Results of the bearing capacity analysis (Table 2.6-8 of the SAR) indicate adequate safety factors against bearing capacity failure under dynamic loading for pads loaded with two or four casks.

The staff reviewed the applicant's evaluation of the stability of the cask storage pads with respect to the potential for bearing-capacity failure under dynamic loading. The evaluation was performed using appropriate techniques, foundation loading, and material properties, and an acceptable safety factor was demonstrated. Based on the results of the analyses, the staff concludes that the proposed cask-pad design is acceptable considering the potential for bearing-capacity failure under dynamic loading, and the information provided in the SAR regarding the dynamic bearing capacity of the storage pads is adequate for use in other sections of the SAR to perform additional safety analysis and demonstrate compliance with regulatory requirements in 10 CFR 72.102(c) and 72.102(d).

Table 2-4. Results of bearing capacity analysis of storage pads under dynamic loading (from Private Fuel Storage Limited Liability Company, 2000)

Load Case	Dynamic Load Factors			Bearing Pressure (ksf)	
	North-South (Pad Long Dimension)	East-West (Pad Short Dimension)	Vertical	Allowable	Calculated
II	1.0	1.0	0.0	4.85	4.56
IIIA	0.4	0.4	-1.0	8.21	2.13
IIIB	0.4	1.0	-0.4	4.78	4.55
IIIC	1.0	0.4	-0.4	8.59	2.61
IVA	0.4	0.4	1.0	10.51	3.76
IVB	0.4	1.0	0.4	7.73	4.09
IVC	1.0	0.4	0.4	9.45	3.83

Stability of the Canister Transfer Building Foundation

The proposed Canister Transfer Building will be founded on a rectangular reinforced concrete mat 240-ft wide (east-west direction), 279.5-ft long (north-south direction), and 5-ft thick (Figure 4.7-1 of SAR). The perimeter of the foundation mat to a distance of 6.5 ft from the edge will be extended to a depth of 1.5 feet below the base of the mat to form a shear key into the underlying soil. The natural soil around the foundation will be replaced by soil cement to a depth of 5 ft below the top of the foundation mat and laterally to a distance of one mat dimension from the edge of the mat in every direction {i.e., 240 ft out from the mat in the east and west directions and 279.5 ft out from the mat in the north and south direction (Stone and Webster Engineering Corporation, 2001c; Enclosure 2 of Parkyn, 2001)}. The soil cement will have a minimum unconfined compressive strength of 250 psi.

The foundation loading was determined through a lumped-mass analysis of the Canister Transfer Building, which gave a vertical static load of 97,749 kips and dynamic load of 79,779 kips vertical, 111,108 kips north-south, and 99,997 kips east-west (Table 2.6-11 of the SAR). The dynamic loads were calculated using horizontal and vertical ground accelerations of 0.711 g and 0.695 g, respectively, which represent the 2,000-year return-period earthquake for the facility design. The lumped-mass analysis has been reviewed and accepted by the NRC staff (Chapter 5). The stability of the Canister Transfer Building foundation was evaluated with respect to the potential for bearing-capacity failure or excessive settlement under static loading, and the potential for base sliding or bearing-capacity failure under dynamic loading. These aspects of the stability evaluation are reviewed in the following sections.

Stability Against Bearing-Capacity Failure Under Static Loading

The Canister Transfer Building foundation will be loaded to a bearing pressure of 1.46 ksf, considering the vertical static load of 97,749 kips supported by a total bearing area of 240 x 279.5 ft^2 (Stone and Webster Engineering Corporation,2001c). The stability of the Canister Transfer Building foundation under static loading was determined through the allowable bearing pressure using a factor of safety of 3.0. This is a standard procedure for the design of shallow foundations (e.g., Terzaghi et al., 1996). Two calculations of the allowable bearing pressure under static loading were provided: one based on undrained analysis using an undrained shear strength (c_u) of 3.18 ksf; and another based on drained analysis using a friction angle of 30° (with zero cohesion). The c_u value of 3.18 ksf is a depth-weighted average for layer 1 soil from the base of the Canister Transfer Building foundation to a depth of 20-25 ft below the foundation. The average was calculated using the c_u value of 2.2 ksf for layer 1B soil from laboratory compression test and the variation of relative strength with depth from CPT data. The relatively stiff layer 2 soil, which lies at a depth of 20–25 feet below the Canister Transfer Building foundation, was not included in the calculation of average strength. The friction angle of 30° is a lower bound estimate from the CPT data. Values of friction angle from the CPT data are generally greater than 35°. Therefore, either of these strength-parameter values (i.e., c_u value of 3.18 ksf, or friction angle of 30° with zero cohesion) is accepted as representing the average strength of layer 1 soil for the purpose of determining the allowable bearing pressure for the Canister Transfer Building foundation. The allowable bearing pressure was determined to be 6.54 ksf based on the undrained analysis, or 56.6 ksf based on the drained analysis (Table 2.6-9 of the SAR). Both values of allowable bearing pressure exceed the actual bearing pressure of 1.46 ksf under static loading. The staff reviewed the applicant's evaluation regarding the estimated allowable bearing pressure under static loading and found it acceptable.

The staff reviewed the applicant's evaluation of the stability of the Canister Transfer Building foundation with respect to the potential for bearing-capacity failure under static loading. The evaluation was performed using appropriate techniques, foundation loading, and material properties; and an acceptable safety factor was demonstrated. Therefore, the staff concludes that the proposed design of the Canister Transfer Building foundation is acceptable considering the potential for bearing-capacity failure under static loading, and the information provided in the SAR regarding the static bearing capacity of the Canister Transfer Building foundation is adequate for use in other sections of the SAR to perform additional safety analysis and demonstrate compliance with regulatory requirements in 10 CFR 72.102(c, d) and 72.122(b).

Stability Against Excessive Settlement Under Static and Dynamic Loading

The settlement of the Canister Transfer Building foundation under the bearing pressure of 1.67 ksf is given in the SAR as 3 inches, which is considered an upper bound estimate having been calculated using laboratory compressibility data for layer 1B soil and a bearing pressure larger than the estimated static bearing pressure of 1.46 ksf for the foundation. The estimated settlement of 3 inches can be accepted as the upper bound considering the Q_t profiles for the site (discussed under *Geotechnical Site Characterization*), which indicate that layer 1B is the most compressible soil layer. As discussed under "Stability of Cask-Storage-Pad Foundation" subsection titled "Stability Against Excessive Settlement Under Static and Dynamic Loadings," the applicant's calculation indicates that a maximum settlement of about 1.2 in. can be expected from soil compaction owing to the design-basis earthquake.

The staff reviewed the applicant's evaluation of the stability of the Canister Transfer Building foundation with respect to the potential for excessive settlement under static and dynamic loadings. The evaluation was performed using appropriate techniques, foundation loading, and material properties. The staff considers this stability analysis acceptable, and concludes that the information provided in the SAR regarding potential settlement of the Canister Transfer Building foundation is adequate for use in other sections of the SAR to perform additional safety analysis and demonstrate compliance with regulatory requirements in 10 CFR 72.102(c,d) and 72.122(b).

Stability Against Sliding Under Dynamic Loading

The proposed design of the Canister Transfer Building foundation relies on two features to resist foundation sliding. First, a 1.5-foot deep perimeter key at the base of the foundation would constrain potential sliding surfaces to pass through the underlying soil, such that the strength of the soil can be relied upon to resist sliding. Second, sliding of the foundation would be resisted by the compression strength of the surrounding soil cement, such that passive resistance of the soil cement can be relied upon to contribute to the overall sliding resistance. Two assessments of the sliding stability of the foundation are provided in the SAR and in the supporting calculation package (Stone and Webster Engineering Corporation, 2001c). One assessment is based on combining the residual strength of the natural soil with the full passive resistance of the soil cement, and the other on a combination of the peak strength of the natural soil with 50 percent of the passive resistance of the soil cement. The first assessment gave a minimum safety factor of 1.26 for all the load cases examined whereas the second assessment gave a minimum safety factor of 1.15 for the same load cases. Six load cases (IIIA, IIIB, IIIC, IVA, IVB, and IVC in Table 2-4) were used in the assessment. The load case with a load factor of 1.0 for the north-south component and 0.4 for the other two components produced the minimum safety factor. The applicant varied the vertical load factors as well as the horizontal factors in its assessment of the sliding stability as shown in Table 2.6-13 of the SAR. Changing the vertical dynamic load should have no effect, however, because the potential sliding surface is horizontal and the sliding resistance used in the analyses is independent of vertical loading. Therefore, the factor of safety against sliding may vary with the horizontal load factors but not with the vertical. Information presented in the SAR indicates that the case with 100 percent north-south and 40 percent east-west dynamic loads is bounding. The factor of safety obtained for this load case is larger than the minimum acceptable value of 1.1. This assessment of the sliding stability assumes that the soil cement around the foundation has a minimum unconfined compressive strength of 250 psi in every direction and at every point within the soil cement layer.

The applicant also presented analyses to assess stability against sliding on a deep-seated failure surface localized within layer-1C soil (SAR, pp. 2.6-79 through 2.6-81). Such analyses generally require an examination of several potential failure surfaces and an assessment of stability using the conditions on the most critical failure surface. This approach was not followed in the analyses presented by the applicant for stability against sliding on a deep-seated surface. The NRC staff, concluded that an explicit analysis of deep-seated sliding is not necessary for the proposed facility because of the following reasons: (i) subsurface investigations conducted at the site do not indicate the occurrence of any deep-seated and relatively weak soil layer in which sliding may be localized; (ii) the assessment of stability against bearing-capacity failure (evaluated next) is based on the bearing capacity theory (e.g., Terzaghi et al., 1996, p. 258–261), which considers sliding on a series of failure surfaces that may develop in a thick soil deposit of uniform shear strength; and (iii) the subsurface conditions

at the site, (i.e., shear strength increasing with depth) satisfy the assumptions used to develop the bearing-capacity theory.

The staff reviewed the applicant's evaluation of the stability of the Canister Transfer Building foundation with respect to the potential for sliding under dynamic loading. The evaluation was performed using appropriate techniques, foundation loading, and material properties; and an acceptable safety factor was demonstrated. Therefore, the staff concludes that the proposed design of the Canister Transfer Building foundation is acceptable considering the potential for sliding under dynamic loading, and the information provided in the SAR regarding the sliding stability of the Canister Transfer Building foundation is adequate for use in other sections of the SAR to perform additional safety analysis and demonstrate compliance with regulatory requirements in 10 CFR 72.102(c, d) and 72.122(b).

Stability Against Bearing Capacity Failure Under Dynamic Loading

The assessment of stability against bearing capacity failure of the Canister Transfer Building foundation under dynamic loading was based on bearing-capacity analyses for the load cases shown in Table 2-7. In each load case, the vertical static load of 97,749 kips was combined with dynamic-load components using the load factors shown in the table. The dynamic force applied in a given direction is equal to the product of the load factor and the appropriate component of dynamic load (79,779kips vertical;111,108 kips north-south; and 99,997kips east-west). A negative load factor for vertical force indicates that the vertical force is applied upward. The combinations of dynamic-load factors shown in the table satisfy NRC requirements in Newmark and Hall (1978). The table shows values of the calculated and allowable bearing pressures for each load case. The allowable bearing pressure was determined using a factor of safety of 1.1 and a value of undrained shear strength (c_u) of 3.18 ksf. This value of c_u is a depth-weighted average for layer 1 soil from the base of the Canister Transfer Building foundation to a depth of 20-25 feet below the foundation. The average was calculated using the c_u value of 2.2 ksf for layer 1B soil from laboratory compression test and the variation of relative strength with depth from CPT data. The relatively stiff layer 2 soil, which lies at a depth of 20–25 ft below the Canister Transfer Building foundation, was not included in the calculation of average undrained strength. The average c_u would be larger if layer 2 soil was included in the calculation. Therefore, this value of c_u is accepted as an estimate of the average c_u value along a potential failure surface that may result from Canister Transfer Building foundation loading. Values of the calculated and allowable bearing pressures vary because of changes in the effective bearing area of the foundation caused by the eccentricity of the resultant applied loading for each load case. As Table 2-5 shows, the calculated bearing pressure for each load case is smaller than the allowable bearing pressure.

The staff reviewed the applicant's evaluation of the stability of the Canister Transfer Building foundation with respect to the potential for bearing-capacity failure under dynamic loading. The evaluation was performed using appropriate techniques, foundation loading, and material properties; and an acceptable safety factor was demonstrated. Therefore, the staff concludes that the proposed design of the Canister Transfer Building foundation is acceptable considering the potential for bearing-capacity failure under dynamic loading, and the information provided in the SAR regarding the dynamic bearing capacity of the Canister Transfer Building foundation is adequate for use in other sections of the SAR to perform additional safety analysis and demonstrate compliance with regulatory requirements in 10 CFR 72.102(c, d) and 72.122(b).

Table 2-5. Results of bearing capacity analysis of Canister Transfer Building foundation under dynamic loading (from Private Fuel Storage Limited Liability Company, 2001)

Load Case	Dynamic Load Factors			Bearing Pressure (ksf)	
	North-South	East-West	Vertical	Allowable	Calculated
II	1.0	1.0	0.0	11.97	2.39
IIIA	0.4	0.4	-1.0	12.54	0.99
IIIB	0.4	1.0	-0.4	12.82	1.70
IIIC	1.0	0.4	-0.4	13.67	1.65
IVA	0.4	0.4	1.0	16.26	2.92
IVB	0.4	1.0	0.4	14.19	2.50
IVC	1.0	0.4	0.4	14.53	2.47

Liquefaction Potential

The subsurface materials are not likely to undergo liquefaction. The relatively compressible soil layers within the top 25–30 feet depth would not undergo liquefaction because of the depth of the water table (125 feet below the ground surface). Also, the material below 25–30 feet consists of dense granular soil with high (> 50) *N* values. Such materials experience dilation when subjected to shear strain, decreasing the pore pressure (e.g., Lambe and Whitman, 1969, Figure 29.6 and Table 7.4). As a result, the materials within the saturated zone are not likely to undergo liquefaction.

Staff Evaluation

The staff has reviewed Section 2.6.4, Stability of Subsurface Materials, of the SAR and concludes that the information presented in this section is adequate for use in other sections of the SAR to develop the design bases of the Facility, perform additional safety analysis, and demonstrate compliance with regulatory requirements in 10 CFR 72.102(c) and 72.102(d).

2.1.6.5 Slope Stability

There are no natural slopes close enough to the proposed Facility that require stability evaluation. The foundation excavations would be backfilled to the current ground-surface elevation, so there will not be any excavated slopes at the site.

The site layout includes four embankments: the railroad embankment, the Facility berm, the access road embankment, and the road berm. However, these embankments have been classified as not important to safety in Section 2.5.4.4 of the SAR. Also, evaluations in Section 2.1.4.4 of this SER show that failure of the embankments would not affect any structures

important to safety. Consequently, the geotechnical design of the embankments is not presented or evaluated.

The staff reviewed the applicant's discussion of slope stability and found it acceptable because:

- The slopes and slope materials of the site and vicinity have been adequately described such that safety of the site can be assessed and design bases for slope stability during external events can be developed.
- The slope stability that directly affects site conditions and the likely environmental impact of activities at the site have been sufficiently investigated and assessed.
- The severity of slope instability that may directly affect site safety has been sufficiently investigated and assessed.
- Slope stability is not a safety concern during natural or man-induced events. Therefore, no specific designs or mitigation actions with regard to slope stability are required.
- There is no known landslide area near the site that may affect site safety.

This information is acceptable for use in other sections of the SAR to develop the design bases of the Facility, perform additional safety analysis, and demonstrate compliance with regulatory requirements in 10 CFR 72.90(a-d), 72.92(a-c), and 72.122(b) with respect to this issue.

2.1.6.6 Volcanism

The staff has reviewed information presented in Section 2.6.1 and Appendix 2E of the SAR with regard to volcanism. Chemical analyses of ash layers exposed in trenches and boreholes at the Facility indicate they are chemically similar to the Walcot Tuff, which erupted approximately 6.4 Ma near Heise, Idaho (see Appendix 2E of the SAR). The closest Quaternary volcanic activity (which occurred between 950 and 880 Ka) is located more than 50 miles south of the Facility at Fumarole Butte. Therefore, volcanism is not deemed a credible event at the site.

The staff reviewed the discussion on volcanism and found it acceptable because the applicant demonstrated that volcanism is not a credible phenomenon at the Facility. This information is acceptable for use in other sections of the SAR to develop the design bases of the Facility, perform additional safety analysis, and demonstrate compliance with regulatory requirements in 10 CFR 72.92(a-c) and 72.122(b) with respect to this issue.

2.2 Evaluation Findings

The staff has reviewed the site characteristics presented in the SAR. The staff finds that the SAR provides an acceptable description and safety assessment of the site on which the PFS Facility is to be located, in accordance with 10 CFR 72.24(a). The staff also finds that the proposed site complies with the criteria of 10 CFR 72 Subpart E, as required by 10 CFR 72.40(a)(2).

2.3 References

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3 OPERATION SYSTEMS

3.1 Conduct of Review

The objective of the operations system review is to determine if the operations presented in the SAR are clear and comprehensive and fulfill the NRC regulatory requirements. The review of the operation systems included Chapter 5, Operation Systems, and selected sections of Chapters 1, 3, 4, 6, 8, 9, and 10 of the SAR, and documents cited in the SAR.

3.1.1 Operation Description

The description of the operating system was reviewed for conformance with the following regulations:

- 10 CFR 72.40(a)(5) and (13) require that the proposed activities can be conducted without endangering the health and safety of the public.
- 10 CFR 72.104(b) requires that the as low as reasonably achievable (ALARA) principle is considered in the design.
- 10 CFR 72.122(i) requires that the operation descriptions provide acceptable descriptions and discussions of the projected operating characteristics and safety considerations.
- 10 CFR 72.126(b–c) require that the design consider radiological alarm systems and direct radiation monitoring.
- 10 CFR 72.128(a)(1) requires that the design and procedures provide acceptable capability to test and monitor components important to safety.

In SAR Chapter 5, the applicant describes the generic operations to be performed in preparing the HI-STORM 100 Cask System for storage and during actual storage. The operations to be performed at the site include receipt and inspection of incoming shipping casks with canisters containing the spent fuel, transfer of the canisters containing the spent fuel from the shipping casks to the storage casks via the transfer cask, placement of the storage casks on the storage pads, surveillance of the storage casks, security of the Facility, maintenance of the health physics conditions consistent with ALARA requirements and site technical specifications, maintenance of the site and storage casks, removal of spent fuel canisters from the site, and inventory documentation management.

The shipping casks containing canisters will arrive at the site from the originating power plant either by rail or heavy haul tractor/trailer transport. When a shipping cask arrives at the site, the shipping cask, impact limiters, and shipping cradle will be visually inspected. Personnel will then transfer the shipping cask into a designated area to perform radiological monitoring. After the receipt inspection is complete, the shipping casks are transferred into the Facility restricted area and then into the Canister Transfer Building.

The transfer of the spent fuel canister from the shipping cask to the storage cask, via a transfer cask, will occur in the Canister Transfer Building. The transfer activities will use a combination of fixtures and equipment designed by the cask system vendors and equipment specifically designed for the Canister Transfer Building. After the storage cask has been loaded, the casks will be transferred from the Canister Transfer Building to the storage pad.

SAR Section 5.1 describes in detail the activities that will be performed to ensure that the stored casks do not endanger public health and safety. In summary, these activities include the following actions: after the storage casks are placed on the storage pad, the cask temperatures are measured periodically to ensure the temperature limits specified in the Technical Specifications for the specific cask design are not exceeded; security personnel control access to the storage area and identify/assess off-normal and emergency events during off-shift hours; health physics personnel ensure that the contamination levels are within the PFS Facility Technical Specifications; and maintenance personnel maintain the facilities including the storage casks, building equipment, buildings, emergency equipment, and transport systems.

The staff reviewed the operating functions described in SAR Chapters 1, 3, 4 and SAR Section 5.1 to ensure that the applicant adequately described the appropriate procedures, equipment, and personnel requirements. SAR Section 5.1 identifies the specific equipment and the personnel to accomplish the transfer, storage, and retrieval of the casks. The staff determined that the detailed procedure descriptions for operating, inspecting, and testing are consistent with the operation system.

The staff found the general description of the proposed Facility operations to be adequate. PFS Facility operations can be conducted without endangering the health and safety of the public and are, therefore, in compliance with 10 CFR 72.40(a)(5) and (13). Additionally, the SAR provides acceptable descriptions and discussions of the projected operating characteristics and safety considerations as required by 10 CFR 72.122(i). The staff found that the design and procedures provide acceptable capability to test and monitor components important to safety, in compliance with 10 CFR 72.128(a)(1).

The applicant's ALARA considerations are reviewed in Chapter 11 of this SER. Based on this review, the staff found that the design and operations consider ALARA, as required by 10 CFR 72.104(b). Radiological alarm systems and direct radiation monitoring are also considered in the design in compliance with the requirements of 10 CFR 72.126(b–c).

3.1.2 Spent Nuclear Fuel Handling Systems

Handling of the HI-STORM 100 Cask System, including the MPC, are described in detail in the HI-STORM 100 Cask System FSAR (Holtec International, 2000), which the staff has previously reviewed and found acceptable (Nuclear Regulatory Commission, 2000a, 2000b). Handling operations at the Facility will be consistent with the handling operations described in the HI-STORM 100 FSAR.

3.1.3 Other Operating Systems

The description of the other operating systems were reviewed for conformance with the following regulations:

- 10 CFR 72.104(b) requires that ALARA is considered in the design.
- 10 CFR 72.122(k)(2) requires that emergency utility services be designed to permit testing and to permit the operation of associated safety systems.
- 10 CFR 72.122(k)(3) requires that proposed design of the Facility include provisions so that emergency power is provided to permit continued functioning of all systems essential to safe storage.
- 10 CFR 72.126(b) and (c) require that the design consider radiological alarm systems and direct radiation monitoring.

In Section 3.4.5 of the SAR, the applicant discusses the structures, systems, and components (i.e., security systems, standby electrical power, cask transport vehicles, flood prevention earthworks, fire protection systems, radiation monitoring systems, and temperature monitoring systems) classified as not important to safety, but having security or operational importance. The SAR states that the design of the structures, systems, and components classified as not important to safety comply with applicable codes and standards. Further, the SAR states that the structures, systems, and components classified as not important to safety will be compatible with structures, systems, and components classified as important to safety and be designed to a level of quality to ensure that they will mitigate the effects of off-normal or accident-level events, as required.

Radiological surveys are planned for all incoming canisters as normal receiving operations at the Facility. In the event contamination above the acceptance levels is discovered, the canister will be returned to the shipper.

The staff reviewed the description of the other operating systems described in Section 5.3, and relevant information in appropriate sections of Chapters 1 and 3. The applicant's ALARA considerations are reviewed in Chapter 11 of this SER. Based on this review, the staff found that the design and operations consider ALARA as required by 10 CFR 72.104(b). Radiological alarm systems and direct radiation monitoring are considered in the design, in compliance with the requirements of 10 CFR 72.126(b–c).

The proposed design of the Facility does not require utility systems during spent fuel storage. Therefore, the emergency utility services required by 10 CFR 72.122(k)(2) are not applicable. The proposed design of the Facility does not include systems and subsystems that require continuous electric power to permit continued functioning. Since the design of the Facility does not require emergency power, 10 CFR 72.122(k)(3) is also not applicable.

3.1.4 Operation Support Systems

The descriptions of the operation support systems were reviewed for conformance with the following regulations:

- 10 CFR 72.122(i) requires that instrumentation and control systems be provided to monitor systems that are classified as important to safety.
- 10 CFR 72.122(k)(1) requires that each utility system important to safety include redundant systems to maintain the ability to perform safety functions assuming a single failure.
- 10 CFR 72.122(k)(3) requires that proposed design of the Facility include provisions so that emergency power is provided to permit continued functioning of all systems essential to safe storage.

The applicant classifies the instrumentation systems to be used to periodically monitor the Facility as not important to safety. The operation of the Facility is passive and self-contained. These storage casks do not require any instrumentation and control systems to ensure safe operation when they are placed into storage. During operation of the Facility, however, temperatures of the storage casks will be monitored. These measurements will provide a means to assess the thermal performance of the storage casks. The temperature monitors to be used at the Facility will be equipped with data recorders and alarms located in the Security and Health Physics building. The temperature monitors are not classified as important to safety. The storage casks to be used will be passively cooled; therefore, failure of a temperature monitor does not initiate an off-normal or accident condition. In addition, a periodic check for air cooling effectiveness is included as a technical specification. The proposed design of the Facility does not require utility systems during spent fuel storage. As stated above, the proposed design of the Facility does not include systems and subsystems that require continuous electric power to permit continued functioning and the design of the Facility does not require emergency power.

The staff reviewed the proposed operation support systems described in Section 5.4 of the SAR. In addition, the staff evaluated SAR Section 5.1 and appropriate sections in Chapters 3, and 8 of the SAR that identify the structures, systems, and components important to safety. The staff agrees that instrumentation systems to be used to periodically monitor the Facility are appropriately classified as not important to safety; therefore, 10 CFR 72.122(i) is not applicable. The staff found that the proposed self-contained, passive storage facility requires no permanently installed auxiliary systems. All auxiliary systems required to support loading and off-loading the system, periodic monitoring, and maintenance are designed to be portable systems. The systems are not important to safety and therefore 10 CFR 72.122(k)(1) is not applicable. Additionally, as stated above, the requirements of 10 CFR 72.122(k)(3) are not applicable because the design of the Facility does not require emergency power for systems essential to safe storage, and there are no systems essential to safe storage requiring electrical power.

3.1.5 Control Room and Control Area

The descriptions of the control room and control area were reviewed for conformance with the following regulation:

- 10 CFR 72.122(j) requires that, if appropriate, a control room or control area must be designed to permit occupancy and actions to be taken to monitor the ISFSI under normal conditions and provide safe control under off-normal and accident conditions.

The storage casks are passive storage systems. The control room and control area are not necessary to maintain the conditions required for safe operation of the Facility, to store spent fuel safely, prevent damage to the spent fuel during handling and storage, or provide reasonable assurance that the spent fuel can be received, handled, packaged, stored and retrieved without undue risk to the health and safety of the public.

The staff reviewed the control room and control areas described in Section 5.5 of the SAR. In addition, the staff has evaluated sections pertaining to monitoring instruments, limits and controls of the proposed cask systems from Chapters 1, 3, 4, 5, and 10 of the SAR. The staff found that the control room and control area are not important to safety. The Facility is a self-contained, passive storage facility that requires no permanent control room or control area to ensure safe operation; therefore, the requirements of 10 CFR 72.122(j) are not applicable.

3.1.6 Analytical Sampling

As discussed in the SAR, no analytical sampling is required. The HI-STORM 100 Cask System design will preclude release of effluents for normal, off-normal, and accident conditions during storage.

Prior to opening the shipping cask, the gas inside should be sampled to verify that canister confinement boundary is intact. The staff has determined that a license condition to this effect should be imposed.

3.1.7 Shipping Cask Repair and Maintenance

The shipping cask that will be used to transport the spent fuel to and from the Facility must be approved under 10 CFR Part 71. Repair or maintenance of such cask must be conducted in accordance with the requirements specified in the 10 CFR Part 71 certificate of compliance for that cask.

3.1.8 Pool and Pool Facility Systems

The Facility utilizes the dry cask storage technology, which houses spent fuel inside sealed, inerted canisters rather than in a spent fuel pool. Therefore, neither the use of a pool nor any system supporting a pool is incorporated into the Facility.

3.2 Evaluation Findings

The staff found that the proposed operating procedures are adequate. PFS Facility operations meet the regulatory requirements and can be conducted without endangering the health and safety of the public. Therefore, the staff found that the operation system description is acceptable.

License Condition

LC3-1 Prior to removing the shipping cask closure lid, the gas inside the cask shall be sampled to verify that canister confinement boundary is intact.

3.3 References

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4 STRUCTURES, SYSTEMS, AND COMPONENTS AND DESIGN CRITERIA EVALUATION

4.1 Conduct of Review

Chapter 3 of the SAR identifies the principal design criteria for the Facility. These design criteria are derived from the requirements of 10 CFR Part 72 and applicable industry codes and standards. The SAR identifies PWR and BWR spent fuel as the material to be stored. It also identifies the general design criteria for structures, systems, and components classified as important to safety with respect to withstanding the effects of environmental conditions and natural phenomena. The worst case loads for normal, off-normal, and accident conditions are identified. Structures, systems, and components important to safety are designed for safe confinement and storage of the spent nuclear fuel without the release of radioactive material. This chapter also categorizes all structures, systems, and components as either important to safety or not important to safety. Table 3.6-1 of the SAR provides a summary of the key design criteria for the Facility. The design criteria are compared to the actual design in subsequent chapters.

The storage cask to be used at the Facility is the HI-STORM 100 Cask System as described in the HI-STORM 100 FSAR (Holtec International, 2000). The HI-STORM 100 Cask System has been approved by NRC for general use under Certificate of Compliance No. 1014 (Nuclear Regulatory Commission, 2000a). Where applicable, the staff relied on the review carried out during the certification process of the cask system, as documented in the NRC's HI-STORM 100 SER (Nuclear Regulatory Commission, 2000b).

4.1.1 Materials to be Stored

The materials to be stored at the Facility are PWR and BWR spent fuel assemblies that are approved for storage in the HI-STORM 100 Cask System. The approved contents are specified in Appendix B of Certificate of Compliance No. 1014. The physical, thermal, and radiological characteristics of the spent nuclear fuel are described in detail and evaluated in the HI-STORM 100 FSAR. Section 3.1.1 of the PFS Facility SAR provides a brief discussion of the materials to be stored at the Facility. This discussion is consistent with the information in the HI-STORM 100 FSAR.

Table 4-1. Summary of Private Fuel Storage Facility materials to be stored and spent fuel specifications

Design Parameters	Design Conditions	Applicable Reference
Type of Fuel	See Appendix B of HI-STORM 100 Certificate of Compliance	HI-STORM 100 FSAR
Fuel Characteristics	See Appendix B of HI-STORM 100 Certificate of Compliance	HI-STORM 100 FSAR

4.1.2 Classification of Structures, Systems, and Components

This section contains a review of SAR Section 3.4, Classification of Structures, Systems, and Components. The staff reviewed the discussion on classifications of structures, systems, and components with respect to the following regulatory requirements:

- 10 CFR 72.120(a) requires that, pursuant to the provisions of 10 CFR 72.24, an application to store spent fuel in an ISFSI include the design criteria for the proposed storage installation. These design criteria establish the design, fabrication, construction, testing, maintenance and performance requirements for structures, systems, and components important to safety as defined in 10 CFR 72.3. The general design criteria identified in this subpart establish minimum requirements for the design criteria for an ISFSI. Any omissions in these general design criteria do not relieve the applicant from the requirement of providing the necessary safety features in the design of the ISFSI.

- 10 CFR 72.144(a) requires that the licensee establish, at the earliest practicable time consistent with the schedule for accomplishing the activities, a quality assurance program which complies with the requirements of this subpart. The licensee shall document the quality assurance program by written procedures or instructions and shall carry out the program in accordance with these procedures throughout the period during which the ISFSI is licensed. The licensee shall identify the structures, systems, and components to be covered by the quality assurance program, the major organizations participating in the program, and the designated functions of these organizations.

In 10 CFR 72.3, structures, systems, and components are identified as items whose functions are to: (1) maintain the conditions required to store spent nuclear fuel safely; (2) prevent damage to the spent fuel container during handling and storage; and (3) provide reasonable assurance that spent nuclear fuel can be received, handled, packaged, stored, and retrieved without undue risk to the health and safety of the public. The SAR list the structures, systems, and components based on this definition as required by 10 CFR 72.120(a).

SAR Section 3.3, Safety Protection Systems identifies safety protection systems and provides a brief description of the important characteristics of each system. The classification consists of two levels: important to safety and not important to safety. The important to safety classification contains three categories based on the potential impact to safe operation:

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

Classification Category B—Major Impact on Safety, whose failure or malfunction could indirectly result in a condition adversely affecting public health and safety. The failure of a Category B item, in conjunction with failure of an additional item, could result in an unsafe condition.

Classification Category C—Minor Impact on Safety, whose failure or malfunction would not be likely to create a situation adversely affecting public health and safety.

4.1.2.1 Classification of Structures, Systems, and Components – Items Important to Safety

Those structures, systems, and components considered important to safety are identified in Sections 3.3 and 3.4 of the SAR and are provided, with corresponding categories, in Table 4-2 in this SER. Details associated with the first four items (spent nuclear fuel canister, storage cask, transfer cask, and associated lifting devices) in Table 4-2 are cask-specific and presented in the HI-STORM 100 FSAR.

Table 4-2. Quality assurance classification of structures, systems, and components important to safety (Based on SAR Table 3.4-1)

Category	Structures, Systems, and Components	Logic
A	Spent Nuclear Fuel Canister	Serves as the primary confinement structure for the Spent Nuclear Fuel assemblies and is designed to remain intact under all accident conditions analyzed. It provides confinement, criticality control, heat transfer capability, and radiation shielding
B	Storage Cask	Serves as the primary component for protecting the canister during storage from environmental conditions and provides radiation shielding and canister heat rejection.
B	Transfer Cask	Designed to support the canister during transfer lift operations and provide radiation shielding and canister heat rejection.
B	Associated Lifting Devices	Designed to preclude the accidental drop of a canister.
B	Canister Transfer Building	Designed to protect the canister from adverse natural phenomena during shipping cask load/unload operations and canister transfer operations, provide radiological shielding to workers during transfer operations, and support for the canister transfer crane. The Canister Transfer Building also houses the fire-suppression system.
B	Canister Transfer Overhead Bridge Crane	Designed as a single failure proof system to preclude the accidental drop of a shipping cask during load/unload operations or a canister during the canister transfer operations.
B	Canister Transfer Semi-Gantry Crane	Designed as a single failure proof system to preclude the accidental drop of a shipping cask during load/unload operations or a canister during the canister transfer operations.
B	Seismic Support Struts	Designed to ensure that the transfer, storage, and shipping casks will remain stable and not topple in the event of an earthquake.
C	Cask Storage Pads	Designed to ensure a stable and level support surface for the storage cask under normal, off-normal, and accident conditions. It provides a yielding surface for the drop/tip-over of the storage cask.

The spent nuclear fuel canister has been properly classified as a Category A important to safety item because it serves as the primary confinement structure. Its failure could lead to the release of radioactive material. Sufficient description of the spent nuclear fuel canister is provided in SAR Sections 3.4.1.1, Canister, and 4.2.1.4, Components.

The following components have been properly classified as Category B important to safety items: the storage cask, the transfer cask, the canister transfer building, the canister transfer overhead bridge crane, the canister transfer semi-gantry cranes, associated lifting devices, and the seismic support struts. Each of these items is designed to protect the spent nuclear fuel canister during specific phases of handling and storage of the spent fuel. Failure of one or more of these Category B items, combined with the subsequent failure of the canister, is necessary to lead to a condition adversely affecting public health and safety.

The HI-STORM 100 storage cask protects the spent nuclear fuel canister during storage. Sufficient description of the storage cask is provided in SAR Sections 3.4.1.2, Concrete Storage Cask, and 4.2.1.4, Components. The transfer cask protects the spent nuclear fuel canister during transfer lift operations in the Canister Transfer Building. Sufficient description of the transfer cask is provided in SAR Sections 3.4.1.3, Transfer Cask, and 4.7.3, HI-STORM Transfer Equipment. The Canister Transfer Building is designed to protect the canisters from adverse natural phenomena during cask loading/unloading and canister transfer operations. The building also provides radiation protection to workers during canister transfer operations and support of the canister transfer cranes. Sufficient description of the Canister Transfer Building is provided in SAR Section 4.7.1, Canister Transfer Building. The canister transfer cranes and associated lifting devices are important to safety because they will be used to support the spent nuclear fuel canister and transfer cask during the transfer process. Sufficient description of the canister transfer cranes is provided in SAR Sections 3.4.4, Canister Transfer Cranes, and 4.7.2, Canister Transfer Cranes. Sufficient description of the associated lifting devices is provided in SAR Sections 3.4.1.4, Lifting Devices, and 4.7.3, HI-STORM Transfer Equipment. Seismic support struts are used to support the storage, transfer, and shipping casks during canister transfer operations in case of an earthquake. These support struts are to be attached to the walls of the Canister Transfer Building and provide support during a seismic event. Adequate description of the seismic struts is provided in SAR Sections 3.4.5, Seismic Support Struts, and 4.7.1.4.1, Seismic Support Struts. Based on the above discussion, the staff concludes that these Category B important to safety items are correctly classified.

Failure of the cask storage pads would not create a situation adversely affecting public health and safety. Sufficient description of the cask storage pad is provided in SAR Sections 3.4.2, Cask Storage Pads, and 4.2.3, Cask Storage Pads. Consequently, the staff concluded that the cask storage pads have been correctly identified as Category C important to safety items.

4.1.2.2 Classification of Structures, Systems, and Components – Items Not Important to Safety

Based on SAR Table 3.4-1, the classification of structures, systems, and components not important to safety includes items or services that do not involve a safety related function and that are not subject to special utility requirements or NRC-imposed regulatory requirements.

Structures, systems, and components not important to safety include: the PFS Facility infrastructure, Security and Heath Physics Building, Administration Building, Operations and Maintenance Building, fire detection and suppression systems, security systems, electrical systems, radiation monitors, temperature monitoring system, flood control berm, cask transporter, and offsite transportation components. The storage facility infrastructure, buildings, and facilities are necessary to support operation of the Facility. However, they are not necessary to ensure safe storage of the spent fuel because the storage cask system is passive. Therefore, they are classified as not important to safety.

The fire detection and suppression systems are contained within the Canister Transfer Building. The construction materials of the Canister Transfer Building do not support combustion, and the fire-prone materials are limited to diesel fuel and tires of the heavy haul trucks. Fires are analyzed in the accident analysis section of the SAR. The area surrounding the storage pads and Canister Transfer Building includes a gravel-covered fire break with vegetation control to limit potential fuel for fires. The nonflammable nature of the materials of construction, other passive design features, and the limited fuel sources at the Facility lead to the conclusion that the fire detection and suppression systems are correctly classified as not important to safety.

There are a number of systems that are security related: intrusion detection system, closed circuit television system, restricted area lighting, and security alarm stations. Each system is used to support the activities of the security personnel who monitor the controlled area of the facility. If systems fail, the security personnel can still perform their required functions. Therefore, the security systems are correctly classified as not important to safety.

Because the HI-STORM 100 storage cask system is a passive system, the uninterrupted power supply, backup diesel generator, and normal electrical power can also be classified as not important to safety. No electrical power is required for the storage system to perform its design functions.

The passive design of the cask also affects classification of the radiation monitors and temperature monitoring system. The radiation monitors are established to protect the health and safety of the workers. It has been demonstrated by analysis that the radiation levels at the site boundary will be below those identified in the applicable radiation protection regulations. The public is restricted from access into the controlled area. Therefore, the radiation monitors are correctly classified as not important to public safety.

The thermal monitors track the temperature of the air in the cooling passages of the storage cask. Upon loss of thermal monitoring, an alarm will sound and repair of the monitoring system will begin. The thermal monitoring system is intended to identify blockage of the cask cooling air passages and resulting rise of the cask temperature. The cask, by design, is not adversely affected by complete blockage of the air passages for 72 h. It is not necessary to continuously monitor the temperature since the canister since at least 72 h. must pass before the canister and fuel cladding temperature reach the allowable limits. Therefore, the thermal monitoring system is appropriately classified as not important to safety.

The flood control berm and drainage ditch are to prevent sheet flow over the site, to facilitate maintenance at the site and to maintain access to the casks on the storage pads in case of flooding. The flood control berm is not important to safety because the Facility elevation is

above the PMF level. Further, the HI-STORM 100 storage cask is designed to resist the effects of full immersion in flood waters.

The cask transporter is also classified as not important to safety. Potential failure mechanisms of the transporter involve the drive-train, brakes, electrical system, or lift beam hydraulic ram. None of these potential failures would cause the transporter or the cask to tipover. Of these potential failures, only those that could drop the cask would have a possibility of damaging the cask or its internal components. The HI-STORM 100 FSAR (Holtec International, 2000) has demonstrated that the storage cask can be dropped a height of 11 in. without impairing confinement system integrity or fuel retrievability. However, the 11 in. drop height is based on a softer pad than is proposed at the PFS Facility. The cask storage pads at the proposed PFS Facility will be stiffer due to increased stiffness of the soil-cement layer overlaying the existing soil. Therefore, the applicant has stated that the transporter will be designed to limit the lift height of the cask to 9 in. This height is based on site-specific analyses of drop events on the PFS Facility storage pads to estimate the limiting deceleration level on the fuel rods (Holtec International, 2001). As calculated by PFS, a vertical drop of the PFS cask upon the cask storage pad, up to 9 in. will produce decelerations bounded by the 45g design basis. The cask transporter will also be designed to preclude tipover under site-specific seismic, tornado winds, and tornado missile loads. Therefore, the cask transporter can be classified as an item not important to safety.

Another group of structures, systems, and components that are classified as not important to safety are the road transport and railroad line alternatives. These are classified as such because the shipping casks that will be used to transport the spent fuel are designed and approved under 10 CFR Part 71. Transportation equipment is outside the scope of this review.

4.1.2.3 Classification of Structures, Systems, and Components - Conclusion

The staff evaluated the classification of structures, systems, and components important to safety by reviewing SAR Chapter 3, Principal Design Criteria; documents cited in the SAR; and other relevant literature. The staff found that the SAR appropriately classifies the structures, systems, and components important to safety. The design criteria for the structures, systems, and components important to safety are adequately identified as required by 10 CFR 72.120(a). Details of the quality assurance program evaluation are contained in Chapter 12 of this SER. The staff determined that the classification of the structures, systems, and components important to safety and their associated categories are consistent with the regulatory requirements of 10 CFR 72.144(a) and associated technical information content of the application, as specified in 10 CFR 72.24(n).

4.1.3 Design Criteria for Structures, Systems, and Components Important to Safety

The principal design criteria identified for structures, systems, and components important to safety at the Facility are described in SAR Chapter 3, Principal Design Criteria. This section contains a review of Section 3.2, Structural and Mechanical Safety Criteria; Section 3.3, Safety Protection Systems; and SAR Section 3.6, Summary of Design Criteria. Details of the design criteria evaluation are provided in Sections 4.1.3.1 to 4.1.3.7 of this SER.

4.1.3.1 General

The staff reviewed the discussion of the general design criteria for structures, systems, and components with respect to the following regulatory requirements:

- 10 CFR 72.120(a) requires that, pursuant to the provisions of 10 CFR 72.24, an application to store spent fuel in an ISFSI include the design criteria for the proposed storage installation. These design criteria establish the design, fabrication, construction, testing, maintenance and performance requirements for structures, systems, and components important to safety as defined in 10 CFR 72.3. The general design criteria identified in this subpart establish minimum requirements for the design criteria for an ISFSI. Any omissions in these general design criteria do not relieve the applicant from the requirement of providing the necessary safety features in the design of the ISFSI.

- 10 CFR 72.122(h) specifies the criteria for confinement barriers and systems, including: 72.122(h)(1), which requires that the spent fuel cladding be protected during storage against degradation that leads to gross ruptures or the fuel must be otherwise confined such that degradation of the fuel during storage will not pose operational safety problems with respect to its removal from storage; 72.122(h)(3), which 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; 72.122(h)(4), which requires that storage confinement systems have the capability for monitoring in a manner such that the licensee will be able to determine when corrective action needs to be taken to maintain safe storage conditions; and 72.122(h)(5), which requires that the high-level radioactive waste be packaged to allow handling and retrievability without the release of radioactive materials to the environment or radiation exposures in excess of Part 20 limits (also, the package must be designed to confine the high-level radioactive waste for the duration of the license).

- 10 CFR 72.144(c) requires that the licensee base the requirements and procedures of its quality assurance program on the following considerations concerning the complexity and proposed use of the structures, systems, or components: (1) The impact of malfunction or failure of the item on safety; (2) The design and fabrication complexity or uniqueness of the item; (3) The need for special controls and surveillance over processes and equipment; (4) The degree to which functional compliance can be demonstrated by inspection or test; and (5) The quality history and degree of standardization of the item.

A summary of the PFS Facility general design criteria is provided in Table 4-3 of this SER.

Table 4-3. Summary of Private Fuel Storage Facility Design Criteria—General (Based on SAR Table 3.6-1)

Design Parameters	Design Conditions	Reference
Design Life	40 yr	PFS Facility SAR
Storage Capacity	40,000 MTU of commercial spent nuclear fuel	PFS Facility SAR
Number of Casks	Approximately 4,000 casks	PFS Facility SAR

The design life of structures, systems, and components important to safety is based on their ability to withstand the applied loads. The applied loads are defined in terms of an annual probability of exceeding the design load. Analysis procedures are used to demonstrate the ability of the structures, systems, and components to withstand the applied loads with additional factors applied to the loads and material allowables by the referenced codes and standards. The majority of design loads for the PFS Facility, identified in Table 3.6-1 of the SAR, are based on a 50 yr mean recurrence interval ($P_a = 0.02$) or longer. The loads specified in the SAR are consistent with standard engineering practice, as identified in American Society of Civil Engineers (ASCE) ASCE 7-95 (American Society of Civil Engineers, 1996) that identifies the minimum design loads for buildings and other structures. The storage capacity and number of casks to be stored at the Facility have been identified in the SAR. Based on a given annual probability (P_a) of exceeding the design load and service life (n), the probability (P_n) that the design load will be equaled or exceeded at least once during the service life is given by (Commentary Section of ASCE 7–95, 1996):

$$P_n = 1 - (1 - P_a)^n$$

The principal criteria used in the design are given in Section 3.2, Structural and Mechanical Safety Criteria, and summarized in SAR Section 3.6, Summary of Design Criteria. The storage system characteristics are given in Table 4-4 of this SER. These characteristics are based on information provided in the HI-STORM 100 FSAR.

Table 4-4. Summary of Private Fuel Storage Facility Design Criteria—Storage System Characteristics (Based on SAR Table 3.6-1)

Design Parameters	Design Conditions		Applicable Criteria and Codes
Canister Capacity	Maximum 24 PWR assemblies/canister Maximum 68 BWR assemblies/canister		HI-STORM FSAR, Section 1.1
Weights (maximum)	Storage Cask	268,334 lb	HI-STORM FSAR, Table 3.2.1
	Loaded Canister	87,241 lb	HI-STORM FSAR, Table 3.2.1
	Transfer Cask	152,636 lb	HI-STORM FSAR, Table 3.2.2
	Shipping Cask	153,080 lb	Shipping Cask SAR

The staff reviewed the general design criteria for the storage system characteristics identified in Tables 4-3 and 4-4 of this SER. The staff found that they are consistent with the HI-STORM 100 FSAR. Definitions of the normal, off-normal, and accident loads are given in SAR Section 3.2, Structural and Mechanical Safety Criteria. The quality standards for design basis of structures, systems, and components important to safety are provided in SAR Chapters 3, Principal Design Criteria, and 11, Quality Assurance. These design criteria, in part satisfy the requirements of 10 CFR 72.120(a) and 72.122(h) in that design criteria are identified, and structures, systems, and components important to safety will be designed to quality standards commensurate with the important to safety functions to be performed to satisfy the requirements of 10 CFR 72.144(c).

4.1.3.2 Structural

The staff reviewed the discussion on structural design criteria of structures, systems, and components in the SAR with respect to the following regulatory requirements:

- 10 CFR 72.102(f) requires that the design earthquake for use in the design of structures be determined as follows: (1) for sites that have been evaluated under the criteria of appendix A of 10 CFR Part 100, the design earthquake must be equivalent to the safe shutdown earthquake for a nuclear power plant; and (2) regardless of the results of the investigations anywhere in the continental U.S., the design earthquake must have a value for the horizontal ground motion of no less than 0.10 g with the appropriate response spectrum.

- 10 CFR 72.120(a) requires that, pursuant to the provisions of 10 CFR 72.24, an application to store spent fuel in an ISFSI include the design criteria for the proposed storage installation. These design criteria establish the design, fabrication, construction, testing, maintenance and performance requirements for structures, systems, and components important to safety as defined in 10 CFR 72.3. The general design criteria identified in this subpart establish minimum requirements for the design criteria for an ISFSI. Any omissions in these general design criteria do not relieve the applicant from the requirement of providing the necessary safety features in the design of the ISFSI.

- 10 CFR 72.122(b)(1) requires structures, systems, and components important to safety be designed to accommodate the effects of, and to be compatible with, site characteristics and environmental conditions associated with normal operation, maintenance, and testing of the ISFSI and to withstand postulated accidents.

- 72.122(b)(2) requires structures, systems, and components important to safety to be designed to withstand the effects of natural phenomena such as earthquakes, tornadoes, lightning, hurricanes, floods, tsunami, and seiches, without impairing their capability to perform safety functions. The design bases for these structures, systems, and components must reflect: (i) appropriate consideration of the most severe of the natural phenomena reported for the site and surrounding area, with appropriate margins to take into account the limitations of the data and the period of time in which the data have

accumulated, and (ii) appropriate combinations of the effects of normal and accident conditions and the effects of natural phenomena. The ISFSI should also be designed to prevent massive collapse of building structures or the dropping of heavy objects, as a result of building structural failure, on the spent fuel or high-level radioactive waste or on structures, systems, and components important to safety.

- 72.122(b)(4) specifies that if the ISFSI is located over an aquifer which is a major water resource, measures must be taken to preclude the transport of radioactive materials to the environment through this potential pathway.

- 10 CFR 72.122(c) requires that structures, systems, and components important to safety be designed and located so that they can continue to perform their safety functions effectively under credible fire and explosion exposure conditions. Noncombustible and heat-resistant materials must be used wherever practical throughout the ISFSI, particularly in locations vital to the control of radioactive materials and to the maintenance of safety control functions. Explosion and fire detection, alarm, and suppression systems shall be designed and provided with sufficient capacity and capability to minimize the adverse effects of fires and explosions on structures, systems, and components important to safety. The design of the ISFSI must include provisions to protect against adverse effects that might result from either the operation or the failure of the fire suppression system.

SAR Section 3.2, Structural and Mechanical Safety Criteria, addresses the structural and mechanical design criteria. The design criteria for the site include: dead loads, live loads, wind, tornado, tornado missiles, flood, seismicity, snow and ice, soil pressure, explosion overpressure, fire, ambient temperature and humidity, solar radiation, and lightning. Information on the derivation of site-specific design criteria for the meteorology, hydrology, and seismology are contained in SAR Chapter 2, Site Characteristics.

The structural design criteria for major components are provided in Table 4-5 of this SER. The design of the proposed Facility is based on the use the HI-STORM 100 Cask System, which has been approved by the NRC for use under the general license provisions of 10 CFR Part 72.

The design criteria for the pressure vessel portions of the cask system conform to standard engineering practice, as identified in the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code (American Society of Mechanical Engineers, 1998). The ASME Boiler and Pressure Vessel Code establishes rules of safety governing the design, fabrication, and inspection during construction of boilers and pressure vessels. This code contains mandatory requirements, specific prohibitions, and nonmandatory guidance for selection of materials, design, fabrication, examination, inspection, testing, certification, and pressure relief.

For concrete components, as identified in Table 4-5, the design criteria are based on the American Concrete Institute's ACI 349-90. ACI 349-90 specifies the proper design and construction of concrete structures that form part of a nuclear power plant and that have nuclear safety related functions, but does not cover concrete reactor vessels and concrete

containment structures. The structures covered by the ACI code include concrete structures inside and outside the containment system.

The design criteria for structural steel, as identified in Table 4-5, are based on the American Institute of Steel Construction's ANSI/AISC N690 (American National Standards Institute/American Institute of Steel Construction, 1994). ANSI/AISC N690 is the specification and commentary for the design, fabrication, and erection of structural steel for safety-related structures for nuclear facilities.

The design criteria for the cranes are in accordance with ASME NOG–1. The ASME NOG–1 covers electric overhead and gantry multiple girder cranes with top running bridge and trolley and components of cranes used at nuclear facilities. In addition, NUREG–0554 (Nuclear Regulatory Commission, 1979), and NUREG–0612 (Nuclear Regulatory Commission, 1980) are identified for the design criteria of the cranes. NUREG–0554 identifies design criteria for single-failure-proof cranes for nuclear power plants. NUREG–0612 identifies controls for handling heavy loads at nuclear power plants.

Table 4-5. Summary of Private Fuel Storage Facility Design Criteria— Component Structural Design Criteria (Based on SAR Table 3.6-1)

Design Parameters	Design Conditions	Applicable Criteria and Codes
HI-STORM 100 Cask System Load Criteria	Canister: See HI-STORM 100 FSAR Table 2.2.6 Internals: See HI-STORM 100 FSAR Table 2.2.6 Storage Cask: See HI-STORM 100 FSAR Table 2.2.6 Transfer Cask: See HI-STORM 100 FSAR Table 2.2.6	ASME III, NB ASME III, NG ASME III, NF, ACI–349 ASME III, NF, ANSI N14.6
Canister Transfer Crane Designs	Type I, single-failure-proof 200-ton overhead bridge crane 150-ton semi-gantry crane	ASME NOG–1, NUREG–0554, and NUREG–0612
Cask Storage Pad Designs	Normal, off-normal, and accident loading	ACI 349-90
Canister Transfer Building Reinforced Concrete Designs	Normal, off-normal, and accident loading	ACI 349-90
Canister Transfer Building Structural Steel Designs	Normal, off-normal, and accident loading	ANSI/AISC N690

The structural design loads for structures, systems, and components important to safety are provided in Table 4-6 of this SER. As identified, the structures, systems, and components important to safety are designed to withstand the effects of environmental conditions and natural phenomena for normal, off-normal, and accident conditions. Important to safety design criteria for the HI-STORM 100 storage system are described in its FSAR. Table 4-6 identifies the HI-STORM 100 design criteria in relationship to the PFS Facility design criteria. Review in

this SER is limited to identification of enveloping design criteria. Adequacy of the HI-STORM 100 Cask System design criteria are discussed in the NRC's HI-STORM SER.

Site-specific design criteria not enveloped by the HI-STORM FSAR criteria are identified in Section 3.2 of the SAR. These criteria include the specific site criterion, storage system affected, and the corresponding section in the SAR where it is addressed. Structural design criteria and radiological protection and confinement criterion are identified. The structural criteria are discussed in this chapter of the SER. Consideration of the radiological protection and confinement criterion are contained in other chapters of the SER.

Table 4-6. Summary of Private Fuel Storage Facility Design Criteria—Structural Design Loads (Based on SAR Table 3.6-1)

Design Parameters	PFS Facility Design Criteria	Applicable Criteria and Codes	HI-STORM 100 MPC Design Criteria (HI-STORM 100 FSAR, Table 2.0.1)	HI-STORM 100 Overpack Design Criteria (HI-STORM FSAR, Table 2.0.2)	HI-TRAC Transfer Cask Design Criteria (HI-STORM 100 FSAR, Table 2.0.3)
Wind	90 mph, normal speed	ASCE–7 (0.02 annual frequency)	Protected by overpack	Enveloped by Tornado Wind	Protected in transfer facility
Tornado	240 mph, maximum speed 190 mph, rotational speed 50 mph, translational speed 150 ft, radius of maximum speed 1.5 psi, pressure drop 0.6 psi/sec rate of drop	Regulatory Guide 1.76	Protected by overpack	360 mph, maximum speed 290 mph, rotational speed 70 mph, translational speed 3.0 psi, pressure drop	Protected in transfer facility
Tornado Missiles	3990 lb automobile, 134 ft/sec 750 lb 12 in. schedule 40 pipe, 23 ft/sec 1124 lb wooden utility pole, 85 ft/sec 9 lb 1 in. diameter steel rod, 26 ft/sec 287 lb 6 in. schedule 40 pipe, 33 ft/sec 115 lb wood plank, 190 ft/sec	NUREG–0800, Section 3 5.1.4	Protected by overpack	3990 lb automobile, 185 ft/sec 275 lb 8 in. rigid solid steel cylinder, 185 ft/sec 1 in. diameter steel sphere, 185 ft/sec	3990 lb automobile, 185 ft/sec 275 lb 8 in. rigid solid steel cylinder, 185 ft/sec 1 in. diameter steel sphere, 185 ft/sec
Flood	PFS Facility is not in a flood plain and is above the PMF elevation. Details contained in Section 2.3.2.3 of the PFS Facility SAR.	NUREG–0800, Section 3.4.1	125 ft. water depth	125 ft. flood height 15 ft/sec flood velocity	Protected in transfer facility
Seismic	PGA of 0.711g, horizontal (both directions) and 0.695g vertical. Probabilistic design basis ground acceleration identified in Section 2.6 of the PFS Facility SAR.	10 CFR 72.102	$G_H + 0.53\ G_V \leq 0.53$	$G_H + 0.53\ G_V \leq 0.53$	NA

Table 4-6. Summary of Private Fuel Storage Facility Design Criteria—Structural Design Loads (Based on SAR Table 3.6-1)

Design Parameters	PFS Facility Design Criteria	Applicable Criteria and Codes	HI-STORM 100 MPC Design Criteria (HI-STORM 100 FSAR, Table 2.0.1)	HI-STORM 100 Overpack Design Criteria (HI-STORM FSAR, Table 2.0.2)	HI-TRAC Transfer Cask Design Criteria (HI-STORM 100 FSAR, Table 2.0.3)
Snow and Ice	P(g) = 45 psf	ASCE–7, Tooele County Building Department	Protected by Overpack	100 psf	Protected in transfer facility
Allowable Soil Pressure	Static = 4 ksf max Dynamic = Varies by footing type/size. Details contained in Section 2.6.1.12 of the SAR.	NUREG–0800, Section 2.5.4	NA	NA	NA
Explosion Overpressure	The PFS Facility design and layout shall assure that the peak positive incident overpressure at important to safety structures, systems, and components does not exceed 1.0 psi from credible and offsite explosions.	Reg. Guide 1.91	60 psig (external)	10 psid for 1 seconds 5 psid steady state	NA
Ambient Conditions	Low Temperature = -30 °F Max. Annual Average Temp. = 51 °F Average Daily Max. Temp. = 95 °F Humidity = 0–100 percent	National Oceanic and Atmospheric Administration Data–Salt Lake City, Utah, Climate Data	See Tables 2.0.2 and 2.0.3	Min. Ambient Temp. = -40 °F Max. Ambient Temp. = 100 °F Max. Yearly Average Temp. = 80 °F Extreme Environmental Temperature = 125 °F	Min. Ambient Temp. = 0 °F Max. Ambient Temp. = 100 °F Max. Yearly Average Temp. = 100 °F

G_H = peak seismic horizontal ground acceleration
G_V = peak seismic vertical ground acceleration
NA = not applicable

Wind

Figure 6-1 in ASCE 7-95 (American Society of Civil Engineers, 1996) identifies a design basis wind speed of 90 mph for the region. Information provided in SAR Section 2.3.1.3.2, Extreme Winds, for Salt Lake City region identifies the wind speed with a 50-yr return period as 70.4 mph, which, taking into account a gust response factor, results in a 88.7 mph design wind speed. As identified in Table 2.3.5 of the SAR, the mean wind speed and direction at the PFS Facility site and Salt Lake City are consistent. Therefore, data obtained at Salt Lake City can be used to represent the site. The staff reviewed the design basis wind (90 mph) for the Facility and found that it is consistent with that identified in ASCE 7-95 (American Society of Civil Engineers, 1996) for this location. The requirements of 10 CFR 72.120(a) and 72.122(b) are satisfied in that the effects of site conditions and environmental conditions are considered in the Facility design.

Tornado

The design basis tornado wind loads are based on information provided in Regulatory Guide 1.76 (U.S. Atomic Energy Commission, 1974). Tooele County is located in Tornado Intensity Region III, as defined by Regulatory Guide 1.76. The parameters for the tornado identified in the SAR are those given in Regulatory Guide 1.76. Based on data provided in SAR Section 2.3.1.3.3, Tornadoes, the most severe tornado observed in the region was classified as F1 with a corresponding wind speed of 73 to 112 mph. The specified design criteria specify greater wind speeds than those observed. Specifically, the PFS Facility design criterion for tornado specifies a maximum speed of 240 mph with an associated pressure drop of 1.5 psi. The probability of a tornado striking the PFS Facility site is given as 1.37×10^{-6} per year in the PFS Facility SAR Section 2.3.1.3.3.

Tornado Missiles

The tornado missiles, identified in the SAR, are those specified as Spectrum II missiles for Region III in NUREG–0800, Section 3.5.1.4 (Nuclear Regulatory Commission, 1981). These are considered to be representative of potential missiles present at the site. As identified in the HI-STORM 100 FSAR, the cask-specific tornado missiles correspond to Spectrum I missiles in NUREG–0800. Use of either Spectrum I or II missile is considered acceptable by the NRC. The staff reviewed the design basis tornado conditions for the Facility and found that they are consistent with design criteria, as specified by NUREG–0800, Section 3.5.1.4, to withstand tornadoes, in accordance with the requirements of 10 CFR 72.120(a) and 72.122(b).

Flood

The maximum probable flood for the site is at elevation 4,468.8 ft above mean sea level in the southeast corner and 4,456.8 ft above mean sea level in the northeast corner. The corresponding site elevations are 4,475 and 4,463 ft above mean sea level, respectively. Analysis of the maximum probable flood level is based on the maximum probable precipitation and the surface hydrology of the region given in SAR Section 2.4, Surface Hydrology. In addition, an earthen berm and drainage ditch system are to be constructed at the site. The berm is designed to ensure that sheet flow will not approach the storage casks on the pads or the Canister Transfer Building. Therefore, the forces due to flood waters and flood protection

measures do not need to be considered in design of structures, systems, and components important to safety. The staff therefore concludes that the Facility design is consistent with design criteria of NUREG–0800 and ASCE 7–95 to withstand floods as required by 10 CFR 72.120(a) and 72.122(b).

Seismicity

The staff reviewed the data presented in the SAR associated with seismic design criteria at the Facility. SAR Section 3.2.10, Seismic Design, gives the seismic design criteria, based on probabilistic site-specific seismology studies summarized in SAR Section 2.6, Geology and Seismology. PFS has requested an exemption from the seismic requirement of 10 CFR 72.102(f). Discussions of the implications of this request for exemption are contained in Section 2.1.6 of this SER. The resulting site-specific design response spectra are anchored at a peak ground acceleration (PGA) of 0.711g in both horizontal directions and 0.695g in vertical direction. The horizontal and vertical design response spectra curves have been identified in the Geomatrix Consultants, Inc. report (Geomatrix Consultants, Inc., 2001). The site-specific seismic design criteria of the Facility are not bounded by the HI-STORM 100 seismic design criteria. The seismic design criteria are based on the site-specific probabilistic seismic hazards analysis given in SAR Chapter 2, Site Characteristics, which has been evaluated in Chapter 2 of this SER. The applicant's analysis of the HI-STORM 100 storage cask under the site-specific design basis seismic event is evaluated in Chapters 5 and 15 of this SER. The staff reviewed the seismic design criteria for the Facility and found that they are properly identified as required by 10 CFR 72.120(a) and 72.122(b).

Snow and Ice

Figure 7-1 of ASCE 7-95 (American Society of Civil Engineers, 1996) identifies that the snow load at the proposed site should be estimated from results of a site-specific case study. Based on the elevation of the Goshute Reservation (elevation 4,600–4,700 ft above mean sea level), the Tooele County Building Department stated that a ground snow design load of 43 lb/ft^2 would be required to comply with the Uniform Building Code (UBC) (International Conference of Building Officials, 1997). SAR Section 3.2.3, Snow and Ice Load, states that the PFS Facility has a design ground snow load of 45 lb/ft^2 that bounds the UBC requirements. As identified in the HI-STORM 100 FSAR (Holtec International, 2000), the overpack is designed for 100 lb/ft^2 snow and ice load. This bounds the PFS Facility site design criteria. The staff reviewed the snow and ice loading criteria and determined that they are appropriate and in accordance with the requirements of 10 CFR 72.120(a) and 72.122(b).

Allowable Soil Pressure

The allowable soil pressure design criteria are based on site-specific investigations summarized in SAR Section 2.6.4, Stability of Subsurface Materials. Review of the soil classification and soil properties identified at the site are presented in Section 2.1.6.4, Stability of Subsurface Materials, of this SER. The allowable soil pressure design criteria are applicable to the storage pad and Canister Transfer Building designs. The applicant has presented acceptable analysis to determine values of allowable bearing pressure consistent with the site-specific soil properties and accepted design earthquake loading, as identified in Section 2.1.6.4 of this SER. The staff reviewed the SAR and determined that the allowable soil pressure design criteria are

appropriately specified in accordance with the requirements of 10 CFR 72.120(a) and 10 CFR 72.122(b).

Explosive Overpressure

The explosive overpressure design criterion for the Facility is based on the assumption that all credible events that produce a peak positive incident overpressure will result in a pressure at the structures, systems, and components important to safety of less than 1 psi.

The 1 psi limit is based on Regulatory Guide 1.91 (Nuclear Regulatory Commission, 1978) that states:

> A method for establishing the distances referred to above can be based on a level of peak positive incident overpressure (designated as P_{so} in Ref.1) below which no significant damage would be expected. It is the judgement of the Staff that, for the structures, systems, and components, this level can be chosen as 1 psi (approximately 7 kPa).

During a blast event, the normal reflected pressure is twice the peak incident pressure P_{so} of 1 psi. Therefore, a normal reflected pressure of 2 psi should be considered in the analysis of the structure when considering the blast pressure loading. The following identifies the analyses used by PFS to demonstrate that the site is configured so that no credible explosion will produce a peak positive incident overpressure greater than 1 psi.

The location of the Facility is beyond the range at which cargo explosions, such as a fuel tank truck or a truck loaded with explosives traveling on Skull Valley Road, could produce peak positive incident overpressures greater than 1 psi, as per Table 4-7 of this SER. The amount of hazardous cargo and distance are those identified in Regulatory Guide 1.91.

Explosive overpressure from other accident conditions is discussed in SAR Section 8.2.4, Explosion. Potential explosions at nearby industrial, transportation, and military facilities are considered but the distance from the Facility is such that the resulting pressure will be less than 1 psi. Potential explosions of the onsite diesel fuel oil storage tanks are not considered credible because of the low volatility and high flash point of the fuel. Diesel fuel is not a flammable liquid but is classified as a Class II combustible liquid with a flash point between 100 °F and 140 °F.

The propane tanks located on the Facility are also considered in Section 8.2.4 of the SAR. PFS has proposed four 5,000 gallon propane storage tanks to be located at least 1,800 ft from the closest structures, systems, and components important to safety. The postulated explosion includes atmospheric dispersion modeling to determine the maximum downwind distance from the tank that the concentration of propane in the plume could be above the lower explosive limit and to determine the overpressure created by delayed ignition of the resulting cloud.

As identified in the SAR, the explosive overpressure design criterion is enveloped by the wind pressure loading from a tornado, 1.5 psi differential plus the dynamic wind pressure load. The dynamic wind pressure is given by $q_z = 0.00256V^2$, where q_z is in lb/ft^2 and V is in mph, as identified in ASCE 7-95. Therefore, a 240-mph wind speed relates to a dynamic wind pressure equal to 1.02 psi. To obtain the load on the structure, this value is multiplied by the wall

pressure coefficient that varies from 0.5–0.8 which gives dynamic pressure load from 0.51 to 0.82 psi. Adding the tornado differential pressure drop to the dynamic wind pressure results in a total pressure load of 2.01 to 2.32 psi. This load exceeds both the explosion peak incident and normal reflected pressure. Therefore the staff concludes that the tornado total pressure load is a bounding load with respect to the explosive overpressure load.

The staff reviewed the explosion considerations in the SAR and found that they are consistent with standard design criteria, NFPA 30 (National Fire Protection Association, 1996) and Regulatory Guide 1.91 (Nuclear Regulatory Commission, 1981) as required by 10 CFR 72.122(c).

Table 4-7. Summary of Private Fuel Storage Facility Blast Overpressures (Based on Regulatory Guide 1.91)

Condition	Trinitrotoluene (TNT) Equivalent (lb)	Distance for Pressure ≤1.0 psi (ft)	Comment
Offsite based on maximum cargo for a Highway Truck	50,000	1,658	1.9 mi to Facility (10,032 ft)
Offsite based on maximum cargo for a Railroad Car	132,000	2,291	24 mi to Facility (126,720 ft)
Offsite based on maximum cargo for a River Vessel	10,000,000	9,696	No nearby river
Offsite Space Shuttle Rocket Testing at Tekoi	1,200,000	4,782	2.3 mi to Facility (12,144 ft)
Onsite 5,000 gal. Propane Tanks (with atmospheric dispersion)	7,135	866	1,800 ft to Canister Transfer Building and nearest storage cask

Lightning

During thunderstorms, a lightning strike is possible. Therefore, structures, systems, and components located outdoors will be designed to withstand the effects of a lightning strike. The Canister Transfer Building is provided with lightning protection in accordance with the requirements of NFPA 780 (National Fire Protection Association, 1997). NFPA 780 provides for the protection of people, buildings, special occupancies, structures containing flammable liquids and gases, and other entities against lightning damage. The HI-STORM 100 storage cask is also designed for lightning protection. Any lightning strike on the cask will discharge through its

steel shell to the ground and have no adverse impact on the cask or fuel. The staff reviewed the lightning design criterion and determined that it is acceptable for the design of structures, systems, and components important to safety as required by 10 CFR 72.122(b).

Load Combinations

The load combinations identified in Table 3.6-1 of the SAR are used in the analysis of structures, systems, and components important to safety. These load combinations are based on the requirements of ANSI/ANS 57.9 (American National Standards Institute/American Nuclear Society, 1992), American Concrete Institute ACI 349-90 (American Concrete Institute, 1989), ANSI/AISC N690 (American National Standards Institute/American Institute of Steel Construction,1994), and ASME NOG–1 (American Society of Mechanical Engineers,1995a). The staff reviewed the PFS Facility documentation and determined that the load combinations design criteria are appropriately considered for the design of structures, systems, and components important to safety as required by 10 CFR 72.122(b). Appropriate combinations of the effects of normal and accident conditions and the effects on natural phenomena are considered.

Structural Design Criteria Conclusion

The structural design criteria discussed above represent the structural loads that may be present at the site. The PFS Facility structures, systems, and components that are important to safety must be designed to withstand these structural loads, as applicable. The ability of the structures, systems, and components to perform their intended safety functions under the applicable structural design loads is evaluated in Chapters 5 and 15 of this SER.

As shown in Table 4-6 of this SER, the PFS Facility site-specific structural design criteria are bounded by the applicable structural design criteria for the HI-STORM 100 Cask System, except for the seismic design criteria. Thus, except for the seismic analysis, the structural analysis presented in the HI-STORM 100 FSAR and the NRC's structural evaluation as documented in the HI-STORM 100 SER are valid for the PFS Facility. Because the seismic design loads for the PFS Facility are not enveloped by the seismic design loads for the HI-STORM 100 Cask System, the applicant performed an analysis to demonstrate that the HI-STORM 100 storage cask would perform acceptably under the site-specific design basis seismic event. This analysis is also evaluated in Chapters 5 and 15 of this SER.

The staff reviewed the PFS Facility documentation and determined that the principal design criteria, given in SAR Section 3.2, Structural and Mechanical Safety Criteria, considered for the design of structures, systems, and components are developed from appropriate site characteristics and are used in the determination of appropriate structural loads and load combination analyses. The values for these parameters form the basis for the structural design, mechanical design, and criticality assessment of the Facility.

4.1.3.3 Thermal

The staff reviewed the discussion on thermal design criteria of structures, systems, and components with respect to the following regulatory requirement:

- 10 CFR 72.120(a) requires that pursuant to the provisions of 10 CFR 72.24, an application to store spent fuel in an ISFSI must include the design criteria for the proposed storage installation. These design criteria establish the design, fabrication, construction, testing, maintenance and performance requirements for structures, systems, and components important to safety as defined in 10 CFR 72.3. The general design criteria identified in this subpart establish minimum requirements for the design criteria for an ISFSI. Any omissions in these general design criteria do not relieve the applicant from the requirement of providing the necessary safety features in the design of the ISFSI.

Thermal design criteria are based on both environmental conditions and heat generated by the materials stored.

Ambient condition design criteria are based on site-specific meteorological conditions. Additionally, an onsite meteorological measurement program has been in place. The minimum and maximum annual average and average daily maximum design temperatures identified for the site are -30, 51, and 95 °F, respectively. The design temperatures are based on data from the region, which are consistent with the values measured to date under the onsite meteorological measurement program. The staff reviewed the ambient condition loading design criteria and determined that they are acceptable because they are based on site-specific information, and the values are consistent with data from the National Oceanic and Atmospheric Administration for the region. Consequently, the ambient condition loading design criteria satisfy the requirements of 10 CFR 72.122(b).

The site-specific maximum total insolation for a 12-hour period was 684.6 W/m^2 (706.5 g cal/cm^2). Using the 30-yr database for Salt Lake City, the maximum total insolation for a 12-hr period was 730.4 W/m^2 (753.8 g cal/cm^2). The HI-STORM 100 cask has been evaluated for the solar insolation values specified in 10 CFR Part 71.71(c)(1) which are 774 W/m^2 (800 g cal/cm^2) for flat surfaces and 387 W/m^2 (400 g cal/cm^2) for curved surfaces. The Standard Review Plan for dry cask storage systems, NUREG–1536 (Nuclear Regulatory Commission, 1997) states, "The NRC staff accepts insolation presented in 10 CFR Part 71 for 10 CFR Part 72 applications. Because of the large thermal inertia of a storage cask, the values listed in 10 CFR 71.71 may be treated as the average insolence, calculated by averaging over a 24-hour day the reported 10 CFR Part 71 values for insolence over a 12-hour solar day, in a steady-state calculation." The staff concluded that both site-specific measurement and regional data are bounded by the design of the HI-STORM 100 cask.

No specific design criteria are identified in Table 3.6-1 for fire. As identified in SAR Section 8.2.5, a fire is classified as a human-induced Design Event IV as defined in ANSI/ANS 57.9 (American National Standards Institute/American Nuclear Society, 1992). The design of the Facility is such that all structures, systems, and components are located within a region covered with crushed rock with at least 100 ft to the boundary of the restricted area fence. Additionally, PFS will plant a 300 ft crested wheatgrass barrier around the restricted area. Therefore, there is no credible wildfire load on structures, systems, and components important to safety. A range of onsite fire scenarios has been evaluated. Bounding fire events are based on 50 gal. of diesel fuel in the transfer cells or cask storage pads from the cask transporter tank and 300 gal. of diesel fuel from the heavy haul vehicle tanks and vehicle tires in the cask load/unload bay. Operational restrictions are in place to ensure that these levels are not exceeded. The crushed rock fire break and operation restrictions are sufficient to limit the fire load to the identified

structures, systems, and components. The staff reviewed the fire considerations in the SAR and found that they are consistent with equipment used at the facility and operational restraints as required by 10 CFR 72.122(c). Appropriate design criteria are specified to ensure that structures, systems, and components important to safety will be designed and located so that they can perform their safety functions effectively under credible fire exposure conditions. Fire design criteria for the HI-STORM 100 cask system are based on 50 gal. of combustible transporter fuel. They are identified in the FSAR as a 1,475 °F fire with duration from 217 to 288 seconds.

The storage systems are passive and incorporate passive heat removal. Evaluation of the thermal design criteria for the cask system was carried out during licensing of the HI-STORM 100 Cask System and is documented in the NRC's HI-STORM 100 SER.

Design temperatures for various materials are identified in the SAR and are in compliance with acceptable codes. ACI 349-90 (American Concrete Institute, 1989) specifies the maximum temperature for normal operation and accident conditions. Allowable temperature for the fuel cladding is based on NUREG–1536. The Boiler and Pressure Vessel Code, ASME Section II, Part D, Table 1A specifies a design temperature for steel casks under all load conditions (American Society of Mechanical Engineers, 1999). The performance requirements for all materials, as identified by the acceptable temperature, required for compliance with 10 CFR 72.120(a) are in conformance with accepted standards. Cask-specific material properties given in the SAR are derived from the HI-STORM FSAR Table 2.2.3. The PFS Facility design temperatures are identical to those given in the HI-STORM 100 FSAR (Holtec International, 2000).

4.1.3.4 Shielding and Confinement

The staff reviewed the discussion on shielding and confinement design criteria of structures, systems, and components with respect to the following regulatory requirements:

- 10 CFR 72.104(a) requires that, during normal operations and anticipated occurrences, the annual dose equivalent to any real individual who is located beyond the controlled area be limited to 0.25 mSv (25 mrem) to the whole body, 0.75 mSv (75 mrem) to the thyroid, and 0.25 mSv (25 mrem) to any other critical organ.

- 10 CFR 72.106(a) requires that for each ISFSI, a controlled area be established.

- 10 CFR 72.122(h)(1) requires the spent fuel cladding be protected during storage against degradation that leads to gross ruptures or the fuel must be otherwise confined such that degradation of the fuel during storage will not pose operational safety problems with respect to its removal from storage.

- 10 CFR 72.126(a) requires that radiation protection systems must be provided for all areas and operations where onsite personnel may be exposed to radiation or airborne radioactive materials. Structures, systems, and components for which operation, maintenance, and required inspections may involve occupational exposure must be designed, fabricated, located, shielded, controlled, and tested

so as to control external and internal radiation exposures to personnel. The design must include means to: (1) prevent the accumulation of radioactive material in those systems requiring access; (2) decontaminate those systems to which access is required; (3) control access to areas of potential contamination or high radiation within the ISFSI; (4) measure and control contamination of areas requiring access; (5) minimize the time required to perform work in the vicinity of radioactive components (for example, by providing sufficient space for ease of operation and designing equipment for ease of repair and replacement); and (6) shield personnel from radiation exposure.

- 10 CFR 72.126(b) requires that radiological alarm systems be provided in accessible work areas as appropriate to warn operating personnel of radiation and airborne radioactive material concentrations above a given set point and of concentrations of radioactive material in effluents above control limits. Radiation alarm systems must be designed with provisions for calibration and testing their operability.

- 10 CFR 72.126(c) requires that effluent and direct radiation monitoring meet the following criteria: (1) as appropriate for the handling and storage system, effluent systems must be provided; and (2) areas containing radioactive materials must be provided with systems for measuring the direct radiation levels in and around these areas.

- 10 CFR 72.126(d) requires that the ISFSI be designed to provide means to limit to ALARA levels the release of radioactive materials in effluents during normal operations; and control the release of radioactive materials under accident conditions. Analyses must be made to show that releases to the general environment during normal operations and anticipated occurrences will be within the exposure limit given in 10 CFR 72.104. Analyses of design basis accidents must be made to show that releases to the general environment will be within the exposure limits given in 10 CFR 72.106. Systems designed to monitor the release of radioactive materials must have means for calibration and testing their operability.

- 10 CFR 72.128(a) requires that spent fuel storage, high-level radioactive waste storage, and other systems that might contain or handle radioactive materials associated with spent fuel or high-level radioactive waste, be designed to ensure adequate safety under normal and accident conditions. These systems must be designed with: (1) a capability to test and monitor components important to safety, (2) suitable shielding for radioactive protection under normal and accident conditions, (3) confinement structures and systems, (4) a heat-removal capability having the stability and reliability consistent with its importance to safety, and (5) means to minimize the quantity of radioactive wastes generated.

- 10 CFR 72.128(b) requires that radioactive waste treatment facilities be provided. Provisions must be made for the packing of site-generated, low-level wastes in a form suitable for storage onsite awaiting transfer to disposal sites.

Criteria used in the design of cask radiological protection features and confinement design of the cask systems are provided in the SAR and the HI-STORM 100 FSAR and are summarized in Tables 4-9 and 4-10 of this SER. The basic concept for the PFS Facility shielding and confinement system is protection by multiple barriers and systems, as required by 10 CFR 72.126(a)–(c). The use of the HI-STORM 100 Cask System, which is a sealed canister-based system, satisfies the requirements of 10 CFR 72.122(h)(1). Operating procedures, shielding design, and access controls provide the necessary radiological protection to ensure radiological exposures to facility personnel and the public are ALARA as required by 10 CFR 72.126(d).

Table 4-9. Summary of Private Fuel Storage Facility Design Criteria—Radiation Protection/Shielding Design (Based on SAR Table 3.6-1)

Design Parameters	Design Conditions		Reference
Storage Systems Design Dose Rate Limits	Cask Side Surface Cask Inlet/Exit Vent Area Cask Top Surface	40 mrem/hr 60 mrem/hr 10 mrem/hr	HI-STORM 100 FSAR, Section 2.3.5.2
Individual Workers Dose Rate	Total effective dose equivalent Dose to eye lens Dose to skin and extremities	5 rem/yr 15 rem/yr 50 rem/yr	10 CFR 20.1201
Restricted Area Boundary Dose Rate	2 mrem/hr, maximum		10 CFR 20.1301
Owner-Controlled Area Boundary Dose Rate	25 mrem/yr whole body and 75 mrem/yr thyroid, maximum 25 mrem/yr to any other critical organ 5 rem accident dose or 50 rem total organ dose equivalent (one time)		10 CFR 72.104 10 CFR 72.106

The bounding dose rate design criteria are consistent with the requirements of 10 CFR 72.104(a) and 72.106(a).

Table 4-10. Summary of Private Fuel Storage Facility Design Criteria—Confinement Design (Based on SAR Table 3.6-1)

Design Parameters	Design Conditions	Reference
Confinement Method	Welded closed steel canister	HI-STORM 100 FSAR, 2.3.2.1
Confinement Barrier Design	HI-STORM canister: ASME III, NB	HI-STORM 100 FSAR, 2.3.2.1

The staff reviewed the design criteria for spent nuclear fuel storage and handling, and determined that they are appropriately identified as required by 10 CFR 72.128(a) and (b). A shielding evaluation has been performed in Chapter 7 of this SER, a confinement evaluation

has been performed in Chapter 9 of this SER, and a radiation protection evaluation has been performed in Chapter 11 of this SER. Evaluation findings given in this chapter are drawn from Chapters 7, 9, and 11 of this SER.

4.1.3.5 Criticality

The staff reviewed the discussion on criticality design criteria of structures, systems, and components with respect to the following regulatory requirements:

- 10 CFR 72.124(a) requires that spent fuel handling, packaging, transfer, and storage systems be designed to be maintained subcritical and to ensure that, before a nuclear criticality accident is possible, at least two unlikely, independent, and concurrent or sequential changes have occurred in the conditions essential to nuclear criticality safety. The design of handling, packaging, transfer, and storage systems must include margins of safety for the nuclear criticality parameters that are commensurate with the uncertainties in the data and methods used in calculations and demonstrate safety for the handling, packaging, transfer, and storage conditions and in the nature of the immediate environment under accident conditions.

- 10 CFR 72.124(b) requires that when practicable the design of an ISFSI be based on favorable geometry, permanently fixed neutron absorbing materials (poisons), or both. Where solid neutron absorbing materials are used, the design shall provide for positive means to verify their continued efficacy.

- 10 CFR 72.124(c) requires that a criticality monitoring system be maintained in each area where special nuclear material is handled, used, or stored which will energize clearly audible alarm signals if accidental criticality occurs. Monitoring of dry storage areas where special nuclear material is packaged in its stored configuration under a license issued under Part 72 is not required.

Criteria used in criticality design of the cask systems are provided in the PFS Facility SAR and the HI-STORM 100 FSAR and are summarized in Table 4-11 of this SER. The staff's criticality evaluation is discussed in Chapter 8 of this SER. The design criteria for criticality are identified in the SAR as required by 10 CFR 72.124(a) through (c).

Table 4-11. Summary of Private Fuel Storage Facility Design Criteria—Criticality Design

Design Parameters	Design Conditions	Applicable Criteria and Codes
Control Method	Geometry of fuel assemblies assuming no moderator	HI-STORM 100 FSAR, Section 2.3.4.1
k_{eff}	≤ 0.95	NUREG–1567

4.1.3.6 Decommissioning

The staff's decommissioning evaluation is presented in Chapter 13 of this SER.

4.1.3.7 Retrieval

The staff reviewed the discussion on retrieval design criteria of structures, systems, and components with respect to the following regulatory requirements:

- 10 CFR 72.122(l) requires that storage systems be designed to allow ready retrieval of spent fuel for further processing or disposal.

- 10 CFR 72.128(a) requires that spent fuel storage, and other systems that might contain or handle radioactive materials associated with spent fuel or high-level radioactive waste, be designed to ensure adequate safety under normal and accident conditions. These systems must be designed with: (1) a capability to test and monitor components important to safety, (2) suitable shielding for radioactive protection under normal and accident conditions, (3) confinement structures and systems, (4) a heat-removal capability having the stability and reliability consistent with its importance to safety, and (5) means to minimize the quantity of radioactive wastes generated.

The spent fuel will be stored in and handled with the HI-STORM 100 Cask System which has been approved for use under the general license provisions of 10 CFR Part 72. As discussed in the HI-STORM 100 FSAR and the staff's related SER, the HI-STORM 100 Cask System is designed to ensure adequate safety and to protect fuel integrity and retrievability under the design basis loads specified in the HI-STORM 100 FSAR. The design basis loads considered in the HI-STORM 100 FSAR bound the structural and thermal loads found at the Facility except for the seismic load (see SER Sections 4.1.3.2 and 4.1.3.3). For the seismic event, the applicant provided an analysis which demonstrated that the HI-STORM 100 storage cask would neither tipover nor slide during a site-specific seismic event. Further, the loads on the canister would remain bounded by the canister loads considered in the HI-STORM 100 FSAR. (The seismic analysis is evaluated in Chapters 5 and 15 of this SER.)

Based on the foregoing discussion, there is reasonable assurance that the HI-STORM 100 Cask System will provide adequate safety and maintain fuel retrievability under the PFS Facility site-specific conditions. Therefore, the staff finds that the requirements of 10 CFR 72.122(l) and 72.128(a) are satisfied.

4.1.4 Design Criteria for Other Structures, Systems, and Components

No specific requirements are identified in 10 CFR Part 72 for other structures, systems, and components not important to safety. Therefore, no evaluation findings are made in this section; only discussion of the information provided in the SAR is given. The design criteria for structures, systems, and components classified as not important to safety, but which have security or operational importance, are addressed in SAR Sections 4.3, 4.5.4, 4.5.5, and 4.7.5. The SAR specifies that these structures, systems, and components will be designed to comply

with their applicable codes and standards to maintain the capability to mitigate the effects of off-normal or accident events.

4.2 Evaluation Findings

Based on the review of the information presented in the SAR, the following evaluation findings are made regarding the proposed PFS Facility ISFSI:

- The staff finds that the materials to be stored at the Facility are appropriately identified as those that are approved for storage in the HI-STORM 100 Cask System.

- The staff finds that the structures, systems, and components important to safety have been properly classified and their associated categories are consistent with the regulatory requirements of 10 CFR 72.144(a) and associated technical information content of the application, in accordance with 72.24(n). This list of structures, systems, and components is based on the definition in 10 CFR 72.3 of structures, systems, and components important to safety. The SAR appropriately specifies the design criteria for the structures, systems, and components important to safety in accordance with 10 CFR 72.120(a). The design criteria are to be included in the quality assurance procedures, as required in 10 CFR 72.144(a).

- The staff finds that the structural design criteria, given in SAR Section 3.2, Structural and Mechanical Safety Criteria, considered for the structures, systems, and components important to safety, are developed from site characteristics and are used in the determination of structural loads and load combination analyses. The values for these parameters form the basis for the structural design, mechanical design, and criticality assessment of the Facility. These design criteria satisfy the requirements of 10 CFR 72.120(a), and 72.122(h). Additionally, the structures, systems, and components important to safety will be designed to quality standards commensurate with important to safety functions performed to satisfy the requirements of 10 CFR 72.144(c).

- The staff finds that the seismic design criteria are appropriately identified in accordance with 10 CFR 72.120(a) and 72.122(b). The seismic design criteria are in accordance with the site-specific seismic hazards analysis given in SAR Chapter 2, Site Characteristics. The applicant has demonstrated that an exemption to the requirements of 10 CFR 72.120(f) is acceptable.

- The staff finds that the explosion considerations in the SAR are consistent with standard design criteria specified by Regulatory Guide 1.91,as required by 10 CFR 72.122(c). No credible onsite or offsite explosions will result in a peak positive incident overpressure of greater than 1.0 psi (the threshold air overpressure specified in Regulatory Guide 1.91) for structures, systems, and components important to safety.

- The staff finds that the load combinations design criteria are adequately considered for the design of structures, systems, and components, as required by 10 CFR 72.122(b). Appropriate combinations of the effects of normal and accident conditions, and the effects of natural phenomena have been considered.

- The staff finds that the bounding dose rate design criteria given in the SAR are consistent with the requirements of 10 CFR 72.104(a). The design criteria for spent nuclear fuel storage and handling have been properly specified, as required by 10 CFR 72.128. A shielding evaluation has been performed in Chapter 7 of this SER. A confinement evaluation has been performed in Chapter 9 of this SER. A radiation protection evaluation has been performed in Chapter 11 of this SER.

- The staff finds that design criteria for criticality are identified in the SAR, as required by 10 CFR 72.124(a)–(c). A criticality evaluation has been performed in Chapter 8 of this SER.

- The staff's decommissioning findings are discussed in Chapter 13 of this SER.

- The staff finds that the Facility design, which includes use of the HI-STORM 100 Cask System, allows for retrieval of the spent nuclear fuel in accordance with 10 CFR 72.122(l). Storage systems are designed to ensure adequate safety during normal and accident conditions in accordance with 10 CFR 72.128(a).

4.3 References

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5 INSTALLATION AND STRUCTURAL EVALUATION

5.1 Conduct of Review

This chapter of the SER contains reviews of the information presented in Chapter 4 of the SAR which identifies the Facility Design. The review also considers selected sections and documents referenced in Chapters 1, 2, 3, and 8 of the SAR. Respectively, these chapters discuss general information, the site characteristics, the principal design criteria, and the accident analysis.

The objective of the installation design review is to ensure compliance with the required site features and to support other evaluation areas. The objective of the structural evaluation review is to ensure the structural integrity of structures, systems, and components with emphasis on those that are important to safety.

Spent fuel dry storage facilities are designed for safe confinement and storage of the spent nuclear fuel. The major categories of safety protection systems discussed in the following sections include: confinement structures, systems, and components; reinforced concrete structures; other structures, systems, and components important to safety; and other structures, systems, and components not important to safety.

The design of the proposed Facility is based on the use of the HI-STORM 100 Cask System, which has been reviewed by the NRC and approved for general use under Certificate of Compliance No. 1014 (Nuclear Regulatory Commission, 2000a). The Facility relies on the HI-STORM 100 Cask System, as described in the HI-STORM 100 FSAR (Holtec International, 2000), for confinement and radiological safety. Where applicable, the staff relied on the review carried out during the certification process of the cask system, as documented in the NRC's HI-STORM 100 SER (Nuclear Regulatory Commission, 2000b).

The staff reviewed the Facility installation and structural evaluation with respect to the following regulatory requirements:

- 10 CFR 72.24(a) requires a description and safety assessment of the site on which the ISFSI is to be located, with appropriate attention to the design bases for external events. Such assessment must contain an analysis and evaluation of the major structures, systems, and components of the ISFSI that bear on the suitability of the site when the ISFSI is operated at its design capacity.

- 10 CFR 72.24(b) requires a description and discussion of the ISFSI structures with special attention to design and operating characteristics, unusual or novel design features, and principal safety considerations.

- 10 CFR 72.24(c)(1) requires that the design of the ISFSI be described in sufficient detail to support the findings in Section 72.40, including the design criteria for the ISFSI pursuant to subpart F of this part, with identification and justification for any additions to or departures from the general design criteria.

- 10 CFR 72.24(c)(2) requires that the design of the ISFSI be described in sufficient detail to support the findings in Section 72.40, including the design bases and the relation of the design bases to the design criteria.

- 10 CFR 72.24(c)(3) requires that the design of the ISFSI be provided in sufficient detail to support the findings in Section 72.40, including information relative to materials of construction, general arrangement, dimensions of principal structures, and descriptions of all structures, systems, and components important to safety, in sufficient detail to support a finding that the ISFSI will satisfy the design bases with an adequate margin for safety.

- 10 CFR 72.24(c)(4) requires that the design of the ISFSI be described in sufficient detail to support the findings in Section 72.40, including applicable codes and standards.

- 10 CFR 72.24(d)(1) requires that an analysis and evaluation be provided of the design and performance of structures, systems, and components important to safety, with the objective of assessing the impact on public health and safety resulting from operation of the ISFSI and including determination of the margins of safety during normal operations and expected operational occurrences during the life of the ISFSI.

- 10 CFR 72.24(d)(2) requires that an analysis and evaluation be provided of the design and performance of structures, systems, and components important to safety, with the objective of assessing the impact on public health and safety resulting from operation of the ISFSI and including determination of the adequacy of structures, systems, and components provided for the prevention of accidents and the mitigation of the consequences of accidents, including natural and manmade phenomena and events.

- 10 CFR 72.24(i) requires the identification of any structures, systems, or components important to safety whose functional adequacy or reliability have not been demonstrated by prior use for that purpose or cannot be demonstrated by reference to performance data in related applications or to widely accepted engineering principles, along with a schedule showing how safety questions will be resolved prior to the initial receipt of spent fuel or high-level radioactive waste for storage at the ISFSI.

- 10 CFR 72.120(a) requires that, pursuant to the provisions of Section 72.24, an application to store spent fuel in an ISFSI include the design criteria for the proposed storage installation. These design criteria establish the design, fabrication, construction, testing, maintenance and performance requirements for structures, systems, and components important to safety as defined in Section 72.3. The general design criteria identified in this subpart establish minimum requirements for the design criteria for an ISFSI. Any omissions in these general design criteria do not relieve the applicant from the requirement of providing the necessary safety features in the design of the ISFSI.

- 10 CFR 72.122(a) requires that structures, systems, and components important to safety be designed, fabricated, erected, and tested to quality standards commensurate with the importance to safety of the function to be performed.

- 10 CFR 72.122(b)(1) requires that structures, systems, and components important to safety be designed to accommodate the effects of, and to be compatible with, site characteristics and environmental conditions associated with normal operation, maintenance, and testing of the ISFSI and to withstand postulated accidents.

- 10 CFR 72.122(b)(2) requires that structures, systems, and components important to safety be designed to withstand the effects of natural phenomena such as earthquakes, tornadoes, lightning, hurricanes, floods, tsunami, and seiches, without impairing their capability to perform safety functions. The design bases for these structures, systems, and components must reflect: (i) Appropriate consideration of the most severe of the natural phenomena reported for the site and surrounding area, with appropriate margins to take into account the limitations of the data and the period of time in which the data have accumulated, and (ii) Appropriate combinations of the effects of normal and accident conditions and the effects of natural phenomena. The ISFSI should also be designed to prevent massive collapse of building structures or the dropping of heavy objects as a result of building structural failure on the spent fuel or onto structures, systems, and components important to safety.

- 10 CFR 72.122(c) requires that structures, systems, and components important to safety be designed and located so that they can continue to perform their safety functions effectively under credible fire and explosion exposure conditions. Noncombustible and heat-resistant materials must be used wherever practical throughout the ISFSI, particularly in locations vital to the control of radioactive materials and to the maintenance of safety control functions. The design of the ISFSI must include provisions to protect against adverse effects that might result from either the operation or the failure of the fire suppression system.

- 10 CFR 72.122(f) requires that systems and components that are important to safety be designed to permit inspection, maintenance, and testing.

- 10 CFR 72.122(g) requires that structures, systems, and components important to safety be designed for emergencies. The design must provide for accessibility to the equipment of onsite and available offsite emergency facilities and services such as hospitals, fire and police departments, ambulance service, and other emergency agencies.

- 10 CFR 72.122(h)(1) requires that the spent fuel cladding be protected during storage against degradation that leads to gross ruptures or the fuel must be otherwise confined such that degradation of the fuel during storage will not pose operational safety problems with respect to its removal from storage.

- 10 CFR 72.122(h)(4) requires that the storage confinement systems have the capability for monitoring in a manner such that the licensee will be able to

determine when corrective action needs to be taken to maintain safe storage conditions.

- 10 CFR 72.122(l) requires that storage systems be designed to allow ready retrieval of spent fuel for further processing or disposal.

- 10 CFR 72.128(a) requires that spent fuel storage and other systems that might contain or handle radioactive materials associated with spent fuel, be designed to ensure adequate safety under normal and accident conditions. These systems must be designed with: (1) a capability to test and monitor components important to safety; (2) suitable shielding for radioactive protection under normal and accident conditions; (3) confinement structures and systems; (4) a heat-removal capability having testability and reliability consistent with its importance to safety; and (5) means to minimize the quantity of radioactive wastes generated.

- 10 CFR 72.128(b) requires that radioactive waste treatment facilities be provided. Provisions must be made for the packing of site-generated low-level wastes in a form suitable for storage onsite awaiting transfer to disposal sites.

- 10 CFR 72.236(b) requires that design bases and design criteria be provided for structures, systems, and components important to safety.

- 10 CFR 72.236(c) requires that the cask be designed and fabricated so that the spent fuel is maintained in a subcritical condition under credible conditions.

- 10 CFR 72.236(e) requires that the spent fuel storage cask be designed to provide redundant sealing of confinement systems.

- 10 CFR 72.236(f) requires that the spent fuel storage cask be designed to provide adequate heat removal capacity without active cooling systems.

- 10 CFR 72.236(g) requires that the spent fuel storage cask be designed to store the spent fuel safely for a minimum of 20 years and permit maintenance as required.

- 10 CFR 72.236(l) requires that the spent fuel storage cask and its systems important to safety be evaluated, by appropriate tests or by other means acceptable to the Commission, to demonstrate that they will reasonably maintain confinement of radioactive material under normal, off-normal, and credible accident conditions.

5.1.1 Confinement Structures, Systems, and Components

The discussion on confinement structures, systems, and components is presented in Section 4.1, Summary Description, of the SAR and in Chapter 4 of the HI-STORM 100 FSAR. The staff reviewed the discussion on confinement structures, systems, and components with respect to

the applicable regulatory requirements of 10 CFR 72.24, 72.106, 72.122, 72.128, and 72.236 as discussed below in details.

5.1.1.1 Description of Confinement Structures

The Facility's confinement structure is the spent nuclear fuel canister, specifically the MPC component of the HI-STORM 100 Cask System. A detailed description of the MPC is provided in the HI-STORM 100 FSAR. The staff has previously reviewed and found this description acceptable, as documented in the staff's HI-STORM 100 SER. Sections 4.2.1.2.2 and 4.2.1.5.5 of the PFS Facility SAR reference the HI-STORM 100 FSAR and provide a summary description of the confinement structure. The staff finds the summary to be consistent with the information in the HI-STORM 100 FSAR. The confinement structure has been sufficiently described in accordance with 10 CFR 72.24.

5.1.1.2 Design Criteria for Confinement Structures

The spent nuclear fuel is contained within the MPC. The design criteria for the MPC are presented in the HI-STORM 100 FSAR and evaluated in the staff's related SER.

5.1.1.3 Material Properties for Confinement Structures

The structural components of the MPC are made of stainless steel. Material properties for confinement structures of the MPC are presented in the HI-STORM 100 FSAR and evaluated in the staff's HI-STORM 100 SER.

5.1.1.4 Structural Analysis for Confinement Structures

The staff reviewed the information presented on structural analysis in SAR Section 4.2.1, HI-STORM 100 Cask System. The detailed structural analysis of confinement structures is presented in the HI-STORM 100 FSAR. The staff has previously reviewed this structural analysis and found it acceptable, as documented in the staff's HI-STORM 100 SER. As documented in that SER, the structural analysis shows that the structural integrity of the HI-STORM 100 Cask System is maintained under all credible loads. Based on the results presented in the HI-STORM 100 FSAR, the stresses in the MPC under the most critical load combinations are less than the allowable stresses for the MPC materials.

A discussion of the MPC design relative to the storage requirements of the PFS Facility is in SAR Chapter 4, Facility Design. The PFS Facility SAR provides a summary of the analysis performed in the HI-STORM 100 FSAR. The PFS Facility SAR states that for the following loads and combined loading conditions, the spent fuel canister is shown to be within allowable limits of American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code, Section III (American Society of Mechanical Engineers, 1998) and, therefore, meets the Facility design criteria given in SAR Section 3.2, Structural and Mechanical Safety Criteria:

- Dead loads (D)
- Live loads (L)
- Internal (P_i) and external (P_o) pressure loads
- Temperature gradients (T)

- Deceleration loads (A)
- Design tornado wind loads (W_t)
- Tornado-generated missile loads (M)
- Probable maximum flood loads (F_a)
- Explosion pressure loads (E*)
- Thermal loads due to fire (T*)6
- Lightning

The loading conditions at the Facility are enveloped by the loading conditions considered in the HI-STORM 100 FSAR, except for the seismic loads. A structural analysis was performed for the HI-STORM 100 Cask System considering the PFS Facility site-specific seismic loads (Holtec International, 2001a). The analysis shows that the storage system will withstand the imposed loads and not tipover or slide into contact with an adjacent cask when subjected to the PFS Facility site-specific seismic event. The basis for the conclusions for cask stability under the site-specific load is in Section 5.1.4.4 of this SER. Although the Facility site-specific seismic loads are higher than the seismic loads considered in the HI-STORM 100 FSAR, resulting loads on the MPC and fuel assemblies remain bounded by the loads considered in the HI-STORM 100 FSAR. An evaluation was performed for the HI-STORM 100 Cask System for thermal loads (Holtec International, 1999, 2001b) which supports the conclusion that the resulting loads on the MPC and fuel assemblies remain bounded by the loads considered in the HI-STORM 100 FSAR. Therefore, the staff's conclusions in its HI-STORM 100 SER with respect to the structural integrity of the MPC are valid for the PFS Facility.

As demonstrated in the HI-STORM 100 FSAR and as documented in the staff's HI-STORM 100 SER, the cask is stable and will not tipover in the event of tornado winds with concurrent impact of the tornado-driven design missile (an automobile) at the top of the storage cask. Additionally, the design of the 36-inch-thick reinforced concrete pad at the Facility is based on the use of concrete with a compressive strength of 3,000-psi (at 28 days), reinforcing steel having 60,000-psi yield strength. Concrete for the PFS Facility storage pad has a lower compressive strength than the 4,200-psi concrete (at 28 days) assumed for the reference target ISFSI pad in the HI-STORM 100 FSAR. Also, the PFS soil foundation for the pad consists of soil and soil-cement layers with properties that are different than the soil foundation considered in the HI-STORM 100 FSAR. Therefore, a PFS site specific analysis was performed (Holtec International, 2001c, d) which concluded that, a non-credible, hypothetical tipover of a HI-STORM 100 storage cask at the PFS Facility would result in a deceleration of less than 45g and lower stresses in the MPC than those evaluated in the HI-STORM 100 FSAR. Therefore, the staff's conclusions in its HI-STORM 100 SER with respect to the structural integrity of the MPC are valid for the PFS Facility.

5.1.2 Pool and Pool Confinement Facilities

The PFS Facility will not have a pool or pool confinement facility.

5.1.3 Reinforced Concrete Structures

This section contains a review of SAR Sections 4.1, Summary Description, 4.2.3, Cask Storage Pads, and 4.7.1, Canister Transfer Building. The staff reviewed the discussion on reinforced

concrete structures important to safety with respect to the regulatory requirements of 10 CFR 72.24 (a) and (b), 72.106 (b), 72.122 (b) and (f), and 72.236 (f) and (g).

5.1.3.1 Description of Reinforced Concrete Structures

There are two reinforced concrete structures in the Facility that have been classified as important to safety:

- Canister Transfer Building (QA Category B)
- Cask storage pad (QA Category C)

The staff reviewed the description of reinforced concrete structures important to safety with respect to the regulatory requirements of 10 CFR 72.24(a) and (b), 72.122(f), (g) and (l), and 72.128(a).

Canister Transfer Building

As identified in Section 4.7.1 of the SAR, the Canister Transfer Building provides physical protection and shielding of the canisters during transfer from the transportation cask to the storage cask. The Canister Transfer Building consists of the shipping cask loading/unloading bays, canister transfer cells, a 200/25-ton overhead bridge crane, a 150/25-ton semi-gantry crane, crane runway girders and their supports, cask transporter bay, tornado-missile barriers, a low-level waste storage room, radiation shield walls and doors, equipment lay-down areas, storage cask delivery and staging platform, mechanical and electrical equipment areas, personnel offices, and restroom areas. Figure 4.7-1 of the SAR illustrates the layout of the Canister Transfer Building. The SAR provides a design description of the Canister Transfer Building in sufficient detail to support a detailed review and evaluation. Consequently, the requirements of 10 CFR 72.24(a) and (b) have been satisfied.

The Canister Transfer Building is a massive reinforced concrete structure with a slab on grade designed in accordance with American Concrete Institute (ACI) 349-90 (American Concrete Institute, 1989). The ACI 349-90 Code provides the minimum requirements for the design and construction of nuclear safety-related concrete structures and structural elements for nuclear power generating stations. The thicknesses of the slab, wall, and roof members were initially sized based on shielding requirements. The calculated thickness was then checked for penetration resistance to the Spectrum II tornado-driven missiles as identified in Section 3.5.1.4 of NUREG–0800 (Nuclear Regulatory Commission,1981). The thickness was shown to be adequate to resist penetration and damage that would subsequently affect performance of the Canister Transfer Building (Stone & Webster Engineering Corporation, 1998b). The reinforced concrete roof will be supported by structural steel elements, designed in accordance with the allowable stress design method of ANSI/AISC N690–1994 (American National Standards Institute/American Institute of Steel Construction, 1994).

Analyses were presented for the site-specific seismic loading conditions (Stone & Webster Engineering Corporation, 1998 a, b, c). The size and placement of the reinforcing steel were based on the results of the seismic analysis. Beams and columns were sized to provide support under all analyzed loading conditions and combinations (Stone & Webster Engineering

Corporation, 1998c, 2001i). Details of the analysis of the Canister Transfer Building are presented in Section 5.1.3.4 of this SER.

Sections 4.7, 4.7.2.1, 5.1.4.7, 5.1.6.5, and 9.2.2 of the SAR provide descriptions of the Canister Transfer Building and associated conduct of operational procedures. Components requiring inspection, testing, and maintenance are identified and adequately described in accordance with 10 CFR 72.122(f). Pre-operational, startup, and operational tests will be performed to verify the functional operations of structures, systems, and components important to safety. Design in accordance with ACI 349-90 and ANSI/AISC N690–1994 addresses these topics. Design of the Canister Transfer Building allows for access to all locations and regions in the event of emergencies. The design is an open structure with door ways and access corridors provided in accordance with the requirements of 10 CFR 72.122(g).

The design of the Canister Transfer Building allows for handling and storage of the limited radioactive waste generated at the Facility within the low-level waste storage room in accordance with the requirements of 10 CFR 72.128(a). A waste confinement and management evaluation is contained in Chapter 14 of this SER.

Cask Storage Pads

Information on the cask storage pad design and analysis is given in Section 4.2.3 of the SAR. The cask storage pads are independent structural units constructed of reinforced concrete, designed in accordance with ACI 349-90. Each pad is 30 ft × 67ft × 3 ft and is capable of supporting eight loaded HI-STORM 100 storage casks. Figure 4.2-7 of the SAR shows the general layout of the storage pads. The size of the pad is based on a 15-ft center-to-center spacing of the storage casks in the 30 ft width (transverse direction) and 16-ft center-to-center in the 67 ft length (longitudinal direction) of the cask storage pads. The cask storage pad is a conventional cast-in-place reinforced concrete mat foundation structure. It provides a level and stable surface for placement and storage of the storage casks. The cask storage pad design is based on the maximum loaded weight of a storage cask of 360,000 lb, the weight of the HI-STORM 100 storage cask loaded with either MPC-24 or MPC-68 canisters. The SAR provides a design description of the cask storage pads in sufficient detail to support a detailed review and evaluation. Consequently, the requirements of 10 CFR 72.24(a) and (b) have been satisfied.

Inspection and maintenance operations are identified in Sections 5.1.4.7 and 5.1.6.5 of the SAR with additional details provided in Section 9.2.2 of the SAR. ACI 349–90 (American Concrete Institute, 1989) specifies the inspection requirements during the construction of the cask storage pads. The storage casks are passive systems so the necessary inspection and maintenance include only the temperature monitoring system. Pre-operational, startup, and operational tests will be performed to verify the functional operations of structures, systems, and components important to safety. Description of the cask storage pads and associated operations procedures include consideration of inspection, maintenance, and testing as required in 10 CFR 72.122(f). The design of the reinforced concrete pads, a simple slab on grade, provides for access to all locations and allows for access to the storage casks in the event of emergencies. There are no barriers built into the cask storage pads that would prevent access to any location on the pads adjacent to the storage casks. This design allows for emergency capability, as required in 10 CFR 72.122(g). This simple slab on grade concept

also incorporates the capability for retrieving the spent nuclear fuel canisters. The cask transporter can drive onto the pad to access any storage cask and transport it back to the Canister Transfer Building. Settlement of the pad has also been taken into account, as discussed in Section 2.1.6.4 of this SER. The requirements of 10 CFR 72.122(l), therefore, are satisfied.

5.1.3.2 Design Criteria for Reinforced Concrete Structures

The design bases for the reinforced concrete structures are given in SAR Section 4.2.3, Cask Storage Pad, and SAR Section 4.7.1, Canister Transfer Building. Table 4.1 of the SAR identifies details of the Facility's compliance with the general design criteria of 10 CFR Part 72, Subpart F. The staff reviewed the design criteria for reinforced concrete structures with respect to the regulatory requirements of 10 CFR 72.24(c), 72.120(a), 72.122(b)(1) and (b)(2), 72.122(c), (f), (g) and (l).

Canister Transfer Building

As identified in Section 4.7.1.1 of the SAR, the Canister Transfer Building is designed in accordance with the design criteria contained in Chapter 3 of the SAR. This conclusion is supported by the structural analysis performed as described in Section 5.1.3.4 of this SER. Design criteria have been shown in Chapter 4 of this SER to be representative of the site.

The design criteria for the Canister Transfer Building, supported by the requirement of ACI 349-90, establish the minimum design, fabrication, construction, testing, maintenance, and performance requirements for this reinforced concrete structure important to safety. Additionally, the design criteria address the site characteristics and environmental conditions during normal operations and during postulated off-normal and accident events. Further, PFS uses National Fire Protection Association (NFPA) codes and standards and the UBC as appropriate design bases for fire protection of the Canister Transfer Facility. Additionally, Regulatory Guide 1.91 (Nuclear Regulatory Commission, 1978) has been taken as the appropriate design basis for explosive air overpressure protection. A complete discussion of the design criteria applicable to the Canister Transfer Building is given in Section 4.1.3 of this SER. The conclusions in this section regarding the Canister Transfer Building design criteria are based on the evaluation findings in Section 4.1.3 of this SER.

Cask Storage Pads

As identified in Section 4.2.3 of the SAR, the cask storage pads are designed in accordance with the design criteria contained in Chapter 3 of the SAR. This conclusion is also supported by the structural analysis described in Section 5.1.3.4 of this SER. Design criteria for the storage pad have been shown in Chapter 4 of this SER to be representative of the site.

Cask storage pads are designed in accordance with ultimate strength design methods specified in ACI 349-90 (American Concrete Institute, 1989) with the load combinations specified in ANSI/ANS 57.9 (American National Standards Institute/American Nuclear Society, 1992). The ACI 349-90 Code specifies the minimum requirements for the design and construction of nuclear safety-related concrete structures and structural elements for nuclear power generating stations. Additionally, ANSI/ANS 57.9 establishes design criteria for an ISFSI. PFS also used

EPRI NP–7551 (Electric Power Research Institute, 1991) to calculate the target hardness of the storage pads. EPRI NP–7551 deals with the structural design of concrete pads for storing spent fuel casks.

The design criteria for the cask storage pads establish the minimum design, fabrication, construction, testing, maintenance, and performance requirements for reinforced concrete storage pads. Additionally, the design criteria of the storage pads address site characteristics and environmental conditions under normal operations and under postulated off-normal and accident events. An evaluation of the design criteria applicable to the cask storage pad has been presented in Section 4.1.3 of this SER. The conclusions drawn in this section on the storage pad design criteria are based on the evaluation findings made in Section 4.1.3 of this SER.

5.1.3.3 Material Properties for Reinforced Concrete Structures

The staff reviewed the material properties for reinforced concrete structures with respect to the regulatory requirements of 10 CFR 72.24(c)(3) and (4).

Canister Transfer Building

The staff has reviewed SAR Section 4.7.1, Canister Transfer Building. The Canister Transfer Building will be constructed of poured-in-place normal weight concrete with a minimum 28-day compressive strength of 4,000 psi following ACI 349-90 (American Concrete Institute, 1989). The concrete will be reinforced with Grade 60 deformed bars (Stone & Webster Engineering Corporation, 1998e) following American Society for Testing and Materials (1990). Because PFS has adequately identified the properties of materials to be used in reinforced concrete of the Canister Transfer Building, the requirements of 10 CFR 72.24 (c)(3) have been satisfied. These materials are used by the construction industry and will be in accordance with ACI 349-90. Therefore, the requirements of 10 CFR 72.24(c)(4) have been satisfied.

Cask Storage Pads

The staff has reviewed SAR Section 4.2.3, Cask Storage Pads. Materials of construction of the cask storage pads, as identified in Section 4.2.3.4 of the SAR include concrete with a minimum 28-day compressive strength of 3,000 psi following ACI 349-90 (American Concrete Institute, 1989) and reinforcing steel with a minimum yield strength of 60,000 psi following American Society for Testing and Materials (1990). These materials are used by the construction industry. For assessing potential impact of storage cask drop and tipover on the cask storage pad, the stiffness of the concrete pad is a critical factor. As the stiffness of the pad is increased by increasing the compressive strength of the concrete or the pad thickness, the resulting deceleration loads on the dropped cask will increase. Therefore, the compressive strength of the concrete to be used in constructing the storage pads must be kept below 4,200 psi, the value used in the analysis carried out by the cask-vendor, Holtec International, for the HI-STORM 100 Cask System. Based on the review of information presented by PFS, the staff concludes that materials to be used to construct the cask storage pads have been adequately identified. Consequently, the requirements of 10 CFR 72.24 (c)(3) have been satisfied. The applicant has identified the appropriate codes and standards for the cask storage pads and, therefore, the requirements of 10 CFR 72.24(c)(4) have been satisfied.

5.1.3.4 Structural Analysis for Reinforced Concrete Structures

The staff reviewed the structural analysis for reinforced concrete structures with respect to the regulatory requirements of 10 CFR 72.24(c)(2) and (c)(4), 72.122(b)(1), (b)(2), and (c), and 72.128(a).

The Facility reinforced concrete structures, as described in the SAR (Private Fuel Storage Limited Liability Company, 2001), are designed to meet the requirements of ACI 349-90 (American Concrete Institute, 1989) and will be constructed to ACI 318-95 requirements (American Concrete Institute, 1995). The staff accepts the strength design method, as presented in the ACI 349-90, for concrete structures important to safety. Reinforced concrete structures were designed and analyzed to resist the loads and load combinations specified. Static analysis methods determined forces and moments on the structural members as a result of applied service loading conditions. Dynamic analysis methods determined structural member forces and moments for factored loading conditions where structural components were subjected to seismic or tornado-generated missile impact loads.

The reinforced concrete structures important to safety were analyzed for normal, off-normal, and accident loading conditions. These analyses were carried out to ensure that they would be able to perform their intended safety functions under the extreme environmental and natural phenomena as specified in 10 CFR 72.122(b)(1) and (b)(2) and ANSI/ANS 57.9 (American National Standards Institute/American Nuclear Society, 1992). The ultimate strength method of analysis is used with the appropriate load factors for the following loads:

- Dead loads (D)
- Live loads (L)
- Soil pressure loads (H)
- Temperature gradients (T)
- Wind loads (W)
- Earthquake loads (E)
- Accident (A) loads including explosion over pressure, drop/tipover, accidental pressurization, fire, and aircraft impact
- Design basis tornado wind loads and tornado-generated missile loads (W_t)
- Probable maximum flood loads (F)
- Lightning

The staff has reviewed the SAR and found that the structural analysis procedures have been identified and are in conformance with standard engineering practice, as described in ACI 349-90 (American Concrete Institute, 1989). The relationship between the design criteria, identified in Chapter 3 of the SAR, and the analysis procedures were established in accordance with the requirements of 10 CFR 72.24(c)(2). The applicable codes and standards used in the analysis of the reinforced concrete structures have also been identified in the SAR, in accordance with the requirements of 10 CFR 72.24(c)(4).

Canister Transfer Building

The staff has reviewed Section 4.7 of the SAR and found that structural analysis of the Canister Transfer Building to mitigate environmental effects has been conducted by PFS. The structural analysis under accident loads is given in Sections 8.2.1.1, 8.2.1.2, and 8.2.2.2 of the SAR. The adequacy of the reinforced concrete structures has been demonstrated by the analysis results given in the SAR, as designed to satisfy the requirements of ACI 349-90 (American Concrete Institute, 1989).

The structural analysis of the Canister Transfer Building is described in Section 4.7.1.5.1 of the SAR. The building structure has been analyzed, and structural elements have been designed for the bounding load cases. Section 4.7.1.5.1 of the SAR provides a detailed discussion associated with determination of the governing load combinations. The original nine load combinations were reduced to the two that control the horizontal and vertical loads on the Canister Transfer Building. The staff concurs with the controlling load combinations, which are as follows:

$U_c > D + L + H + T + E$, and

$U_c > D + L + H + T + W_t$,

where U_c is the minimum available strength of a cross section or member calculated according to the requirements and assumptions of ACI 349-90. The dead loads (D) for the Canister Transfer Building include the self weight of the structure and all permanently attached equipment. Live loads (L) include snow and ice loads, bridge and semi-gantry crane loads, normal crane handling loads, normal wind loads, vehicle loads, and equipment loads. For the shallow foundation design considered here, the soil pressure loads (H) are insignificant. To accommodate thermally induced movements, expansion joints will be provided based on the site-specific extreme temperatures (T).

The horizontal loading is controlled by the earthquake (E) and tornado wind (W_t) loading. For the tornado wind loads, the lateral force is proportional to the velocity of the wind squared. Therefore, the load due to 240 mph tornado wind ($\propto 240^2 \Rightarrow \propto 57{,}600$) is significantly greater than that due to the 90 mph normal wind including the load factors ($\propto 0.75 \times 1.7 \times 90^2 \Rightarrow \propto 10{,}328$). For out of plane pressures, the tornado wind velocity and associated pressure drop result in a maximum pressure of 319 psf, based on the procedures of ASCE 7–95. Based on PFS's lumped mass model results for the Canister Transfer Building, the peak horizontal acceleration at elevated locations is bounded by 0.9 g. Assuming 0.9 g horizontal acceleration to account for the acceleration level at elevated locations, the equivalent pressure of 270 psf. Although the tornado pressure is higher, the resulting shear in the wall due to seismic load will be higher because of the inclusion of the full weight of the Canister Transfer Building times the peak ground acceleration in the seismic load. For the wind loading, the shear is proportional to the dynamic pressure times the cross-sectional area of the Canister Transfer Building. When compared on this global basis, the lateral force due to the tornado wind is 4,658 kips versus 36,500 kips for the earthquake, excluding the force due to acceleration of the base mat. Therefore, the design of the Canister Transfer Building's structural elements for resisting horizontal loads is controlled by the earthquake loading.

The vertical loading of the Canister Transfer Building is also controlled by the earthquake loading. For a vertical acceleration of 0.9 g at the roof structure, the resulting uniform load is 335 psf. The 0.9 g is based on response of the roof node of the lumped mass model of the Canister Transfer Building under seismic loading. This uniform pressure load is greater than the uniform load combination specified in ANSI/ANS 57.9 and ACI 349-90, which is 1.4 times the dead load and 1.7 times the live load or 295 psf. Therefore, the design of the Canister Transfer Building structural elements to resist vertical loads is controlled by the earthquake loading.

The staff concurs with the methods used to identify the controlling load combinations. The overall design of the reinforced concrete members is, therefore, appropriately based on the earthquake event.

Analysis of the reinforced concrete Canister Transfer Building has been provided (Stone & Webster Engineering Corporation, 2001m, 1998b,c). A 3-dimensional finite element model, using ANSYS was developed that adequately represents the structural elements of the Canister Transfer Building. Using the controlling load combinations, a finite element analysis identified shear and axial forces and moments in the structural elements of the Canister Transfer Building. Steel reinforcement size and placement for the foundation pad, wall, roof, beam, and column elements were established (Stone & Webster Engineering Corporation, 1998c) based on these demands. The design of the concrete structure and its reinforcement are based on the requirements in ACI 349-90. The ACI 349-90 Code specifies the minimum requirements for the design and construction of nuclear safety-related concrete structures and structural elements for nuclear power generating stations. The procedures for selection of the reinforcement and checks for axial, shear, moment, and torsional resistance of the elements are in conformance with standard engineering practice, as described in ACI 349-90 (American Concrete Institute, 1989). As noted, this analysis is not a final design and covers only major elements under the two seismic loading conditions that are considered by the staff to be bounding. Results of the analysis for these two bounding load cases indicate that the available design strength exceeds that required for the factored design loads (Stone & Webster Engineering Corporation, 1998c). The structural analysis performed by PFS demonstrates that the structural elements of the Canister Transfer Building are designed to resist the seismic loads based on the site characteristics and environmental conditions, in accordance with the requirements of 10 CFR 72.122(b)(1). PFS's analysis of the stability of the subsurface materials under the Canister Transfer Building loading is evaluated in Section 2.1.6.4 of this SER.

A seismic analysis of the structure was performed to determine the seismic loads for the building design and to generate in-structure response spectra for the design of the overhead and semi-gantry cranes supported by the Canister Transfer Building walls (Stone & Webster Engineering Corporation, 2001m). The seismic analysis was performed following the guidelines of ASCE 4-86 (American Society of Civil Engineers, 1986). ASCE 4-86 provides minimum requirements and indicates acceptable methods for the seismic analysis of safety-related structures of a nuclear facility. The analysis presented in the SAR and supplemental documentation, such as Stone & Webster Engineering Corporation calculations SC-3, SC-4, SC-5, SC-6, and S-10, are based on the site-specific design earthquake anchored at 0.711 g horizontal and 0.695 g vertical (developed from the PSHA). Details of the development of the artificial time histories were based on a near-source recording of a normal-faulting earthquake at Irpinia, Italy (Geomatrix Consultants Inc., 2001b). These time histories were then scaled to

the 2,000-yr return period design response spectra using both frequency and time domain approaches. The resulting time histories were shown to satisfy the requirements of Section 3.7.1 of NUREG–0800 (Nuclear Regulatory Commission, 1989) and ASCE 4-86 (American Society of Civil Engineers, 1986) in terms of the statistical independence of the time histories, envelopment of the shock spectra, and the power spectral density levels. The analysis is documented in Calculation 05996.02-G(PO18)-3 (Geomatrix Consultants, Inc., 2001b).

The dynamic analysis is based on a lumped mass model of the Canister Transfer Building with ten mass locations. These included the basemat, the lower roof, the crane elevation, the upper roof, the local flexibility of the walls supporting the crane, and the local flexibility of the roof in the vertical direction. The mass and stiffness properties of the building were based on hand calculations and represent an ideal case where no rotation is present. The lumped mass model is an acceptable model of the Canister Transfer Building. Impedance functions were developed to represent the subgrade, using the layered dynamic soil properties described in Calculation G(P018)-2 (Stone & Webster Engineering Corporation, 2001b) and SC-4 (Stone & Webster Engineering Corporation, 2001a). Discussions of the soil characteristics are contained in Chapter 2 of this SER. These soil characteristics were subsequently used in the seismic analysis of the Canister Transfer Building (Stone & Webster Engineering Corporation, 2001m).

Two seismic load cases were considered. One included 100 percent of the vertical component combined with 40 percent of each horizontal direction. The second included 100 percent of the east-west horizontal direction combined with 40 percent in the north-south horizontal direction and 40 percent of the vertical direction. These load conditions represent the bounding cases for all possible combinations of seismic components. Peak broadened response spectra at elevations 100 and 170 ft were developed for three mutually perpendicular directions of the Canister Transfer Building (Stone & Webster Engineering Corporation, 2001m). For the north-south direction, the elevated portion of the response spectra, where the response acceleration exceeds the peak ground acceleration, was between 3.5-5.5 Hz. For the east-west direction, the elevated portion of the response spectra was between 2.5-5.1 Hz. In the vertical direction, the elevated portion of the response spectra was between 5–12 Hz. The elevated portions of the response spectra correspond to the natural frequencies of the system. These elevated response levels were used to define the loading in the subsequent three-dimensional equivalent static finite element analysis of the Canister Transfer Building.

Additional seismic (equivalent static) analysis of the Canister Transfer Building, using the conceptual configuration of the building, was performed by the applicant using a three-dimensional ANSYS finite element model of the building and soil below and around the building (Stone & Webster Engineering Corporation, 1998a). The soil is modeled with three-dimensional elastic solid elements, which were assigned properties identified in Chapter 2 of the SAR. The building basemat, walls, and roof were modeled as a grid of 5 x 5-ft shell elements. Elastic beam elements were modeled to represent the beams and columns. The ANSYS model of the Canister Transfer Building is an acceptable representation of the structure and the supporting soil. The basemat was coupled to the soil using gap elements that allow uplift of the foundation. Calculations using two bounding seismic load conditions were performed. As identified in the analysis report (Stone & Webster Engineering Corporation, 1998b), these load conditions were considered by PFS to be the bounding conditions. The staff concurred with these bounding load conditions in the PFS SER, based on its review of the PFS analysis. For each of these cases, an equivalent static analysis was performed based on the zero period accelerations obtained in the seismic analysis described previously (Stone &

Webster Engineering Corporation, 1998a). The structural analysis was used to calculate loads on the Canister Transfer Building structural elements. Maximum loads were identified for all structural elements. Rebar size and placement were selected to ensure that the capacity of all sections exceeded the loads. The selected rebar sizes are based on the analysis results with a small factored increase. This modeling process provided a good indication of the overall response of the structure. The Canister Transfer Building reinforcement was designed to meet the minimum flexural and shear reinforcement requirements of ACI 349-90 (American Concrete Institute, 1989). In the SER of September 2000, the staff concluded that the structural analysis carried out by PFS, based on its previous design configuration, demonstrated that the Canister Transfer Building is designed to withstand the effects of natural phenomena, such as earthquakes, without impairing the capability to perform safety functions in accordance with the requirements of 10 CFR 72.122(b)(2).

The applicant plans to revise the detailed analysis and design of the conceptual design configuration it had submitted (Stone & Webster Engineering Corporation. 1998a, b) to account for recent changes in the configuration of the building, using the methods and codes previously approved by the staff (Section 4.7.1.5.3 of the PFS SAR). The applicant states that the changes in the design configuration would not result in changes in the sizes of various structural elements, but would be limited to the amount and placement of reinforcing steel only.

As stated in the staff's SER of September 2000, the Canister Transfer Building is also designed to withstand the loads due to tornado wind and pressure drop by means of its static strength without the need to resort to venting of the structure. In addition, the components representing the external boundary of the Canister Transfer Building have sufficient strength and stability to prevent penetration of the tornado missile and spalling of the concrete face interior to the point of impact, as shown in the calculation package SC-7 (Stone & Webster Engineering Corporation, 1998b). As identified in Section 4.7.1 of the SAR, the design of the Canister Transfer Building will be in accordance with the requirements of ACI 349-90. Since the tornado loads have not changed since issuance of the staff's SER, the staff concludes that the Canister Transfer Building would continue to perform its safety functions during a tornado event.

The Canister Transfer Building, which is approximately 90 ft tall, is identified as a moderate to severe risk factor for possible lightning strike. The Canister Transfer Building will be designed with lightning protection features in accordance with NFPA 780 (National Fire Protection Association, 1997b). This includes multiple air terminals on the roof with a two-way path to ground for any of the terminals. Because of the massive structure of the Canister Transfer Building, potential of structural damage due to lightning strike is minimal. The Canister Transfer Building is designed to withstand the effects of natural phenomena, such as lightning, without impairing the capability to perform safety functions in accordance with the requirements of 10 CFR 72.122(b)(2).

The Canister Transfer Building will not be subjected to flood loads. The location of the Canister Transfer Building is above the maximum probable flood level. In addition, the area will be protected by an earthen berm to prevent sheet flow around the Canister Transfer Building.

The staff has reviewed Section 4.7.3.5.1 (G) of the SAR and determined that the design of reinforced concrete structures, systems, and components provides fire and explosion protection while Section 8.2.5.2 of the SAR shows the capability of structures, systems, and components important to safety to withstand postulated fire and explosion accidents. The Canister Transfer

Building is a massive reinforced concrete structure. The proposed Facility would be located on an open gravel surface. PFS will be planting a 300 ft wide crested wheatgrass barrier around the restricted area. Therefore, the site will have more than 100 ft of fuel break around any storage cask or site structure important to safety. Consequently, the Canister Transfer Building will not be affected from any credible wildfire. Potential fires in the Canister Transfer Building are based on 50 and 300 gal. of diesel fuel. The extent and duration of fires are such that the capacity of the structural elements will not be degraded as a result of exposure to fire. As identified in the design criteria, the 1 psi overpressure from explosion is bounded by the pressure drop and stress caused by tornado wind and seismic loading, respectively. The design of the reinforced concrete structure has been shown to be acceptable under these greater load conditions. Therefore, the analysis demonstrates that the Canister Transfer Building is designed to continue performing its safety-related functions effectively under credible fire and explosion conditions, in accordance with the requirements of 10 CFR 72.122(c). Additional discussions on fire and explosion are contained in Section 6.1.5 and Chapter 15 of this SER.

The structural analysis also demonstrates that the Canister Transfer Building is designed such that the waste handling system has adequate safety under normal, off-normal, and accident conditions in accordance with the requirements of 10 CFR 72.128(a) because it is completely housed within the Canister Transfer Building.

In sum, the structural integrity of the Canister Transfer Building has been demonstrated under these normal, off-normal, and accident conditions.

Cask Storage Pads

Based on the information presented in SAR Section 4.2.3.5.1, Storage Pad Analysis, the reinforced concrete pads were designed and analyzed in accordance with ACI 349-90 (American Concrete Institute, 1989) and ANSI/ANS-57.9 (American National Standards Institute/American Nuclear Society, 1992). The ACI 349-90 Code specifies the minimum requirements for the design and construction of nuclear safety-related concrete structures and structural elements for nuclear power generating stations. Based on a review of the storage pad analysis and design calculation package (International Civil Engineering Consultant, Inc., 2000), it was noted that the concrete strength was identified as 3,000 psi. The cask transporter weight was identified as 145,000 lb, whereas in Section 8.2.6 of the SAR it is identified as 160,000 to 185,000 lb. The increased weight of the cask transporter, up to 40,000 lb, is minor when compared with the overall weight of the eight casks (2,880,000 lb). The static and dynamic analyses for evaluating the concrete pad response displacements and internal stresses have used the finite element analysis computer programs CECSAP and SASSI respectively (International Civil Engineering Consultants, Inc., 1996, 1997).

The storage pad analysis and design calculation package (International Civil Engineering Consultants, Inc., 2001) includes static analysis with both dead and live loads using CECSAP (International Civil Engineering Consultants, Inc., 1996). The storage pad was modeled using a three-dimensional, flat-shell finite element model. Gross uncracked stiffness of the storage pad was used for the model. Vertical springs were used to model the upper, best, and lower bounds of the soil support of the pads for the long-term static load conditions. The cask pad analysis is based on the maximum loaded cask weighing 360,000 lb. Three loading patterns of

2, 4, and 8 fully loaded casks are considered. In addition, another load case considered 7 loaded casks and one cask being lifted by a cask transporter on the pad. A dynamic amplification factor of 2 is used for this case to account for any dynamic effect of transporting the cask. Cask loadings are lumped to four points on the outer circular perimeter of each cask. Based on a review of the input files for the static analysis, values for the geometry, soil parameters, and loading are consistent with the design.

Static analysis of the stability of subsurface materials under the storage pad loading (including the casks) is reviewed in Section 2.1.6.4, Stability of Subsurface Materials, of this SER.

The results of the static pad analysis for dead and live loads of cask weights are summarized in Table 4.2-7 of the SAR. Based on the results of this analysis, the cask-loading pattern that produces the highest pad internal stresses is that of four casks on the pad. The maximum moment in the longitudinal direction was $-M_{yy}$ = 109 k-ft/ft and $+M_{yy}$ = 138 k-ft/ft. The corresponding capacities identified in Section 4.2.3.5.2 of the SAR are $-M_{yy}$ = 210 k-ft/ft and $+M_{yy}$ = 218 k-ft/ft. The moment and shear capacities of the reinforced concrete pads are calculated based on the procedures identified in ACI 349–90. The maximum shear force was 19 k/ft (beam) and 9 k/ft (punching). The corresponding ultimate static beam shear capacity identified in Section 4.2.3.5.2 of the SAR is 110 k/ft and an ultimate static punching shear capacity of 110 k/ft. The staff has reviewed the procedures used to determine the ultimate static moment and shear capacity calculation for the reinforced concrete slab and found them to be consistent with industry practice, as identified in ACI 349–90. The checks for normal loading conditions were correctly based on the load combination (1.4D + 1.7L + 1.7H) for demand and the strength reduction factor (0.90 for bending and 0.85 for shear) for capacity. Therefore, considering the static pad analysis, the staff concludes that the storage pad, as designed, provides adequate strength for accommodating the design loading conditions.

The worst-case loading that produces the largest soil bearing pressures is from7 casks plus one cask being carried by the transporter. The maximum soil pressure has been calculated to be 3.6 ksf, which is less than the minimum allowable soil bearing pressure for static loads of 4.36 ksf. Based on a uniform distribution of load (dead weight for the slab and live loads for the casks and transporter) and the appropriate load combination, the staff has calculated the stress in the soil to be equal to 3.6 ksf.

Dynamic analysis (International Civil Engineering Consultants, Inc., 2001) has been performed for the site-specific PSHA design basis earthquake (0.711 g horizontal in two directions and 0.695 g vertical) using both CECSAP and SASSI computer codes. Three component time histories (loads representative of the site-specific design basis earthquake) were applied to the model. The modeling procedures used for the static analysis were also used for this dynamic analysis. For the short-term design basis earthquake loading, three-component boundary springs and dashpots representing the dynamic soil stiffness and radial damping characteristics are used. Values ranging from 0.2 to 0.8 were used to account for variations in the coefficient of friction between the pad and concrete casks. The value of 0.2 represents the upper-bound for sliding displacements of the cask. The value of 0.8 represents the upper-bound estimate of the cask dynamic forces acting on the pad. These values bound the range of frictional coefficient for concrete to steel interfaces. Three loading patterns of 2, 4, and 8 fully loaded casks were considered. The loads cover the range that can be expected at the PFS Facility.

The results of the dynamic pad analysis are summarized in Table 4.2-8 of the SAR. Based on the results of this analysis, the cask-loading pattern that produces the highest pad internal stresses is that of eight casks on the pad. The maximum moment in the longitudinal direction is $-M_{yy}$ = 203 k-ft/ft and $+M_{yy}$ = 113 k-ft/ft. The corresponding capacities identified in Section 4.2.3.5.2 of the SAR are $-M_{yy}$ = 232 k-ft/ft and $+M_{yy}$ = 242 k-ft/ft. Again the moment and shear capacities of the reinforced concrete pads are calculated based on the procedures identified in ACI 349–90. The maximum moment in the transverse direction is $-M_{yy}$ = 218 k-ft/ft and $+M_{yy}$ = 133 k-ft/ft. The corresponding capacities identified in Section 4.2.3.5.2 of the SAR are $-M_{yy}$ = 225 k-ft/ft and $+M_{yy}$ = 133 k-ft/ft. The maximum beam shear force is 58 k/ft, and the maximum punching shear force is 98 k/ft. The corresponding capacities identified in Section 4.2.3.5.2 of the SAR are 121 k/ft for both shear forces. The staff has reviewed the procedures used to determine the moment and shear capacity calculation for the reinforced concrete slab and found them consistent with industry practice, as specified in ACI 349-90. The checks for dynamic loading conditions were correctly based on the load combination (D + L + H + E) for demand and the strength reduction factor (0.90 for bending and 1.1 × 0.85 for shear) for the capacity. Therefore, considering the dynamic pad analysis, the staff concludes that the storage pad as designed provides adequate strength for accommodating the site-specific seismic loading conditions.

Dynamic analysis of the stability of subsurface materials under the storage pad loading is reviewed in Section 2.1.6.4, Stability of Surface Materials, of this SER. The maximum dynamic soil pressure has been calculated to be 7.35 ksf, which is less than the minimum ultimate soil bearing pressure of 13.1 ksf.

These static and dynamic analyses confirm the structural adequacy of the reinforced concrete storage pad for supporting the storage casks when subjected to the design loading conditions. From the static and dynamic analyses, pad responses were obtained and then combined to give the maximum response values in accordance with the applicable load combinations. The combined response values were then used for checking the structural adequacy of the concrete pad and the soil bearing and sliding stabilities. The structural analysis performed by PFS demonstrates that the cask storage pads are adequately designed to resist the loads based on the site characteristics and environmental conditions during normal operations and during postulated off-normal and accident events in accordance with the requirements of 10 CFR 72.122 (b)(1). Structural analysis carried out by PFS demonstrates that the cask storage pads are designed to withstand the effects of natural phenomena, such as earthquakes, without impairing the capacity to perform safety functions in accordance with the requirements of 10 CFR 72.122(b)(2).

For the slab on grade design of the storage pads, the tornado winds will not exert any additional load to the structure. Additionally, the cask storage pad will not be subjected to flood load because the storage pads will be above the maximum probable flood level (Private Fuel Storage Limited Liability Company, 2000). In addition, the area is protected by an earthen berm to prevent sheet flow over the pads. Moreover, lightning strikes will not affect the safety function of the pad because it is grounded. Therefore, the cask storage pads are designed to withstand the effects of natural phenomena, such as tornadoes, lightning, and floods without impairing the capacity to perform safety functions in accordance with the requirements of 10 CFR 72.122(b)(2).

The PFS Facility concrete storage pads are surrounded by an open gravel surface. The gravel surface will be kept free of growth so no combustibles will be present. The distance from the edge of the gravel surface to the storage pads is greater than required to ensure that wildfires on the boundary will not endanger the cask storage pads. Additionally, PFS will be placing a fire barrier around the perimeter of the Facility restricted area. An analysis of potential fires on the cask storage pads due to a rupture of the transporter fuel tank (50 gal. of diesel fuel) has been performed. As evaluated in Chapter 15 of this SER, the fire from 50 gal. of diesel fuel will be of short duration and will not cause damage to the storage pads. The staff has reviewed Sections 4.2.1.5.1(l) and (j), 4.2.2.5.1(l) and (j), 4.7.3.5.1(e), and 4.7.4.5.1(d) of the SAR and determined that the design of the cask storage pads provides fire protection while Sections 8.2.4.2 and 8.2.5.2 of the SAR show the capability of structures, systems, and components important to safety to withstand postulated fire and explosion accidents in accordance with the requirements of 10 CFR 72.122(c).

5.1.4 Other Structures, Systems, and Components Important to Safety

This section contains a review of Sections 4.2.1, HI-STORM 100 Cask System; 4.7.1, Seismic Support Struts; Canister Transfer Building Seismic Support Struts Structural Steel Roof Beams; and Transfer Cell Sliding Doors; 4.7.2, Canister Transfer Cranes; and 4.7.3, HI-STORM 100 Transfer Equipment of the SAR. The staff reviewed the discussion on other structures, systems, and components important to safety with respect to the regulatory requirements of 10 CFR 72.120(a), 72.122(a) through (c), (f), and (g), and

5.1.4.1 Description of Other Structures, Systems, and Components Important to Safety

The following structures and components were identified in the SAR as other structures, systems, and components important to safety.

- Storage Cask (QA Category B)
- Transfer cask and associated lifting devices (QA Category B)
- Canister transfer overhead bridge and semi-gantry cranes (QA Category B)
- Seismic support struts (QA Category B)
- Canister Transfer Building Structural Steel Roof Beams (QA Category B)
- Canister Transfer Building Transfer Cell Doors (QA Category B)

The staff reviewed the description of structures, systems, and components important to safety with respect to the regulatory requirements of 10 CFR 72.24(b) and (c)(4), and 72.122(f) and (g).

Storage Cask

As identified in the SAR Section 4.2.1, HI-STORM 100 Cask System, the storage cask is a steel and concrete cylindrical structure that serves as a missile barrier and radiation shield, provides flow paths for natural convective heat transfer and stability for the system, and absorbs energy during non-credible hypothetical tipover accident events. The storage cask is designed to meet ASME Boiler and Pressure Vessel Code, Section III, Subsection NF requirements (American Society of Mechanical Engineers, 1998). Table 4.2-2 and Figure 4.2-3

of the SAR provide a summary of the physical characteristics of the storage cask. A complete design description of the storage cask system is given in the HI-STORM 100 FSAR.

The design criteria, material properties, and structural analysis of the storage cask, based on the generic design base loadings, are contained in Chapter 3 of the FSAR for the HI-STORM 100 Cask System. This cask system has been licensed for these generic design base loadings under Certificate of Compliance 1014. The generic design base loadings specified in the HI-STORM 100 FSAR envelop the PFS Facility site parameters, except for the seismic loadings. An additional site-specific cask stability analysis has been performed by Holtec International to demonstrate that the storage cask will not tipover, collide, or slide off the storage pad during a PFS Facility site-specific design-basis seismic event (Holtec International, 2001c). Site-specific structural analysis of storage casks is discussed in Section 5.1.4.4 of this SER.

To limit the deceleration loads on the cask due to a vertical drop or a non-mechanistic tipover event, PFS plans to use overpack concrete with a strength of 3,000 psi. This is a reduction from the 4,200 psi design compressive strength of the overpack concrete identified in the HI-STORM 100 FSAR.

The staff has reviewed SAR Section 4.2, Storage Structures, with respect to the description of the storage cask. These descriptions include consideration of inspection, maintenance, and testing. Components requiring inspection and maintenance are identified and operational procedures summarized. Inspection is limited to checks of the air vents to ensure that they are not blocked. This design also allows for emergency access. Spacing of the storage casks on the reinforced concrete pads allows for access to critical locations and regions in the event of emergencies.

Additionally, the staff review of Section 4.2, Storage Structures, of the SAR determined that the design features of the storage cask related to shielding and heat removal capability are appropriately described. A comprehensive shielding evaluation is contained in Chapter 7 of this SER. The design of the storage cask places the spent nuclear fuel in a sealed canister to limit the amount of radioactive waste generated at an ISFSI. A comprehensive waste confinement and management evaluation is contained in Chapter 14 of this SER.

Transfer Cask and Associated Lifting Devices

As identified in Section 4.7.3 of the SAR, the HI-STORM canister transfer equipment consists of a metal transfer cask (HI-TRAC), HI-TRAC lifting trunnions, shipping cask and transfer cask lift yokes, canister downloader, canister lift cleats, and HI-STORM lifting lugs. The HI-TRAC transfer cask, as identified in Section 4.7.3.4.1 of the SAR, is a heavy-walled cylindrical vessel constructed of carbon steel with water for neutron and lead for gamma shielding. The transfer cask provides an internal cylindrical cavity of sufficient size for housing a HI-STORM canister. An access hole through the HI-TRAC top lid is provided to allow lowering or raising the canister between the transfer cask and shipping or storage cask. A bottom lid incorporates two sliding doors that allow opening the HI-TRAC bottom for the canister to pass through. Figure 4.7-2 of the SAR shows the major components of the transfer cask. Table 4.7-1 of the SAR identifies the physical characteristics of the HI-TRAC transfer cask.

The remaining components are grouped as associated lifting devices. Trunnions are located beneath the transfer cask top flange for lifting and vertical handling of the cask. The function of the lifting yokes is to provide a lifting interface between the crane and the shipping cask or transfer cask. The canister downloader is a hoist unit attached to the top of the HI-TRAC transfer cask used to raise and lower the canister between the HI-TRAC transfer cask and the HI-STORM 100 storage cask or HI-STAR shipping cask in a single-failure proof mode without risk of over lifting the canister. The function of the canister lift cleats is to provide a means to lift the canister. The function of the HI-STORM 100 storage cask lifting lugs is to provide a means of lifting the storage cask.

The description of the transfer cask and associated lifting devices include consideration of inspection, maintenance, and testing in accordance with ANSI N14.6 (American National Standards Institute/American Nuclear Society, 1993) and NUREG-0612 (Nuclear Regulatory Commission, 1980). Components requiring inspection and maintenance are identified, and operational procedures are summarized. Pre-operational, startup, and operational tests will be performed to verify the functional operations of structures, systems, and components important to safety. This design also allows for emergency load carrying capability. Design of the transfer cask and associated lifting devices allows for control of loads in the event of emergencies.

Detailed design descriptions of the transfer cask and associated lifting devices are given in the HI-STORM 100 FSAR (Holtec International, 2000).

Canister Transfer Overhead Bridge and Semi-Gantry Cranes

The staff has reviewed SAR Section 4.7.2, Canister Transfer Cranes. The Canister Transfer Building houses two cranes, a 200/25-ton overhead bridge crane, (Figure 4.7-5 of the SAR), and a 150/25-ton semi-gantry crane, (Figure 4.7-6 of the SAR). As specified in the Technical Specifications, the cranes are single-failure proof and meet the requirements of NUREG-0612 and NUREG-0554 (Nuclear Regulatory Commission, 1979). The cranes are provided for loading and unloading shipping casks on or off the heavy haul tractor/trailers and transferring spent nuclear fuel canisters between the shipping and storage casks. The canister transfer cranes are designed by Ederer Incorporated. Detailed design of the cranes was performed for the crane vendor by Anatech Corporation (Anatech Corporation, 1998a,b). The staff has determined that the SAR description of the canister transfer overhead bridge and semi-gantry cranes satisfies the requirements of 10 CFR 72.24(b) and 72.24(c)(4) because it provides an adequate description of the cranes with special attention to design characteristics.

Components requiring inspection and maintenance are identified, and operational procedures are summarized. Pre-operational, startup, and operational tests will be performed in accordance with ASME NOG–1 (American Society of Mechanical Engineers, 1989) to verify the functional operations of structures, systems, and components important to safety. Therefore the descriptions include consideration of inspection, maintenance, and testing as required in 10 CFR 72.122(f). Design of the canister transfer overhead bridge and semi-gantry cranes allows for access to the crane structure in the event of emergencies. The cranes are designed to hold the load during emergencies in compliance with the requirements of ASME NOG–1. The design allows for emergency capability as required in 10 CFR 72.122(g).

Seismic Support Struts

The staff has reviewed SAR Section 4.7.1.4.1, Seismic Support Struts, and SC-10 (Stone & Webster Engineering Corporation, 2001d). The seismic support struts are rigid assemblies that secure the shipping, storage, and transfer casks to the Canister Transfer Building transfer cell walls during canister transfer operations. Figure 4.7-7 of the SAR shows the general layout of the seismic support struts. Details of the position of the struts are provided in SC-10 with an indication of the maximum load at each of the strut locations (Stone & Webster Engineering Corporation, 2001d). The struts ensure that the casks will remain stable and will not topple in the event of an earthquake during the transfer operation. Each cask utilizes two struts that provide restraint in both horizontal directions. The struts consist of a rigid tubular body with threaded eye rods on both ends. Each strut is pinned to a bracket that is secured to the cask and to the Canister Transfer Building cell wall. Details of the column anchor locations for attachment of the struts to the Canister Transfer Building are also provided. Two strut types are identified based on the required length and load carrying demands. As required in the Technical Specifications, the connection between the seismic support struts and the transfer cask, storage cask, and shipping cask must be sufficiently rigid to resist the design basis earthquake motions. Based on the SAR description of the seismic support struts and the applicable condition in the Technical Specifications, the staff has determined that the SAR adequately describes the seismic support struts per 10 CFR 72.24(b) and 72.24(c)(4) because it provides an adequate description of the struts with special attention to design characteristics.

The design of the struts is in accordance with ASME Subsection NF, Component Supports (American Society of Mechanical Engineers, 1998). Section NF-5000 Examination identifies the test and acceptance criteria. Pre-operational, startup, and operational tests will be performed to verify the functional operations of structures, systems, and components important to safety. Therefore, the description of the seismic support struts provided is sufficient to conclude that the struts will perform their design function in the event of a design-basis earthquake. These descriptions include consideration of inspection, maintenance, and testing, as required in 10 CFR 72.122(f). Design of the seismic support will not impede access to all locations and regions in the event of emergencies. Therefore, the design allows for emergency capability, as required in 10 CFR 72.122(g).

Canister Transfer Building Structural Steel Roof Beams

A steel frame supports the vertical roof dead weight, snow, and seismic loads. The roof decking spans are approximately 5 ft, and are supported by 16 inch deep steel beams. The 16 in. deep roof beams span up to 30 ft in the north-south direction to the main roof girders. Five feet deep main roof steel girders spanning 65 ft in the east-west direction carry the vertical roof loads to embedded plates set in the building's concrete walls. Based on a review of the SAR description of the Canister Transfer Building structural steel roof beams, the staff has determined that the SAR will provides an adequate description of the structural steel with special attention to design characteristics, and adequately describes the Canister Transfer Building structural steel roof beams in accordance with 10 CFR 72.24(b).

The design of the Canister Transfer Building structural steel roof beams is in accordance with ANSI/AISC N690 (American National Standards Institute/American Institute of Steel Construction, 1994). Therefore, the description of the Canister Transfer Building structural

steel roof beams provided in the SAR is sufficient to conclude that the structural steel will perform their design function in the event of a design-basis earthquake. These descriptions include consideration of inspection, maintenance, and testing, as required in 10 CFR 72.122(f). Design of the Canister Transfer Building structural steel roof beams will not impede access to other locations and regions in the event of emergencies. Therefore, the design allows for emergency capability, as required in 10 CFR 72.122(g). The staff therefore concludes that the design of the Canister Transfer Building structural steel roof beams complies with 72.24(c)(4).

Canister Transfer Building Transfer Cell Doors

There are three openings through the wall on the west side of the transfer cells to allow cask transporter access to each transfer cell. These openings are tornado missile protected during canister transfer operations by 1-foot thick rolling doors fabricated from ½ in. steel plate and 11 in. of concrete fill. These doors are designed to withstand tornado generated missiles in addition to satisfying Seismic II/I and radiation shielding requirements. As discussed in the SAR, the structural profiles will provide adequate radiation shielding.

There are three openings through the wall on the east side of the transfer cells to allow to placement of casks in the transfer cells using the Canister Transfer Building cranes. These doors are designed to remain in place during an earthquake event and satisfy radiation shielding requirements. The doors may yield and become non-functional as a result of the design basis earthquake. However, they are designed to remain on their guide tracks and not detrimentally affect safety-related operations or equipment. The doors consist of 3/8 in. steel cover plates, wide flange internal stiffeners, and 4.125 in. polyethylene fill. The exterior dimensions of the doors are 15 ft wide × 30 ft tall. There are nine equally spaced vertical W4 × 13 stiffeners. The nominal door weight per square foot is 50.36 lb/ft^2.

The design of the Canister Transfer Building transfer cell doors is in accordance with ANSI/AISC N690 (American National Standards Institute/American Institute of Steel Construction, 1994). Description of the transfer cell doors is sufficient to conclude that the doors will perform their design function in the event of a design-basis earthquake. These descriptions include consideration of inspection, maintenance, and testing, as required in 10 CFR 72.122(f). Design of the Canister Transfer Building transfer cells doors will not impede access to other locations and regions in the event of emergencies. Therefore, the design allows for emergency capability, as required in 10 CFR 72.122(g).

Based on the description in the SAR of the Canister Transfer Building transfer cell doors, the staff has determined that an adequate description of the structural steel with special attention to design characteristics has been provided in accordance with the requirements of 10 CFR 72.24(b) and 10 CFR 72.24(c)(4).

5.1.4.2 Design Criteria for Other Structures, Systems, and Components Important to Safety

The design bases for the other structures, systems, and components important to safety are given in the SAR. Table 4.1-1 of the SAR identifies details of the Facility's compliance with the general design criteria of 10 CFR Part 72, Subpart F. The staff reviewed the discussion of design criteria with respect to the regulatory requirements of 10 CFR 72.24(c), 72.120(a), 72.122(b)(1), (b)(2), (c), and (f), and 72.128(a).

Storage Cask

Design criteria for the cask systems are contained in the HI-STORM 100 FSAR. A discussion of the design criteria for the storage cask is given in Section 4.1.3, Design Criteria for Structures, Systems, and Components Important to Safety, of this SER. As identified in Chapter 4 of this SER, the site-specific criteria are enveloped by the design criteria identified in the HI-STORM 100 FSAR with the exception of the seismic loading. Additional site-specific analysis was performed to demonstrate compliance with the seismic design criteria that were not enveloped.

The staff has reviewed SAR Section 4.2.1, HI-STORM 100 Cask System. The design criteria establish the minimum design, fabrication, construction, testing, maintenance, and performance requirements for the storage cask. The design criteria address the site characteristics and environmental conditions under normal operations and under off-normal and accident events. The design criteria include the effects of natural phenomena and cover credible fire and explosion conditions.

Transfer Cask and Associated Lifting Devices

The HI-TRAC transfer cask is designed for all normal, off-normal, and design basis accident loadings during transfer operation to protect the HI-STORM 100 spent nuclear fuel canister from deterioration, provide adequate shielding, and allow the retrieval for the canister under all conditions. The HI-TRAC transfer cask is designed as a special lifting device in accordance with ANSI N14.6–1993 (American National Standard Institute/American Nuclear Society, 1993) and NUREG–0612. Special lifting devices are designed for handling a certain load or loads. In this case, the specific load is the spent nuclear fuel canister. ANSI N14.6–1993 sets forth the requirements for design, fabrication, testing, maintenance, and QA programs for special lifting devices used to handle containers with radioactive materials. The HI-TRAC transfer cask with transfer lid attached, is designed to meet Level A Subsection NF (American Society of Mechanical Engineers, 1998) stress limits while handling the dead load of the heaviest loaded canister.

The HI-TRAC transfer cask lifting trunnions, lift yokes, canister downloader, canister lift cleats, and storage cask lifting lugs are designed as special lifting devices in accordance with ANSI N14.6-1993 (American National Standard Institute/American Nuclear Society, 1993) and NUREG–0612 for non-redundant special lifting devices. Specifics of the design basis for these components are given in Section 4.7.5.3 of the SAR.

A complete discussion of the design criteria for the transfer cask and associated lifting devices is given in Section 4.1.3, Design Criteria for Structures, Systems, and Components Important to Safety, of this SER. The design criteria establish the minimum design, fabrication, construction, testing, maintenance, and performance requirements for structures, systems, and components important to safety. The design criteria address the site characteristics and environmental conditions during normal operations and during postulated off-normal and accident events. The design criteria include the effects of natural phenomena and cover credible fire and explosion.

Canister Transfer Overhead Bridge and Semi-Gantry Cranes

As identified in SAR Section 4.7.2.1, Design Specifications, the canister transfer cranes are designed to meet the requirements of the design criteria contained in Chapter 3, which requires the cranes be designed in accordance with ASME NOG-1 (American Society of Mechanical Engineers, 1989) and be single-failure-proof in accordance with NUREG-0612 and NUREG–0554. ASME NOG–1 covers electric overhead and gantry multiple girder cranes with top running bridges and trolleys used at nuclear facilities, and components of cranes at nuclear facilities. Specifically, the cranes are designated as Type 1 because they are used to handle a critical load and should be designed and constructed so that they will remain in place and support the critical load during and after a seismic event. The cranes do not have to be operational after this event. Single-failure-proof features must be included so that any credible failure of a single component will not result in loss of capability to stop and hold the critical load within acceptable excursion limits. A complete discussion of the design criteria for the canister transfer overhead bridge and semi-gantry cranes is given in Section 4.1.3, Design Criteria for Structures, Systems, and Components Important to Safety, of this SER. The design criteria for the canister transfer overhead bridge and semi-gantry cranes are described in sufficient detail to support the findings in 10 CFR 72.40 as required by 10 CFR 72.24(c)(1). The applicable codes and standards for the canister transfer overhead bridge and semi-gantry cranes are identified and, therefore, support the findings in 10 CFR 72.40 as required by 10 CFR 72.24(c)(4).

Sections NOG-4000, Requirements for Structural Components, NOG–5000, Mechanical, and NOG–6000, Electrical Components of ASME NOG–1, identify specific design criteria. The design criteria establish the minimum design, fabrication, construction, testing, maintenance, and performance requirements for structures, systems, and components important to safety in accordance with the requirements of 10 CFR 72.120(a). As identified in Chapter 4 of the SER, the design criteria address the site characteristics and environmental conditions during normal operations and during postulated off-normal and accident events. The design criteria include the effects of natural phenomena and cover credible fire and explosion conditions. Performance of testing, inspection, and maintenance activities on the cranes in accordance with 10 CFR 72.122(f) is covered in Section NOG–7000, Inspection and Testing, of ASME NOG–1.

Seismic Support Struts

The support struts are procured as standard sway strut assemblies that conform to ASME Subsection NF requirements (American Society of Mechanical Engineers, 1998) for Class 2 nuclear grade supports (Private Fuel Storage Limited Liability Company, 2001, Section 4.7.1.4.1). Design criteria are identified in Chapter 3 of the SAR and in ASME

Subsection NF (American Society of Mechanical Engineers, 1998). The design of the column anchorage is based on ANSI/AISC N690 (American National Standards Institute/American Institute of Steel Construction,1994). ANSI/AISC N690 is the specification and commentary for the design, fabrication, and erection of structural steel for steel safety-related structures for nuclear facilities. The design criteria for the seismic support struts are described in sufficient detail to support the findings in 10 CFR 72.40 in accordance with the requirements of 10 CFR 72.24(c)(1). The applicable codes and standards for the seismic support struts are identified and, therefore, support the findings in 10 CFR 72.40 as required by 10 CFR 72.24(c)(4). A complete discussion of the design criteria for the seismic support struts is given in Section 4.1.3, Design Criteria for Structures, Systems, and Components Important to Safety, of this SER.

The design of the struts are in accordance with ASME Subsection NF, Component Supports. Section NF-5000, Examination, identifies the test and acceptance criteria. The design criteria establish the minimum design, fabrication, construction, testing, maintenance, and performance requirements for the seismic struts. The design criteria address the site characteristics and environmental conditions during normal operations and during postulated off-normal and accident events, include the effects of natural phenomena, and cover credible fire and explosion conditions.

Canister Transfer Building Structural Steel Roof Beams

Design criteria for the Canister Transfer Building are identified in Chapter 3 of the SAR. The design is based on ANSI/AISC N690 (American National Standards Institute/American Institute of Steel Construction,1994). The design criteria for the Canister Transfer Building structural steel roof beams are described in sufficient detail to support the findings in 10 CFR 72.40 in accordance with the requirements of 10 CFR 72.24(c)(1). The applicable codes and standards for the Canister Transfer Building structural steel roof beams are identified and, therefore, support the findings in 10 CFR 72.40 as required by 10 CFR 72.24(c)(4). The design criteria establish the minimum design, fabrication, construction, testing, maintenance, and performance requirements for the Canister Transfer Building structural steel roof beams. The design criteria address the site characteristics and environmental conditions during normal operations and during postulated off-normal and accident events, include the effects of natural phenomena, and cover credible fire and explosion conditions.

Canister Transfer Building Transfer Cell Doors

Design criteria for the Canister Transfer Building are identified in Chapter 3 of the SAR. The applicable loads and load combinations for the transfer cell doors were obtained from Chapter 3 of the PFS Facility SAR. The design acceleration values are identified for the top of the Canister Transfer mat foundation as 1.1 g north-south horizontal, 4.0 g east-west horizontal, and 0.8 g vertical (Stone & Webster Engineering Corporation, 2001m). The governing design load combinations as identified in Section 3.2.11.4 of the PFS Facility SAR:

$1.6S > D + E$
$1.4S_v > D + E$

Allowable stresses for the steel members are based on ANSI/AISC N690 (American National Standards Institute/American Institute of Steel Construction,1994). The design criteria for the Canister Transfer Building transfer cell doors are described in sufficient detail to support the findings in 10 CFR 72.40 in accordance with the requirements of 10 CFR 72.24(c)(1). The applicable codes and standards for the Canister Transfer Building transfer cell doors are identified and, therefore, support the findings in 10 CFR 72.40 as required by 10 CFR 72.24(c)(4). The design criteria establish the minimum design, fabrication, construction, testing, maintenance, and performance requirements for the Canister Transfer Building transfer cell doors. The design criteria address the site characteristics and environmental conditions during normal operations and during postulated off-normal and accident events, include the effects of natural phenomena, and cover credible fire and explosion conditions.

5.1.4.3 Material Properties for Other Structures, Systems, and Components Important to Safety

The staff reviewed the material properties for other structures, systems, and components important to safety with respect to the regulatory requirements of 10 CFR 72.24(c)(3) and (4).

Storage Cask

The material properties of the storage casks are provided in the HI-STORM 100 FSAR. The staff's evaluation of the HI-STORM 100 FSAR is documented in NRC's HI-STORM 100 SER. As identified in the PFS Facility SAR, PFS will use concrete with a compressive strength of 3,000 psi for the storage cask overpack, instead of 4,200 psi identified in HI-STORM 100 storage cask FSAR. This change is to provide energy absorption in the event of a tipover or handling accident.

Transfer Cask and Associated Lifting Devices

Material properties for the HI-TRAC transfer cask and associated lifting devices are provided in the HI-STORM 100 FSAR. The staff's evaluation of the HI-STORM 100 FSAR is documented in NRC's HI-STORM 100 SER.

Canister Transfer Overhead Bridge and Semi-Gantry Cranes

Information on the materials used in the construction of the cranes is contained in seismic qualification analysis reports for the cranes (Anatech Corporation 1998a,b). This conclusion is based on meeting the material requirements of ASME NOG–1, where the materials are identified. The applicable codes and standards are identified in accordance with the requirements of 10 CFR 72.24(c)(4).

Seismic Support Struts

The seismic support struts are designed in accordance with ASME Subsection NF (American Society of Mechanical Engineers, 1998) requirements for Class 2 nuclear grade supports. This ensures that appropriate materials are used for the seismic support struts. Therefore, the requirements of 10 CFR 72.24(c)(3) are satisfied. The applicable codes and standards are identified in accordance with the requirements of 10 CFR 72.24(c)(4).

Canister Transfer Building Structural Steel Roof Beams

As identified in SC-12 the majority of the structural steel members will be fabricated from steel with a yield strength of 50 ksi. Based on the size of the various elements any one of a number of ASTM designated steels could be used to fabricate the structural elements (American Institute of Steel Construction, 1989). Allowable stresses for the steel members were obtained from ANSI/AISC N-690. This ensures that appropriate materials are used for the Canister Transfer Building structural steel roof beams. Therefore, the requirements of 10 CFR 72.24(c)(3) are satisfied. The applicable codes and standards are identified in accordance with the requirements of 10 CFR 72.24(c)(4).

Canister Transfer Building Transfer Cell Doors

As identified in SC-14 the structural steel members for the east doors will be fabricated from steel with a yield strength of 50 ksi. The structural steel members for the west tornado resistant doors will be fabricated from steel with a yield strength of 36 ksi. The density of the polyethylene used is 57.4 lb/ft^3. Based on the review of the material properties of the doors, the staff concludes that the requirements of 10 CFR 72.24(c)(3) are satisfied. The applicable codes and standards are identified in accordance with the requirements of 10 CFR 72.24(c)(4)

5.1.4.4 Structural Analysis for Other Structures, Systems, and Components Important to Safety

The staff has reviewed the SAR and found that the structural analysis procedures have been identified and are in conformance with standard engineering practice. Other structures, systems, and components important to safety were designed and analyzed to resist the loads and loading combinations specified in the design criteria. As identified in Sections 4.7.3.5.1 and 4.7.4.5.1 of the SAR, the analyses of other structures, systems, and components important to safety included loading conditions of dead and live loads, thermal loads, earthquake, and fire. Evaluation for tornado, wind, or tornado missiles is not required for structures, systems, and components inside the Canister Transfer Building, except for the transfer cell doors that are identified as part of the tornado missile boundary. The staff reviewed the structural analysis for other structures, systems, and components important to safety, including the transfer cell doors, that are identified as part of the tornado missile boundary, with respect to the regulatory requirements of 10 CFR 72.24 and 72.122.

Storage Cask

The staff has reviewed Section 4.2 of the SAR and found that the design of storage casks to mitigate environmental effects is identified and that Chapter 8 and Sections 8.2.1.1, 8.2.1.2, and 8.2.2.2 of the SAR demonstrate the capability of structures, systems, and components important to safety to withstand postulated accidents and environmental conditions. The detailed structural analysis of confinement structures is presented in the HI-STORM 100 FSAR. The staff has previously reviewed this structural analysis and found it acceptable, as documented in the staff's HI-STORM 100 SER. As documented in that SER, the structural analysis shows that the structural integrity of the HI-STORM 100 Cask System is maintained under all credible loads analyzed in the HI-STORM 100 FSAR.

The PFS Facility SAR provides a summary of the analysis performed in the HI-STORM 100 FSAR. The loading conditions at the Facility are enveloped by the loading conditions considered in the HI-STORM 100 FSAR (Holtec International, 2000), except for the seismic loads. To fully characterize the response of the system at the PFSF, Holtec International performed site-specific analyses of the HI-STORM 100 Cask system for thermal, seismic, and drop/tipover loads at the PFS site.

For the analysis of the storage cask, the dead load of the cask and the spent nuclear fuel canister were considered. The loads are bounded by those identified in the HI-STAR 100 FSAR. Under the applied dead loads the resulting stress levels are shown in the HI-STORM 100 FSAR to be within applicable code allowables. Since the PFS cask dead loads are bounded by the loads in the HI-STAR 100 FSAR, the stresses in the HI-STORM 100 FSAR for dead loads bound the PFS Facility design criteria in Section 3.2.1 of the SAR for dead loads.

Live loads considered for the storage cask include the snow and ice loads and the HI-TRAC transfer cask weight containing a fully loaded canister. The HI-STORM 100 FSAR uses a 100 psf snow load which bounds the 45 psf snow load applicable at the site. The live load capacity of the storage cask from the weight of the HI-TRAC transfer cask with a fully loaded canister is shown in Section 3.4.4.3.2.1 of the HI-STORM FSAR to be adequate. The live loads used in the HI-STORM analysis bound the PFS Facility design criteria specified in Sections 3.2.2 and 3.2.3 for live loads and snow loads.

Differential internal and external pressure loads are not applicable to the vented concrete storage cask. A discussion of the thermal design of the storage cask is given in Section 4.2.1.5.2 of the SAR. Thermal analyses were performed to evaluate the steady-state temperature for components of the storage system (Holtec International, 1999, 2001a). The analysis included consideration of the physical characteristics of the cask storage pad, adjacent casks, and worst case condition of heat generated within the spent nuclear fuel canisters. A summary of the steady state temperature is given in Table 4.2-3 of the SAR. The thermal design of the HI-STORM 100 storage system bounds the site-specific design requirements. The structural analysis demonstrates that the storage cask is designed to resist the loads based on the site characteristics and environmental conditions during normal operations and during postulated off-normal and accident events in accordance with the requirements of 10 CFR 72.122(b)(1).

The staff has reviewed Section 4.2 of the SAR and found that the design of storage casks to mitigate environmental effects is identified and that Chapter 8 and Sections 8.2.1.1, 8.2.1.2, and 8.2.2.2 of the SAR demonstrate the capability of structures, systems, and components important to safety to withstand postulated accidents and environmental conditions. Analysis of the structural response of the storage cask to the earthquake load was based on generic loading identified in the HI-STORM 100 FSAR and the PFS Facility site-specific seismic events. As identified in Section 4.2.3.5.4 of the SAR, the cask stability analysis ensures the storage casks will not tipover or slide excessively during a seismic event. The cask stability analysis for a generic design basis earthquake is given in Section 3.4.7 of the HI-STORM 100 FSAR. Additionally, a site-specific cask stability analysis was performed by Holtec International that demonstrates the storage cask will not tipover, collide, or slide off the storage pad during a site-specific design basis earthquake (Holtec International, 2001b, c). The cask stability analyses are described in detail in Section 8.2.1 of the SAR.

Analysis of the structural response of the storage cask to the earthquake load was based on loading identified in the HI-STORM 100 FSAR and the PFS Facility site-specific seismic events. As identified in Section 4.2.3.5.4 of the PFS SAR, the cask stability analysis ensures the storage casks will not tipover or slide excessively during a seismic event. The cask stability analysis for a generic design basis earthquake is given in Section 3.4.7 of the HI-STORM 100 FSAR. Additionally, a site-specific cask stability analysis was performed by Holtec International that demonstrates the storage cask will not tipover, collide with another cask, or slide off the storage pad during a site-specific design basis earthquake (Holtec International, 2001a). The analysis was performed two ways. For the first case it was assumed that the concrete pad, the soil-cement layer, and the underlying soil were fully bonded. For the second case, the concrete pad and soil-cement layer were allowed to slide when frictional resistance exceeded the limits. Based on this analysis it was concluded that the casks will remain stable during a seismic event. The cask stability analyses are described in detail in Section 8.2.1 of the PFS SAR. The inertia loads and resulting stresses in the MPC produced by the seismic event are less than the 45 g loads associated with the non-mechanistic tipover and vertical drop events. Therefore, the staff's conclusions in its HI-STORM 100 SER with respect to the structural integrity of the storage cask are valid for the PFS Facility. The HI-STORM design meets the PFS Facility design criteria in Section 3.2.10 of the PFS Facility SAR for seismic design.

The analysis considers a single 30 ft x 67 ft x 3 ft. concrete pad supporting up to eight HI-STORM 100 storage casks. The concrete storage pad is modeled as a rigid plate structure supported on linear springs that characterize the behavior of the underlying foundation under dynamic loading from a seismic event. PFS uses bounding values for soil properties (Young's Modulus, Shear Modulus, and Poisson's Ratio) that conform to those identified in Chapter 2 of the SAR. Using the smaller values of Young's Modulus and Shear Modulus coupled with the larger value of Poisson's Ratio, results in a lower value for the soil spring constants. Conversely, using the larger values of Young's Modulus and Shear Modulus coupled with the smaller value of Poisson's Ratio, results in a higher value for the soil spring constants. The sensitivity analysis demonstrates that variations in soil moduli leading to upper and lower bound estimates of the soil springs at the Facility pad interface have minimal effect on the maximum cask excursion. The lower bound values give rise to large cask displacements, resulting in the bounding analysis for displacement.

The cask system weight and dimensions are the same as the HI-STORM 100 storage system. Each cask is modeled as a mass-spring system with appropriate nonlinear characteristics to simulate compression-only contact, impact, and lift-off of the cask from the slab. The layout of the casks on the slab allots a 15 ft × 16 ft pad space for each of the spent fuel casks. The minimum spacing between casks is 48 in. The model adequately represents the physical system.

The time histories used as the seismic input correspond to the 2000-yr event with 0.71 g in two horizontal directions and 0.695 g in the vertical direction (Stone & Webster Engineering Corporation, 2001b).

The acceptance criterion was that the casks must be stable in the sense that the center of the top cover of the cask must remain within the original contact circle that the cask makes with the pad. The maximum rocking at the top was less than 4 in. The maximum sliding was less than 3 in. This is significantly less than the spacing between the casks themselves and the edge of the pad. Consequently, the cask will not tipover, slide off the pad, or impact adjacent casks

during a site-specific design basis earthquake. Holtec International performed additional analyses (Holtec International, 2001a) that allowed slip between the pad and soil. Under that condition, the maximum cask excursions relative to the pad did not exceed 0.02 in at the top or bottom of the cask. Therefore, the structural analysis demonstrates that the storage cask is designed to withstand the effects of site-specific earthquakes without impairing the capability to perform safety functions, in accordance with the requirements of 10 CFR 72.122(b)(2).

The wind loading is enveloped by the tornado wind load conditions. Tornado wind and tornado missile loads are addressed in HI-STORM 100 FSAR Sections 3.1.2.1.1.5 and 3.4.8. Section 4.2.1.5.1 of the SAR specifies that the postulated missile loads used in the HI-STORM 100 analysis are the same as in the Facility design criteria. The SAR tornado missiles are identified as Spectrum II missiles (Section 3.5.1.4 of NUREG–0800). All six of the Spectrum II missiles identified in NUREG–0800 are used as the Facility design criteria. The HI-STORM 100 FSAR identifies the three design basis tornado missiles as Spectrum I missiles (Section 3.5.1.4 of NUREG–0800). Both the SAR and the HI-STORM 100 FSAR are in compliance with the requirement of NUREG–0800. Since the HI-STORM 100 cask design criteria for tornado wind and tornado-generated missile bound the Facility design criteria, the HI-STORM 100 design meets the Facility design criteria. Holtec International performed an additional analysis to determine the influence of reducing the compressive strength of the cask concrete from 4,200 psi to 3,000 psi (Holtec International, 2001d). The analysis showed that there was additional penetration into the concrete but the confinement barrier was not breached. The structural analysis demonstrates that the storage cask is designed to withstand the effects of natural phenomena such as tornadoes without impairing the capability to perform safety functions in accordance with the requirements of 10 CFR 72.122(b)(2).

The storage casks will not be subject to flood loads. The location of the storage pads is above the maximum probable flood level. In addition, the area is protected by an earthen berm to prevent sheet flow over the pads.

Lightning is addressed in HI-STORM 100 FSAR Sections 2.2.3.11 and 11.2.12. The HI-STORM 100 system is a large steel/concrete cask that will discharge lightning current through the steel shell of the overpack, to the ground. The conductive carbon steel overpack outer shell will provide a direct path to ground. Since the lightning current will discharge through the overpack, the MPC will be unaffected. Therefore, the HI-STORM 100 storage cask design meets the PFS Facility design criteria in Section 3.2.12 of the PFS Facility SAR for lightning protection

The staff has reviewed Sections 4.2.1.5.1(l) and (j), 4.2.2.5.1(l) and (j), 4.7.3.5.1(e), and 4.7.4.5.1(d) of the SAR and determined that the design of the cask storage pads provides fire and explosion protection while Sections 8.2.4.2 and 8.2.5.2 of the SAR show the capability of structures, systems, and components important to safety to withstand postulated fire and explosion accidents. The PFS Facility concrete storage pads are located on an open gravel surface and, therefore, will not be subject to wildfires. Therefore, consideration of fire loading on the storage casks is not necessary. The short duration of the 50-gal. diesel fuel fire does not produce a significant increase in the temperature of the massive concrete structure. A complete analysis of potential fires is presented in Chapter 15 of this SER. The explosive pressure loads for the PFS Facility site are identified as less than 1 psi. This is significantly less than the external pressure load of 60 psi used in the vendor Topical (T)SAR. The design of the storage cask has been shown to be acceptable under these greater load conditions.

Based on a review of the PFS site specific loads as discussed above, the staff concludes that the PFS Facility design criteria meet the loading conditions identified the HI-STORM storage cask design. A discussion of the cask design relative to the storage requirements of the PFS Facility is provided in SAR Chapter 4, Facility Design. The PFS Facility SAR provides a summary of the analysis performed in the HI-STORM 100 FSAR. The loading conditions at the Facility are enveloped by the loading conditions considered in the HI-STORM 100 FSAR (Holtec International, 2001), except for the seismic loads. Holtec International performed site-specific analysis of the HI-STORM 100 Cask system for thermal evaluation, multi-cask response under seismic loading, and drop/tipover analysis, and concluded that the PFS designs meet the requirements of 10 CFR 72.122.

The staff has reviewed these analyses and finds them acceptable

Transfer Cask and Associated Lifting Devices

The staff reviewed Sections 4.7.3, 8.2.1, 8.2.4, and 8.2.5 of the SAR which contain the structural analysis of the HI-TRAC transfer cask and associated lifting devices. The detailed structural analysis of the HI-TRAC transfer cask and the staff's evaluation are respectively provided in the HI-STORM 100 FSAR and the NRC's related SER. The discussion below is based in part on the results presented in the HI-STORM 100 FSAR and summarized in the PFS Facility SAR.

The transfer cask lifting trunnions are designed for a conservative total lifting load of 376,000 lb (150 percent of loaded transfer cask) using a two-point lift with a minimum safety factor of 10 based on the ultimate strength. During a lifting operation, no point in the HI-TRAC body exceeds its material yield strength. The structural analysis for the HI-TRAC transfer cask trunnions is described in the HI-STORM FSAR Appendix 3.E.

The HI-TRAC transfer cask, with the transfer lid attached, is designed to meet ASME Level A Subsection NF (American Society of Mechanical Engineers, 1998) stress limits while handling the dead load of the heaviest loaded canister. The structural analysis for the HI-TRAC transfer cask is described in HI-STORM 100 FSAR Appendix 3.AD.

Structural adequacy of the transfer cask trunnions was evaluated by modeling the trunnions as cantilevers and applying the weight of the loaded transfer cask. The resulting bending and shear stresses in the trunnions were combined to calculate the maximum principal stress and determine the corresponding safety factors. The structural analysis for the transfer cask trunnions is contained in the HI-STORM 100 FSAR.

The shipping cask and transfer cask lift yokes are designed as non-redundant lifting devices with a factor of safety of 10 or greater on material ultimate strength and 6 or greater on yield strength. A dynamic load increase factor of 10 percent has been applied to the lifting loads. Therefore, the lift yokes meet the NUREG–0612 stress limits for non-redundant special lifting devices.

The canister downloader is designed in accordance with NUREG–0612. The downloader consists of a hydraulic ram that is a non-redundant lifting device designed with the safety factors of 10 on ultimate strength and 6 on yield strength. The downloader uses two redundant

sets of anti-drop cam locks to secure the load in the event of a loss of power or hydraulic pressure.

The two canister lift cleats are designed with a minimum factor of safety of 3 on material yield strength and 5 on material ultimate strength, as well as a dynamic load increase factor of 10 percent. Each cleat can totally support the weight of the canister, thereby making them single-failure-proof per NUREG-0612. The cleats are connected to the canister via the 4 lifting bolts, 2 bolts per cleat. The lifting bolts are installed into threaded holes on top of the MPC lid. The MPC lifting analysis, which includes an analysis of the lifting bolts, is described in the HI-STORM FSAR.

The HI-STORM storage cask is designed to be lifted using four lifting lugs (threaded eyebolts) located on top of the cask. The lifting lugs screw into steel lifting blocks that are integrally welded to the storage cask steel. The stresses were compared with ASME III, Subsection NF allowable (American Society of Mechanical Engineers, 1998). The thread shear in the lifting block is compared to 10 percent of the ultimate strength of the base material in accordance with NUREG–0612. The lifting lugs have a net section stress below 10 percent of the ultimate strength of the lug material. The strength qualification analysis is described in HI-STORM FSAR Appendix 3.D. No credit is assumed for the concrete except as a vehicle to transfer compressive loads. A dynamic load factor of 1.15 is applied to simulate anticipated inertia forces during a low speed lift.

The canister hoist rings are designed with a minimum factor of safety of three on material yield strength and five on material ultimate strength, as well as a dynamic load increase factor of 10 percent. Eight rings provide redundant capability since only four are required, therefore, the hoist rings meet the NUREG–0612 requirements for redundancy.

The structural analysis demonstrates that the transfer cask and associated lifting devices are designed to resist the loads based on the site characteristics and environmental conditions during normal operations and during off-normal and accident events. The structural analysis demonstrates that the transfer cask and associated lifting devices are designed to withstand the effects of natural phenomena such as earthquakes, tornadoes, lightning, and floods without impairing the capability to perform safety functions.

The transfer cask has been evaluated for stability during a seismic event when in the stacked cask arrangement. It was concluded that it is necessary to secure the transfer, storage, and shipping casks to the cell walls throughout the transfer operation to prevent the casks from toppling during a seismic event. Therefore, seismic support struts are used to secure the casks to the cell walls when the casks are in a stacked arrangement.

Fire loading conditions of the HI-TRAC transfer cask are addressed in Section 11.2.4 of the HI-STORM 100 FSAR and in Section 8.2.5 of the PFS Facility SAR. As shown in Section 8.2.5 of the PFS Facility SAR, fires near a loaded transfer cask would have a small effect on the canister temperature because of the short duration of the fire accidents. A bounding cask temperature rise of less than 9.3 °F per minute was determined from the combined radiant and convection heat input to the cask. As a result, the fuel cladding was shown not to exceed the accident condition fuel cladding temperature limits. The elevated temperatures from a fire could cause the pressure in the transfer cask water jacket to increase and cause the overpressure relief valve to open and release water from the water jacket. Loss of water in the

HI-TRAC water jacket is analyzed in the HI-STORM 100 FSAR. The FSAR indicates that fuel cladding, MPC, and transfer cask temperatures would remain below the design temperature limits. The dose rates would not exceed the 10 CFR 72.106(b) whole body and organ-specific dose limits. The FSAR also indicates that the estimated occupational exposure for recovery of a damaged HI-TRAC transfer cask would be less than 2000 person-mrem and the 10 CFR Part 20 limits would be met.

Canister Transfer Overhead Bridge and Semi-Gantry Cranes

The cranes will be designed, fabricated, and tested in accordance with ASME NOG–1. The staff has reviewed Sections 4.7.2, 8.1.1, 8.1.4, 8.2.1, 8.2.4, 8.2.5, and 8.2.6 of the SAR and found that the design of the cranes to mitigate environmental effects is identified and that the capability of the cranes to withstand postulated accidents is demonstrated. The structural analyses of the canister transfer overhead bridge and semi-gantry cranes were performed by the applicant using a three-dimensional finite element model of the systems (Anatech Corporation, 1998a,b). In each case, the major structural members were sized based on the preliminary design and then adjusted to provide acceptable stress conditions in the members. The major structural elements were idealized as beam members with appropriate offsets to account for the physical relationships between the centroids of the various beam members. The restraints applied to the structural analysis model were in accordance with the procedures given in ASME NOG–1 (American Society of Mechanical Engineers, 1989) and NUREG–0554. The loading included dead loads, maximum suspended weight, and seismic loads. Load cases were run for each of the following conditions:

- Trolley at one end
- Trolley at ¼ span
- Trolley at ½ span
- Load at maximum height
- Load at minimum height

ASME NOG–1 is accepted by the NRC as a design specification for cranes. The Technical Specifications require that the overhead bridge crane and semi-gantry crane be classified as Type I cranes in accordance with ASME NOG-1, and that the allowable stresses used in the crane designs shall be in accordance with ASME NOG-1. Further, the Technical Specifications require that the cranes, and the canister downloader, be of single-failure-proof design and meet the requirements of NUREG-0554 and NUREG-0612.

Based on the crane design specifications and the Technical Specification requirements, there is reasonable assurance that overhead bridge and semi-gantry cranes will resist site-specific loads during normal operations and during off-normal and accident events, in accordance with the requirements of 10 CFR 72.122(b)(1). There is also reasonable assurance that the canister transfer overhead bridge and semi-gantry cranes will withstand the effects of natural phenomena such as earthquakes, tornadoes, lightning, and floods, without impairing the capability to perform safety functions in accordance with the requirements of 10 CFR 72.122(b)(2).

Seismic Support Struts

The staff has reviewed Sections 4.7.1.4.1 of the SAR and SC-10 (Stone & Webster Engineering Corporation, 1999b) and found that the design of the seismic struts to mitigate environmental effects is identified and that the capability to withstand postulated accidents is demonstrated. The seismic support struts secure the transfer, storage, and shipping casks to the cell walls when the casks are in a stacked arrangement during transfer operations. The seismic support struts prevent the casks from toppling or tipping over during a seismic event. The size of the struts and the design of the attachment to the building were based on the loads from an equivalent static seismic analysis (Stone & Webster Engineering Corporation, 1999b). ASME Subsection NF requirements for Class 2 nuclear grade support are accepted by the staff as a design specification of the support structures. The basic structure of the clevis used for connection of the seismic struts to the cask is based on standard end connections of Berge-Patterson Pipe Corporation. Also, as required in the Technical Specifications, the structural connection between the seismic support struts and the transfer cask, storage cask, and shipping cask will be sufficiently rigid to resist the design basis seismic motions. The structural analysis demonstrates that the seismic support struts are designed to resist the loads based on the site characteristics and environmental conditions during normal operations and during postulated off-normal and accident events, in accordance with the requirements of 10 CFR 72.122(b)(1). The structural analysis demonstrates that the seismic support struts are designed to withstand the effects of natural phenomena such as earthquakes, tornadoes, lightning, and floods, without impairing the capability to perform safety functions in accordance with the requirements of 10 CFR 72.122(b)(2).

Canister Transfer Building Structural Steel Roof Beams

The staff has reviewed SC-12 (Stone & Webster Engineering Corporation, 2001i) and found that the design of the Canister Transfer Building structural steel roof beams to mitigate environmental effects is identified and that the capability to withstand postulated accidents is demonstrated. Vertical loads are transferred to the building walls and structural steel columns that are supported by the mat foundation of the Canister Transfer Building. Horizontal seismic load from the roof mass is transferred to the building's walls by diaphragm action of the roof slab. Specific design and analysis is performed for the 25 ft and 30 ft span upper roof steel, girders G1, G2, and G3, the 17 ft to 30 ft span lower roof steel, roof support beams, columns, embeds, and connections.

Loads on the upper roof steel are based on a conservative spacing of 5.25 ft, which is greater than the actual spacing in all cases. Since the upper roof steel is welded to the decking it is considered to be continuously supported for positive moment. Therefore the allowable moment for the section can be fully realized. For negative moments the beam is unsupported and the section properties and procedures given in ANSI/AISC N690 are used. The appropriate sections were selected to insure that they have sufficient capacity to meet the demands under all loading conditions.

Tapered girders, 5 ft 3.75 inches at the centerline and 4 ft at the ends, are used to support the upper roof steel. The 63 ft long girders have a stiffened ½ inch thick web. The width and thickness of the flanges are adjusted for the three locations , G1 to G3, to account for variations

in demand loads. The design of the girders considers the appropriate load combinations and design considerations given in ANSI/AISC N690.

Design of the lower roof steel follows the same procedures as the upper roof steel. The appropriate sections were selected to assure that they have sufficient capacity to meet the demands under all loading conditions. The design of the lower roof steel considers the appropriate load combinations and design considerations given in ANSI/AISC N690.

The roof steel is supported by support beams. In addition to having sufficient moment and shear capacity, the deflection of these beams is shown to be within acceptable limits in accordance with the requirements of ANSI/AISC N690.

In addition to the reinforced concrete walls, the Canister Transfer Building structural steel roof members are supported in some locations by structural steel columns. These columns are designed considering axial and bending moments, along with combined loading conditions in accordance with the requirements of ANSI/AISC N690.

The roof beams are also supported by embeds into the reinforced concrete walls. These embeds consist of a series of 3/4 inch diameter studs welded to a plate. PFS stated that the capacity of the plate in shear will be calculated using a finite element analysis during the detailed design when the calculations for the wall have been finalized in SC-7. (Stone & Webster Engineering Corporation, 1998b) Stone & Webster in SC-12 calculation also identifies the details associated with the connection between various elements of the structural steel. Both the embeds and connectors are designed in accordance with the requirements of ANSI/AISC N690.

As identified in SC-12, the input accelerations, wind loads, and snowdrift will be verified by the applicant using the results of the updated finite element analysis of the Canister Transfer Building (SC-6) and the design of the reinforcing steel for the Canister Transfer Building (SC-7).

Based on a review of the analysis presented in SC-12, the staff concludes that the methods used by the applicant to analyze the Canister Transfer Building Structural Steel Roof for normal operations, off-normal, and accident events loads are reasonable, and meet the requirements of 10 CFR 72.24 and 72.122.

Canister Transfer Building Transfer Cell Doors

Stone and Webster in SC-14 determined that the design of the east and west sliding doors of the three canister transfer cells is adequate to ensure these doors are capable of resisting the required loadings. The loadings include the design basis ground motions for all doors and the tornado-missile loading for the west doors.

The doors are analyzed for earthquake loads using a static equivalent method and the peak earthquake acceleration. Computed stresses were compared with the requirements of the design code ANSI/AISC N690. Two bounding cases, one with the rolling doors in fully open, the other in fully closed positions were considered in the evaluation for the earthquake loads. For tornado missile protection, a finite element analysis was performed to determine the force

required to obtain a plastic moment in the door, and to develop the force-displacement relationship.

Based on a review of the analysis presented in SC-14, the staff concludes that the methods used by the applicant to analyze the Canister Transfer Building Transfer Cell Doors for normal operations, off-normal, and accident event loads are reasonable, and meet the requirements of 10 CFR 72.24 and 72.122.

5.1.5 Other Structures, Systems, and Components Not Important to Safety

This section contains a review of Sections 4.1, Summary Description; 4.3, Auxiliary Systems; 4.4, Decontamination Systems; 4.5, Shipping Casks and Associated Components; and 4.7.5, Cask Transporter of the SAR for the discussion on other structures, systems, and components not important to safety. There are no specific requirements identified in 10 CFR Part 72 for other structures, systems, and components not important to safety. Therefore, no evaluation findings are made in this section; only a discussion of the information provided in the SAR is given.

5.1.5.1 Description of Other Structures, Systems, and Components Not Important to Safety

This section describes the design, design criteria, and design analysis for other structures, systems, and components not important to safety. As identified in the SAR, the following structures, systems, and components are considered.

- Electrical systems (SAR Section 4.3.2)
- Air supply systems (SAR Section 4.3.3)
- Water supply system (SAR Section 4.3.5)
- Sewage treatment system (SAR Section 4.3.6)
- Communications and alarm systems (SAR Section 4.3.7)
- Fire protection system (SAR Section 4.3.8)
- Maintenance system (SAR Section 4.3.9)
- Propane fuel system (SAR Section 4.3.12)
- Stored fuel systems (SAR Section 4.3.13)
- Decontamination systems (SAR Section 4.4)
- Shipping casks and associated components (SAR Section 4.5)
- Cask transporter (SAR Section 4.7.5)

Descriptions of the other structures, systems, and components are given in the SAR sections identified. They are limited to a general description of the various systems. The majority of these systems will be based on commercially available systems which are designed, fabricated, constructed, tested, and maintained in accordance with approved engineering practices.

Section 4.3.1 of the SAR states that there are no ventilation or off-gas systems because of the use of a sealed canister design.

As identified in Section 4.3.2 of the SAR, normal electrical power will be provided to the Facility through an upgraded 12.5-kV offsite distribution power line. Lines installed at the site will be

according to the National Electric Code. The normal power will be provided for lighting, general utilities, security system, HVAC loads, crane loads, and miscellaneous equipment. Emergency backup power for up to 24 hours is provided at the Facility by a diesel-generator. An uninterruptible power source is utilized to support security loads for up to 1 hr until the diesel starts and comes up to speed. Restricted area lighting is provided to maintain a minimum lighting distribution at the Facility according to the requirements of 10 CFR 73.50.

As identified in Section 4.3.3 of the SAR, an air supply is provided at the Facility for maintenance purposes. The system will be designed and installed in accordance with ASME B31.1.

Section 4.3.4 of the SAR states that there is no requirement for a steam supply and distribution system at the Facility.

As identified in Section 4.3.5 of the SAR, a water supply is provided at the Facility for normal facility services, operation, and maintenance functions. Surface tanks supplied by onsite wells will be used to supply the necessary water. Backup sources of water are identified. The water distribution piping and plumbing will be provided in accordance with the Uniform Plumbing Code.

As identified in Section 4.3.6 of the SAR, a sanitary drainage system will be provided at the Facility in accordance with the Uniform Plumbing Code.

As identified in Section 4.3.7 of the SAR, the communications systems consist of normal telephone service in all the buildings, a site public address system, and a short-wave radio system for security. In the event of an emergency, Facility personnel and onsite visitors will be notified by an announcement over the onsite communications system. Alarms at the Facility are used on area radiation monitors to notify nearby personnel of doses that exceed the alarm limits. The communication system is designed to meet the requirements of 10 CFR 73.51.

Section 4.3.8 of the SAR contains a discussion of the fire protection system for the facility. For the Canister Transfer Building, the system will be designed in accordance with NFPA 801. Where the UBC has more stringent requirements, these requirements will also be met. The Canister Transfer Building is divided into three fire zones, which correspond to the specific occupant classifications. The fire zones are shown in Figure 4.3-1 of the SAR.

Fire Zone 1 consists of the transfer cells, crane bay, cask load/unload bay, and the cask transport bay. The fire source for this region is 50 or 300 gal. of diesel fuel. The cask load/unload bay, which could experience a 300 gal. diesel fuel fire, will utilize a foam-water sprinkler system for fire protection in accordance with NFPA 16 (National Fire Protection Association, 1999a). The required number or foam-water sprinkler zones and sump volume have been identified. In addition to the foam-water sprinkler system, the design of the walls and sliding doors between the canister transfer cells and the cask transporter bay are fire rated. The transfer cell rooms will not be provided with automatic fire suppression systems to prevent dislodging of external radioactive material on the canisters. Areas within the transfer cells will be reachable by a firewater stream in the unlikely event a fire occurs. The crane bay, cask transporter bay, and transfer cells will contain fire extinguishers for fire suppression.

Fire Zone 2 includes the low-level waste storage room. This area will only use fire extinguishers for fire suppression.

Fire Zone 3 includes the office and building services areas of the building. This area will only use fire extinguishers for fire suppression.

The Canister Transfer Building is constructed of noncombustible materials as identified by NFPA 220 (National Fire Protection Association, 1999b). The building is designed to limit the potential effects from a diesel fuel fire with curbs and sloped floors installed so as to contain the spilled diesel fuel away from structures, systems, and components important to safety.

The Security and Health Physics Building fire protection provisions will be designed in accordance with the requirements of the UBC and NFPA 101 (National Fire Protection Association, 1997) as applicable. A fire suppression system will be provided in the diesel generator room.

Section 4.3.9 of the SAR states that the Facility has relatively few maintenance requirements because of the passive nature of the storage system design. Routine maintenance procedures ensure that timely maintenance is performed according to the equipment manufacturer's standards.

As identified in Section 4.3.10 and 4.3.11 of the SAR, there are no cold chemical or air sampling systems required at the Facility.

Section 4.3.12 of the SAR states that propane fuel for all gas heating units are located on the PFS Facility site. The location of these is such that the explosive pressures at structures, systems, and components important to safety are less than 1 psi. The propane storage tanks are above ground and are designed in accordance with the requirements of NFPA 58 (National Fire Protection Association, 1998).

Section 4.3.13 of the SAR states that all diesel fueling at the Facility comply with applicable regulations. Operation and use of the stored diesel fuel will be in accordance with 29 CFR 1910 Occupational Safety and Health Administration regulations to insure employee health and safety requirements are met. The outdoor diesel fuel tank will be designed in accordance with UL–2085 (Underwriters Laboratories Inc., 1997). This code requires that the tank meet 2-hr liquid pool furnace fire test, vehicle impact, and projectile resistance criteria.

SAR Section 4.4, Decontamination Systems, indicates that decontamination of equipment is not required at the Facility because of the sealed nature of the spent nuclear fuel canisters. Contamination of personnel is not expected to occur under normal conditions of operation. Under off-normal conditions, decontamination will be performed using methods that only result in the generation of dry active waste.

Spent fuel shipping casks are used to transport the spent fuel canisters from the originating power plants to the Facility and later offsite. The shipping casks and associated components, as identified in Section 4.5 of the SAR, are not licensed under 10 CFR Part 72. During rail or trailer transport to the Facility, the spent fuel must be packaged in accordance with 10 CFR Part 71. The HI-STAR Cask System, which is approved under 10 CFR Part 71 (Certificate of Compliance 9261, Docket No. 71-9261), is designed to transport the same

sealed metal canister that is stored in the HI-STORM 100 Cask System. Discussion of the Skull Valley Road intermodal transfer point and Low corridor rail line are provided in the SAR.

Section 4.7.5 of the SAR identifies the cask transporter as being used to move the loaded storage cask between the Canister Transfer Building and the storage pad. The cask transporter, Figure 4.7-4 of the SAR, is a commercial grade system that has no specific code or specification criteria. The transporter travels up to 2 mph, has a capacity of 200 tons, and has a maximum weight of approximately 185,000 lbs. The transporter is designed to mechanically limit the lifting height of a canister to a maximum of only 10 in. The transporter is also designed to preclude tipover during a design basis earthquake or if impacted by a design basis tornado missile.

5.1.5.2 Design Criteria for Other Structures, Systems, and Components Not Important to Safety

The design criteria for the various other structures, systems, and components not important to safety that have been identified in the SAR are given in the previous section. These supplement the design criteria identified in Chapter 3 of the SAR. Table 4.1-1 of the SAR identifies details of the Facility's compliance with the general design criteria of 10 CFR Part 72 Subpart F.

The design criteria identified for other structures, systems, and components are based on commonly used codes and standards. The design of the other structures, systems, and components permits inspection, maintenance, and testing. The inspection, maintenance, and testing requirements are based on the appropriate codes and standards. This design also allows for emergency capability. The layout of the facility allows areas to be reached in the event of an accident.

Design code compliance for the fire protection systems include the latest code in effect at the time of the design as described below:

- Foam-water sprinkler systems will be designed in accordance with NFPA 16.
- The sprinkler system for the diesel generator room will be designed in accordance with NFPA 13.
- The fire pumps and water supply tanks will be provided in accordance with NFPA 20 and NFPA 22, respectively.
- The portable fire extinguishers will be provided in accordance with NFPA 10.
- The fire protection equipment at the Facility will be maintained in accordance with NFPA 25.

5.1.5.3 Material Properties for Other Structures, Systems, and Components Not Important to Safety

No specific material properties are identified in the SAR for the other structures, systems, and components not important to safety. However, material properties must satisfy the code or standards used for the structures, systems, and components as required.

5.1.5.4 Structural Analysis for Other Structures, Systems, and Components Not Important to Safety

As described in the previous section, other structures, systems, and components not important to safety will be designed based on standard engineering practice in accordance with the applicable codes and standards. In most cases, these structures, systems, and components are commercially available, and their design to standard industrial requirements is acceptable.

5.2 Evaluation Findings

Based on the review of the SAR, the staff has made the following determinations.

- Information regarding HI-STORM 100 cask-specific structures, systems, and components important to safety is in the HI-STORM 100 Cask System FSAR. The NRC's approval of the HI-STORM 100 Cask System is documented in Certificate of Compliance 1014 and the related HI-STORM 100 SER.

- There will not be a pool or pool confinement facility at the proposed PFS Facility.

- The SAR adequately describes all structures, systems, and components that are important to safety, providing drawings and text in sufficient detail, to allow evaluation of their structural effectiveness to meet the requirements of 10 CFR 72.24(b) and (c). The structural analysis procedures used by PFS have been identified. The relationship between the design basis and the design criteria have been identified. The materials of construction are identified. The applicable codes and standards used in the analysis of the reinforced concrete structures have been established.

- The structures, systems, and components important to safety are designed, fabricated, erected, and tested to quality standards commensurate with the importance to safety functions to be performed. The structures, systems, and components important to safety are classified based on their primary function and importance to overall safety. Therefore, the requirements of 10 CFR 72.122(a) are satisfied.

- The structures, systems, and components important to safety are designed to accommodate the combined loads of normal, off-normal, accident, and natural phenomena events with an adequate margin of safety. The structural analysis performed by PFS demonstrates that structures, systems, and components important to safety are designed to resist the loads based on the site characteristics and environmental conditions under normal operations and under

postulated off-normal and accident events. The PFS structural analysis demonstrates that structures, systems, and components important to safety are designed to withstand the effects of natural phenomena, such as earthquakes, tornadoes, lightning, and floods, without impairing the capability to perform safety functions. Stresses at critical locations of structures, systems, and components for bounding design loads are determined by analysis. The section properties are adjusted to ensure that the capacity of all structural elements at all locations exceeds the demand. Total stresses for the combined loads of normal, off-normal, accident, and natural phenomena events are acceptable and found to be within the limits of applicable codes, standards, and specifications. Analysis of the structural response of the storage cask to earthquake loading was based on generic loading identified in the HI-STORM 100 FSAR. The site-specific cask stability analysis performed by Holtec International demonstrates the storage cask will not tipover, collide, or slide off the storage pad during a site-specific design basis earthquake. The loads on the MPC and fuel assemblies remain bounded by the loads considered in the HI-STORM 100 FSAR. Therefore, the requirements of 10 CFR 72.122(b)(1) and (2) are satisfied.

- The structural analysis demonstrates that the Canister Transfer Building is designed to continue to perform its safety-related functions effectively under credible fire and explosion conditions. As identified in the design criteria, the overpressure from explosion is bounded by the pressure and stress due to tornado wind and seismic loading. The structures, systems, and components important to safety are located on an open gravel surface and, therefore, will not be subject to wildfires. Therefore, the requirements of 10 CFR 72.122(c) are satisfied.

- The descriptions of structures, systems, and components important to safety include consideration of inspection, maintenance, and testing. Components requiring inspection and maintenance are identified, and operational procedures are summarized. Therefore, the requirements of 10 CFR 72.122(f) are satisfied.

- This design also allows for emergency capability in that access to critical locations and regions in the event of emergencies is possible. In addition, the lifting components are designed to hold the load in the event of emergencies. Therefore, the requirements of 10 CFR 72.122(g) are satisfied.

- The design allows for handling and storage of the limited radioactive waste generated at the ISFSI within the Low-level waste storage room. Therefore, the requirements of 10 CFR 72.128(b) are satisfied.

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6 THERMAL EVALUATION

6.1 Conduct of Review

Review of the thermal evaluation included Sections 3.3.6, Fire and Explosion Protection, and 3.3.7, Materials Handling and Storage, of the SAR. The design of the proposed Facility is based on the use of the HI-STORM 100 Cask System as certified by the NRC (Nuclear Regulatory Commission, 2000a) and as described in the FSAR for the HI-STORM Cask System (Holtec International, 2000).

6.1.1 Decay Heat Removal Systems

The HI-STORM 100 Cask System is designed to remove decay heat primarily by convective heat transfer. No active cooling systems are used. The storage cask is equipped with four inlet vents at the bottom and four inlet vents on top. Cool air is drawn into the annulus between the canister and storage cask through the bottom inlet vents. The buoyancy created by the heating of the air creates a chimney effect and the air flows back into the environment through the outlet vents at the top of the cask. The Technical Specifications include surveillance requirements for ensuring that the cask heat removal system is operational during storage (i.e., the air ducts are inspected or the outlet temperature is measured every 24 hours to ensure that the ducts are free of blockages). The cask system design and heat removal capability are described and evaluated in the HI-STORM 100 FSAR. As documented in the NRC's HI-STORM 100 SER (Nuclear Regulatory Commission, 2000b), the staff has previously determined that the HI-STORM 100 Cask System provides adequate heat removal capacity.

6.1.2 Material Temperature Limits

The material temperature limits for the HI-STORM 100 Cask System are given in the HI-STORM 100 FSAR, which has been reviewed and found to be acceptable by the staff.

6.1.3 Thermal Loads and Environmental Conditions

The staff has reviewed the information presented in Section 2.3, Meteorology, and Section 3.2.6, Thermal Loads, of the SAR. The staff reviewed the discussion on thermal loads and environmental conditions with respect to the following regulatory requirements

- 10 CFR 72.92(a) requires that the natural phenomena that may exist or that can occur in the region of a proposed site be identified and assessed according to their potential effects on the safe operation of the ISFSI. The important natural phenomena that affect the ISFSI design must be identified.

- 10 CFR 72.122(b) requires that (1) structures, systems, and components important to safety be designed to accommodate the effects of, and to be compatible with, site characteristics and environmental conditions associated with normal operation, maintenance, and testing of the ISFSI and to withstand postulated accidents. (2) Structures, systems, and components important to safety must be designed to withstand the effects of natural phenomena such as earthquakes, tornadoes, lightning, hurricanes, floods, tsunami, and seiches,

without impairing their capability to perform safety functions. The design bases for these structures, systems, and components must reflect: (i) Appropriate consideration of the most severe of the natural phenomena reported for the site and surrounding area, with appropriate margins to take into account the limitations of the data and the period of time in which the data have accumulated, and (ii) Appropriate combinations of the effects of normal and accident conditions and the effects of natural phenomena. The ISFSI should also be designed to prevent massive collapse of building structures or the dropping of heavy objects as a result of building structural failure on the spent fuel or high-level radioactive waste or on to structures, systems, and components important to safety. (3) Capability must be provided for determining the intensity of natural phenomena that may occur for comparison with design bases of structures, systems, and components important to safety.

The temperatures recorded at different sites are summarized in Table 2-1 of this SER. This table has been developed from Table 2.3-4 of the SAR and data presented in Ashcroft et al. (1992). The data availability period for Salt Lake City in Table 2.3-4 of the SAR is 1951–1980. Similarly, 1950–1992 and 1951–1958 information was available for Dugway and Iosepa South Ranch, respectively. Ashcroft et al. (1992) reported meteorological data for 1950–1992 and 1948–1992 for Dugway and Salt Lake City, respectively. The maximum recorded annual average temperature at the site (average of temperatures recorded both day and night throughout the year) is 49 °F. At Dugway, 12 mi south of the proposed site, the annual average temperature is 51 °F. The average daily maximum temperature is defined as the average of peak temperatures throughout the hottest month, July. The highest average daily temperature is 95 °F, recorded at Iosepa Ranch, 8 mi northwest of the proposed site. Based on information from the National Oceanic and Atmospheric Administration, the lowest ambient temperature recorded near the site is -30 °F, at Salt Lake City. The HI-STORM 100 Cask System was evaluated for an average annual temperature of 80 °F and temperature extremes of -40 °F and 125 °F. The temperatures at and around the PFS Facility site are bounded by the temperatures for which the HI-STORM 100 Cask System was evaluated.

The maximum solar insolation recorded at the site is 685 W/m^2 during a 12-hr period. Also, the maximum solar insolation recorded at Salt Lake City is 730 W/m^2 during a 12-hr period. The HI-STORM 100 Cask System has been evaluated for a bounding solar insolation value of 775 W/m^2 during a 12-hr period, per 10 CFR Part 71.

The staff reviewed the local meteorological data and discussions presented in the SAR and found these acceptable because reliable data sources, such as the National Weather Service, were used, and the data are appropriately summarized. The applicant adequately presented information regarding temperatures recorded during the onsite measurement program and at other nearby sites, and, therefore, satisfied the requirement of 10 CFR 72.92(a). The staff confirmed that temperatures and solar loads at the site are bounded by the HI-STORM 100 Cask System design parameters.

6.1.4 Analytical Methods, Models, and Calculations

The analytical methods, models, and calculations for the thermal evaluation of the HI-STORM 100 Cask System are described in detail in the HI-STORM 100 FSAR. As documented in the NRC's HI-STORM 100 SER, the staff has previously reviewed these methods, models, and calculations and found them acceptable.

6.1.5 Fire and Explosion Protection

6.1.5.1 Fire

The information presented in Section 3.3.6, Fire and Explosion Protection; Section 3.2.12, Lightning; Section 4.3.8, Fire Protection System; Section 4.3.12, Gas Utilities; and Section 8.2.5, Fire; in connection with the protection against potential onsite fire and wildfire has been reviewed for conformance with the following regulatory requirement.

- 10 CFR 72.122(c) requires that structures, systems, and components important to safety be designed and located so that they can continue to perform their safety functions effectively under credible fire and explosion exposure conditions. Noncombustible and heat-resistant materials must be used wherever practical throughout the ISFSI, particularly in locations vital to the control of radioactive materials and to the maintenance of safety control functions. Explosion and fire detection, alarm, and suppression systems shall be designed and provided with sufficient capacity and capability to minimize the adverse effects of fires and explosions on structures, systems, and components important to safety. The design of the ISFSI must include provisions to protect against adverse effects that might result from either the operation or the failure of the fire suppression system.

Facility Restricted Area

The proposed Facility is located on an open gravel surface. Figure 1.2-1 of the SAR gives the layout of the restricted area of the Facility. Concrete storage pads will be separated from the inner fence of the Facility surrounding the restricted area by a minimum distance of 150 ft. The restricted area not covered by the storage pads will have 12-in.-deep crushed rock (Cooper, 1999). The outer fence is separated from the inner fence by 20 ft. The isolation zone (the region between the fences) will also be covered with 12-in.-deep crushed rock. The 20-ft-wide perimeter road, located 10 ft from the outer fence, will also be covered by 12 in.-deep crushed rock. The 10-ft wide zone between the perimeter road and the outer fence will also have a 12-in.-deep crushed rock surface. A maintenance program will control any significant growth of vegetation through the crushed rock. Therefore, the surface of the restricted area, including the region up to the perimeter road, will be noncombustible. Additionally, there will be a 300-ft-wide barrier of crested wheat grass around the restricted area to protect the Facility from wildfires. This barrier would remain in place with little maintenance after the wheat grass has been planted. Crested wheat grass is fire resistant and will greatly reduce the effects of any wildfire approaching the Facility. The staff finds that the width of the fire barrier is sufficient to protect important to safety structures, systems, and components from the effects of wildfires. It may be noted that the U.S. Fire Administration (1993) recommends a 100-ft-wide fire barrier for protecting houses in a pine forest.

Canister Transfer Building

The Canister Transfer Building is made of concrete walls 2 ft thick, a concrete roof 1 ft thick and a 5-ft-thick concrete foundation (Stone & Webster Engineering Corporation, 1998a,b). In SAR Section 4.3.8.1, PFS characterized the Canister Transfer Building as multiple purpose occupancy and classified it as a Type II Fire Rated construction following the UBC (International Conference of Building Officials, 1997) and a construction Type II structure in accordance with NFPA 220, Standard Types of Building Construction (National Fire Protection Association, 1999a), as referenced in NFPA 801, Standard for Fire Protection for Facilities Handling Radioactive Material (National Fire Protection Association, 1998a). UBC provides the minimum standards to safeguard life, health, property, and public welfare by regulating and controlling the design, construction, occupancy, and quality of materials of a building. NFPA 220 provides definitions for standard types of building construction based on the combustibility and the fire resistance rating of a building's structural elements. NFPA 801 addresses the requirements for fire protection at facilities handling radioactive materials to reduce the risk of fires and explosions. Because nuclear materials will be handled in the Canister Transfer Building, the fire protection systems will be designed in accordance with NFPA 801. To meet any fire insurance requirements, however, PFS will design the Canister Transfer Building to envelope UBC requirements if found to be more stringent. The building is designed to limit the potential effects from a diesel fire by providing curbs and sloped floors to contain spilled diesel away from the structures, systems, and components important to nuclear safety.

The Canister Transfer Building, including the cask load/unload bay and cask transfer cells, and other buildings in the Facility will be constructed with automatic fire detection systems. Photosensitive smoke detectors will be placed in all buildings to detect any fire. The smoke detectors will be interconnected within each building and connected to a central alarm panel located in the Security and Health Physics Building. A trip of the fire detection system over the cask load/unload bay will automatically engage the foam-water sprinkler system. The fire-suppression systems and equipment consist of sprinkler systems in the Canister Transfer Building and the Security and Health Physics Building, portable fire extinguishers, hose reels, fire hydrants, fire truck, fire pumps, and water supply tanks. The Canister Transfer Building is divided into three fire zones, based on specific occupant classifications following NFPA 101, Life Safety Code (National Fire Protection Association 1997a). NFPA 101 addresses life safety from fire and similar emergencies. The code provides requirements for construction, protection, and occupancy features necessary to minimize danger to life from fire, including smoke, fumes, and panic. Fire Zone 1 is classified as a Special Purpose Industrial Occupancy and consists of the transfer cells, crane bay, cask load/unload bay, and the cask transporter bay. The low-level waste storage room is included in Fire Zone 2 and classified as a Storage Occupancy, based on NFPA 101. Fire Zone 3 consists of the office and services areas of the building and is classified as a Business Occupancy. Figure 4.3-1 of the SAR illustrates these zones in the Canister Transfer Building. The fire zones will be separated by 1-hr fire rated barrier walls and fire doors, as recommended by UBC.

The transfer cells and crane bay in Fire Zone 1 will not store any combustible liquid. The cask transporter bay could have up to 50 gal. of diesel (Class II combustible liquid) in the cask transporter. However, during transfer operations, the cask transporter will be prevented from entering a transfer cell by the shield doors at either side of the cell. The shield doors will remain closed when canister transfer is in progress by PFS Facility administrative procedures. A cask

transporter can only enter a transfer cell when the canister is either in the shipping cask or in the storage cask with the lid bolted in place. The cask load/unload bay will have the combustible fuel load of 300 gal. of diesel in the heavy haul tractor/trailer. The heavy haul vehicles will enter and exit the cask load/unload bay at the south end of the Canister Transfer Building and will not approach a cask transfer cell. By Facility administrative procedure, train locomotives are required to stay out of the Canister Transfer Building. The Canister Transfer Building and its surroundings are designed in such a way that any fuel spilled outside the building will not flow into the building.

A foam-water sprinkler system will be installed in the cask load/unload bay in accordance with NFPA 16, Standard for the Installation of Foam-Water Sprinkler and Foam-Water Spray Systems (National Fire Protection Association, 1999b). NFPA 16 provides the minimum requirements for the design, installation, and maintenance of foam-water sprinkler and spray systems. Water will flow into the piping system when the foam-water sprinkler system is activated. Foam will be injected into the water resulting in a foam solution being discharged through special sprinkler heads. Foam-water discharge will continue until manually shut off.

Total floor area of the cask load/unload bays is approximately 10,250 sq ft. NFPA 16 allows a maximum of 5,000 sq ft per zone. PFS has selected three foam-water sprinkler zones for the Canister Transfer Building. One zone includes the low bay section of the east half of both bays. A second zone covers the low bay section of the west half of both bays. The high bay section in the middle of the building is the third zone. The area of each low bay zone is approximately 3,500 sq ft. The high bay has an area of 3,250 sq ft. Additionally, the floor of the bay area will be sloped toward one of two sumps located at the center of each low bay zone.

The depth of each sump will be 6 ft. Each sump will be 60 ft long and 6 ft wide with floors that slope downward at a rate of 0.25 in./ft for 24 ft toward a deep end of the bays. The sumps would be away from the locations where a shipping cask and the crane lifting cables will be located. Dimensions of the sump will be adequate to accommodate 300 gal. of diesel fuel spill and 30 min of discharge of the sprinkler system at the discharge rate allowed. The threshold between the crane bay and load/unload bay will be 1 in. high to retain any spilled diesel. The rise of the threshold will be gradual for a 2 ft wide area to avoid personnel tripping hazards.

The walls of the transfer cells will be 2-hr fire rated and the sliding doors will be 2-hr fire rated to prevent any fire in the transporter bay from affecting an exposed canister during transfer operation. No sprinklers will be located in the transfer cells to avoid inadvertently spraying down the canister and dislodging any contamination on the canister. The transfer cells will primarily contain the storage or shipping cask. The only ignition source that can enter the transfer cell is the cask transporter which can contain up 50 gal of diesel fuel. This situation will only arise after the canister is contained in the concrete storage cask. The HI-STORM 100 FSAR has demonstrated that this configuration can adequately withstand a 50-gal fire.

Additionally, the floor of the transfer cells will slope away from the cells to ensure that any spill of diesel in the transporter bay does not flow into the cells. Moreover, PFS will install 100-ft-long hose reels adjacent to the crane bay and the cask transporter bay exit locations to ensure that all areas within the transfer cells are accessible by water in case of a fire.

Fire Zone 2 is a low-level waste (low-level waste) storage area and will store health physics survey samples and dry wipes to be used for removing radioactive contamination of the storage

casks in 55 gal. sealed drums. The holding cell is designed to maintain radiation levels ALARA and will store a few low-level waste containers before they are shipped offsite. Further evaluation of the low-level waste holding cell is given in Chapter 14 of this SER. This zone does not require a fire sprinkler system and will only have fire extinguishers for fire protection.

Fire Zone 3 will have smoke detection and fire extinguishers for fire protection.

The fire detection system is designed in accordance with NFPA 72, National Fire Alarm Code (National Fire Protection Association, 1999c). NFPA 72 includes the application, installation, location, performance, and maintenance of fire alarm systems and their components. This code defines the means of signal initiation, transmission, notification, and annunciation; the levels of performance; and the reliability of the various types of fire alarm systems.

Portable fire extinguishers are provided throughout the Canister Transfer Building in accordance with NFPA 10, Standard for Portable Fire Extinguishers (National Fire Protection Association, 1998b). NFPA 10 provides the requirements for selection, installation, inspection, maintenance, and testing of portable extinguishing equipment. Requirements in NFPA 10 are minimum and fire extinguishers are intended to cope with fires of limited size.

Fire pumps located outside the restricted area will supply water to the sprinkler system, fire hydrants, and hose reels. Water is supplied by a primary and backup water tanks. Water for the foam-water sprinkler system will be fed from one of two fire pumps placed in a pump house outside the restricted area near the Security and Health Physics Building. The foam supply will be connected to the water lines outside the Canister Transfer Building. One pump will be powered by an electrical motor and the other by a diesel engine in case of loss of electric power. The fire pumps and water supply tanks are provided in accordance with NFPA 20, Standard for the Installation of Centrifugal Fire Pumps (National Fire Protection Association, 1999d) and NFPA 22, Standard for Water Tanks for Private Fire Protection (National Fire Protection Association, 1998c), respectively. NFPA 20 gives the requirements for selection and installation of pumps supplying water for fire protection. Requirements have been established for design and installation of these pumps, pump drivers, and associated equipment. NFPA 22 provides the minimum requirements for design, construction, installation, and maintenance of tanks and accessory equipment that supply water for fire protection. Fire hydrants are located near the buildings to support fire suppression. At least one fire truck will be located at the Facility site, and one truck will be at the Goshute Village, 3.5 mi from the site. Smoke from a fire in the Canister Transfer Building will be removed by the exhaust fans.

Section 8.2.5.2 of the SAR states that a fire involving 300 gal. of diesel would not threaten the integrity of a canister in a shipping cask. The shipping casks are required to safely withstand the effects of a fire burning at 1,475 °F for 30 min, as per 10 CFR 71.73(c)(4). A fire involving 50 gal. of diesel from a cask transporter would burn a maximum of 5 min without any assistance from the sprinkler system to extinguish it.

Security and Health Physics Building

The fire protection systems of the Security and Health Physics Building will be designed in accordance with the requirements of UBC and NFPA 101, as applicable. The building has been classified as Group B for business related functions, with an occupancy of less than 50

persons and floor area of less than 12,000 sq ft. The building construction is classified as Type II-N (i.e., non-combustible) based on UBC guidelines. No automatic fire-suppression systems are required by UBC for this building.

A double-wall sub-base diesel tank, designed according to NFPA 37, Standard for the Installation and Use of Stationary Combustible Engines and Gas Turbines (National Fire Protection Association, 1998d), will be located in the Security and Health Physics Building to provide fuel for the backup diesel generator. NFPA 37 specifies the design and spill control requirements for storage tanks. The capacity of the diesel generator will not be more than 150 kW with maximum fuel consumption at the rate of approximately 12 gal./hr. To provide a minimum of 24-hr backup power plus the required 30-min monthly tests, PFS estimates 350 gal. of diesel need to be stored in the tank. More discussion on the diesel storage tanks is given in the following section.

The anticipated capacity of the diesel tank in the Security and Health Physics Building for providing fuel for the backup diesel generator is 350 gal. and exceeds the exempt amount of 120 gal. for Class II combustible liquid prescribed by UBC. Therefore, a water sprinkler will be installed in the diesel generator room in accordance with NFPA 13, Standard for the Installation of Sprinkler Systems (National Fire Protection Association, 1999e). NFPA 13 provides the minimum requirements for the design and installation of automatic water sprinkler systems and exposure protection sprinkler systems, including the character and adequacy of water supplies and the selection of sprinklers, fittings, pipings, valves, and all materials and accessories, together with the installation of private fire service mains. The purpose of this standard is to provide a reasonable degree of protection for life and property from fire through standardization of design, installation, and testing requirements for sprinkler systems, and private fire service mains, based on sound engineering principles, test data, and field experience. The diesel generator room will be separated from all other adjacent interior spaces by a 1-hr rated fire barrier in accordance with UBC.

Diesel Storage Tanks

A diesel fuel oil storage tank will be located within the restricted area for refueling onsite vehicles including the cask transporters. This tank will have a capacity of approximately 1,000 gal. of low-grade sulfur No. 2-D diesel fuel. This tank will be placed approximately 200 ft from the Canister Transfer Building and approximately 700 ft from any storage cask. This above ground tank will have a double wall for primary and secondary spill containment requirements, fill and venting requirements, and fire prevention requirements in accordance with NFPA 30, Flammable and Combustible Liquids Code, (National Fire Protection Association, 1996). The tank will be surrounded with dikes to contain fuel in the event of a leak or spillage. NFPA 30 applies to the storage, handling, and use of flammable and combustible liquids. The code also includes strategies for the protection against flammable liquid hazards (e.g., storage limits and sprinkler designs). The tank will be designed in accordance with the requirements of UL-142, Steel Aboveground Tanks for Flammable and Combustible Liquids (Underwriters Laboratories Inc., 1993) and UL–2085, UL Insulated Aboveground Tanks for Flammable and Combustible Liquids (Underwriters Laboratories Inc., 1997). UL–2085 requires tanks be constructed to limit the heat transferred to the primary tank when exposed to a 2-hr hydrocarbon pool fire. The primary tank should also be provided with additional protection against impact from projectiles

and vehicles. A regional bulk fueling service will deliver diesel to the Facility approximately every two weeks with a tanker truck.

The diesel storage tanks will be located at least 50 ft inside the inner fence of the restricted area. Consequently, there will be approximately 100 ft of firebreak between the outer edge of the perimeter road and the tank. An additional firebreak of approximately 300 ft will be provided by the crested wheatgrass barrier. This barrier will run around the restricted area. The Security and Health Physics Building will be approximately 50 ft inside the crested wheat barrier. Therefore, the emergency diesel generator tank located inside this concrete building would be approximately 350 ft from a potential wildfire. Based on these distances from the potential wildfire, availability of firebreaks, and construction of the storage tanks, the staff concludes that the potential impact of wildfires will be negligible on the diesel storage tank and the emergency generator tank.

Locomotives

A fire associated with the diesel fuel spill from locomotives is not likely given the low travel speed and difficulty in igniting spilled diesel fuel. In addition, PFS will engineer the slope of the terrain to retain a spill of 6,400 gallons of diesel fuel near the rail line, as discussed in Section 15.1.2.4 of this SER. Two coupled main line locomotives on the north rail line can come within 100 ft of the nearest cask storage pads. These railroad locomotives, manufactured by General Motors Electro-Motive Division, are assumed to be model SD–40–2, Type C-C. Each of these locomotives is rated at 3,000 continuous horsepower and carries 3,200 gal. of diesel in two tanks located underneath the locomotive. The locomotives will be fueled outside the restricted area by a regional bulk fueling service. The fuel service must comply with the U.S. Environmental Protection Agency and Occupational Safety and Health Administration regulations and will be responsible for containment and clean up of any spills in accordance with the applicable regulations.

By Facility administrative procedures, the locomotives will not enter the Canister Transfer Building. PFS proposes to use railroad switching locomotive model MP–15AC to push the 150-ton depressed center flat car carrying the spent fuel shipping cask inside the Canister Transfer Building. The locomotive, manufactured by General Motors Electro-Motive Division, has 1,500 hp with an 1100 gal. diesel fuel tank. The shipping cask car has a coupled length of approximately 74 to 105 ft. A spacer car of coupled length of approximately 66 ft will be placed in between the shipping cask car and the locomotive. The rail cars will enter the Canister Transfer Building through the west doorway of the cask load/unload bay. The distance from the west doorway to the location where the shipping cask would be positioned for hoisting is approximately 103 ft. The nearest end of the locomotive fuel tank will be approximately 20 ft outside the Canister Transfer Building. By Facility administrative procedures, the locomotive engineers would be instructed to keep the locomotives outside the Canister Transfer Building. Additionally, PFS proposes to install wheel stops onto the rails in the cask load/unload bay east of the bay centerline to physically prevent locomotives from entering into the Canister Transfer Building.

Propane Storage Tanks

Propane for heating the Canister Transfer Building and the Security and Health Physics Building will be stored in a group of four centralized tanks. Each tank will have a capacity of 5,000 gal. or less so that the combined capacity of all four tanks will be not more than 20,000 gal. The four storage tanks will be separated by missile walls to ensure that a single tornado missile cannot rupture more than one tank. The tanks will be located outside the restricted area and at least 1,800 ft away from the nearest cask storage pads and the Canister Transfer Building and approximately 1,000 ft west of the Operations and Maintenance Building. Additionally, propane for heating the Operations and Maintenance Building and the Administration Building will be stored in relatively small propane tanks located near these structures. PFS specifies that a crushed rock surface devoid of vegetation will be placed a minimum of 100 ft radially outward from the propane tanks to stop propagation of wildfires. Evaluation of potential detonation of the propane in Chapter 15, Accident Analysis, of this SER shows that the generated air overpressure at a distance of 1,800 ft will be less than 1 psi, the recommended safe limit for structural damage by Regulatory Guide 1.91 (Nuclear Regulatory Commission, 1978). These above ground storage tanks will be designed in accordance with the requirements of NFPA 58, Liquified Petroleum Gas Code (National Fire Protection Association, 1998e). NFPA 58 provides the requirements for construction of liquefied petroleum gas storage tanks. It also requires a minimum distance of 50 ft from any nearby building for propane tanks having capacity 2,001–30,000 gal. Heating systems will be designed following NFPA requirements. Additionally, all outdoor pipes between the tanks and the buildings will be located below ground.

Inspection, Testing and Maintenance

The fire protection systems including the foam-water sprinkler system at the Canister Transfer Building, the sprinklers in the diesel generator room at the Security and Health Physics Building, yard hydrants, fire pumps, water storage tank, service mains, and all associated components will be maintained in accordance with NFPA 25, Standard for the Inspection, Testing, and Maintenance of Water Based Fire Protection Systems (National Fire Protection Association, 1998f). NFPA 25 establishes the minimum requirements for the periodic inspection, testing, and maintenance of water-based fire protection systems. The types of systems addressed by this standard include, but are not limited to, sprinkler, standpipe and hose, fixed water spray, and foam water. The standard also includes the water supplies that are part of these systems, such as fire service mains and appurtenances, fire pumps and water storage tanks, and valves that control system flow. PFS is committed to use the latest code in effect at the time to design the fire protection systems.

Water Tanks for Fire Suppression System

PFS plans to construct a water system to provide water for the fixed fire suppression systems, hose lines, and hydrants. The capacity of the primary tank meets the requirement to specify the largest fixed fire suppression system demand and hose stream allowances, per NFPA 13. PFS has calculated this demand and has specified that two, 100,000 gallon water tanks will be provided for a primary and secondary water supply. The largest fixed fire suppression system is the foam-water deluge system installed to protect the Canister Transfer Building load/unload bay area. This system should be adequate to suppress the bounding fire scenario for the

load/unload bay area, involving the heavy haul vehicle. The primary water tank capacity is also within the norms for an industrial facility. Since NFPA 801 requires an 8-hour refill time, PFS plans to provide a secondary tank of equal capacity.

NFPA 16 requires a flow not less than 0.16 gpm/ sq ft for a duration of 60 min. A density of 110 percent of the average density was assumed by the staff to satisfy the maximum change in density across the area of coverage. The high bay of the Canister Transfer Building has a cross-section of 3,250 sq ft. Each low bay has a cross-sectional area of 3,500 sq ft. The design flow time of the hose line is 90 min at a flow rate of 250 gpm. As a worst case scenario, it may be expected that the sprinklers in the high bay area will operate simultaneously with one of the low bay area. Based on the above parameters, the staff has estimated that the minimum capacity of the water tank for fire safety is approximately 94,000 gal. Therefore, a water tank with a capacity of 100,000 gal. would be adequate to fight the fire in the Canister Transfer Building. The second tank with a capacity of 100,000 gal. would be adequate to satisfy the 8-hour refill time requirement.

PFS has also stated that it will obtain water from one or more wells drilled onsite, from the reservation's existing supply, or from additional wells drilled on reservation property. The staff is satisfied with this design and concludes that PFS will have an adequate water supply for fire fighting.

The Facility design must provide accessibility to onsite and offsite emergency equipment and services such as fire departments. In this regard, standpipes and hose systems will be provided throughout the Canister Transfer Building, in accordance with NFPA 14, Standard for the Installation of Standpipes and Hose Systems (National Fire Protection Association, 1996a). In addition, portable extinguishers will be located throughout the Facility per industry standards (NFPA 10). The NRC has accepted these industry standards as adequate for facility fire safety.

Fire Fighting Brigade

The PFS Emergency Plan (Private Fuel Storage, 2000b) indicates that emergency response equipment will be located in the Security and Health Physics Building away from the Canister Transfer Building. One fire truck will be located onsite, one will be located at the Goshute village 3.5 miles away, and additional fire fighting assets will also be available from Tooele County. This dispersion of assets provides adequate accessability of fire fighting equipment and gear for use by response personnel in the event of an emergency at the Facility.

The PFS Emergency Plan indicates that a fire brigade will be available during the normal 40-hour work week and on-call after hours. A manual response from the fire brigade has been determined not to be necessary during off-hours when transfer operations are not being conducted. The brigade will receive training and equipment in accordance with industry standard NFPA 600 (National Fire Protection Association, 2000) and additional training will be provided for fire truck operations. Training to familiarize offsite responders will be offered annually. The staff considers this description of the applicant's fire protection training program to be adequate.

In the Emergency Plan, PFS has committed to have fire fighting equipment and gear stocked, inventoried, and maintained in accordance with NFPA 600. This standard requires equipment

to be maintained in accordance with manufacturers' instructions. PFS also committed to conduct inventories of emergency response equipment and supplies quarterly and after each use. The staff concludes that PFS's commitment to maintain fire fighting equipment in accordance with industry standards is acceptable and will provide adequate maintenance of its fire fighting equipment.

Summary of Review

The staff reviewed the information provided by the applicant regarding protection against potential wildfire, onsite fire, and onsite explosion at the proposed Facility. The staff found the information acceptable because:

- The restricted area with designed fire barriers has been adequately described.
- Adequate information has been presented regarding the fire design of the Facility along with the fire detection, alarm, and suppression systems to be installed. These systems will be designed following acceptable codes and standards. Moreover, these systems have sufficient capacity and capability to minimize the adverse effects of a postulated fire on structures, systems, and components important to safety.
- Noncombustible and heat-resistant materials will be used wherever practical.
- Through design of the Canister Transfer Building and administrative procedures, combustible material (e.g., spill of diesel fuel from cask transporters) will be kept out of the canister transfer cells, especially when the canisters are outside the protections of either a transfer cask or a storage cask during a canister transfer operation.
- By administrative procedures and installation of wheel stops on the rails, the locomotives will remain outside the Canister Transfer Building.
- Fire pumps designed with acceptable codes and standards will be supplied with water by primary and backup water tanks. One pump will be powered by an electrical motor and the other by a diesel engine, in case of loss of electric power.
- PFS's Emergency Plan provides for the availability of a fire brigade and fire fighting equipment and gear. The fire brigade will be organized, operated, trained, and equipped in accordance with NFPA 600. The equipment and gear will be stocked and maintained in accordance with NFPA 600.

This information is also acceptable for use in other sections of the SAR to develop the design bases of the Facility and perform additional safety analyses. Based on the previous information, there is reasonable assurance that the design requirements of 10 CFR 72.122(c) have been met. The information presented is sufficient to conclude with reasonable assurance that the Facility is adequately designed to protect structures, systems, and components important to safety from any postulated onsite fires and wildfires.

The HI-STORM 100 Cask System has been evaluated for a bounding, hypothetical fire caused by 50 gallons of spilled diesel fuel. This evaluation is described in detail in the HI-STORM 100 FSAR and has been reviewed and found to be acceptable by the staff (as documented in the NRC's HI-STORM 100 SER). PFS proposes to use the HI-STORM 100 storage casks with a reduced compressive strength from that of 3,000 psi of the concrete overpack (Private Fuel Storage Limited Liability Company, 2001a). This is a reduction from 4,200 psi identified in the HI-STORM 100 FSAR. Concrete compressive strength is controlled primarily by the water-cement ratio. The density of the concrete is inconsequentially affected by variation of the ratio of these two materials (Holtec International, 2001). The thermal conductivity of the concrete is governed by the concrete density. Therefore, use of lower strength concrete will not have any effect on the thermal performance of the overpack concrete since the material density remains essentially the same. Based on the assessment of the potential fire hazards and the fire protection measures at the Facility, there is reasonable assurance that the cask system will not be exposed to fires that exceed the design basis fire.

6.1.5.2 Explosion

The information presented in Section 2.2.1, Hazards from Facilities and Ground Transportation; Section 3.3.6, Fire and Explosion Protection; Section 4.3.12, Gas Utilities; and Section 8.2.4, Explosion, in connection with the protection against potential onsite and offsite explosions has been reviewed for conformance with the following regulatory requirements.

- 10 CFR 72.122(c) requires that structures, systems, and components important to safety be designed and located so that they can continue to perform their safety functions effectively under credible fire and explosion exposure conditions. Noncombustible and heat-resistant materials must be used wherever practical throughout the ISFSI, particularly in locations vital to the control of radioactive materials and to the maintenance of safety control functions. Explosion and fire detection, alarm, and suppression systems shall be designed and provided with sufficient capacity and capability to minimize the adverse effects of fires and explosions on structures, systems, and components important to safety. The design of the ISFSI must include provisions to protect against adverse effects that might result from either the operation or the failure of the fire suppression system.

The site is approximately 1.9 mi from the nearest highway (Skull Valley Road) or a public railroad. The only rail line close to the proposed Facility is controlled by PFS Facility management. Consequently, consideration of explosive cargo using the railway line is not necessary. There is no river nearby. Therefore, a river vessel carrying explosives need not be considered for design of the Facility. Based on Regulatory Guide 1.91 (Nuclear Regulatory Commission, 1978), the maximum probable hazardous solid cargo for a single highway truck is 50,000 lb. Regulatory Guide 1.91 recommends an air overpressure of 1 psi, below which no damage to structures, systems, and components will occur. An evaluation of potential hazards from accidental explosions at Skull Valley Road and other nearby facilities is presented in Section 15.1.2.10, Accidents at Nearby Sites, of this SER. Based on that evaluation, there is reasonable assurance that any credible explosion outside the Facility will not pose any significant hazard to the Facility.

It is possible that the outdoor tank may rupture from a collision or a tornado-driven missile impact resulting in spillage of diesel fuel. However, rupture of the storage tank and spillage of diesel fuel oil do not create a credible potential for an explosion because the diesel fuel is a Class II combustible liquid with a flash point of 126 °F.

Propane is classified as a flammable liquid. Propane will be stored as a liquified petroleum gas with the tank pressurized to the vapor pressure of the propane liquid, whose temperature will be close to the average ambient daily temperature. Based on the Fire Protection Handbook (National Fire Protection Association, 1997b), the vapor pressure of commercial propane is 132 psig at 70 °F and 216 psig at 105 °F. Relief valves on the tanks will be set at approximately 275 psig. A group of four centralized tanks, with individual capacities of maximum 5,000 gal. each will store a maximum 20,000 gal. of propane for heating the Canister Transfer Building and the Security and Health Physics Building. The storage tanks will be located outside the restricted area. The distances between the storage tanks and nearest storage casks and the Canister Transfer Building will be at least 1,800 ft. The tanks will be separated by missile walls to ensure that rupture of more than one tank at any given time is not credible. A fire barrier (minimum of 100 ft wide radially) will be placed around the tanks to protect them from wildfires.

The Canister Transfer Building is designed to withstand a pressure differential of 1.5 psi from a design basis tornado. Moreover, the design and evaluation of the HI-STORM 100 Cask System show that at least 5 psi pressure differential is needed before any damage takes place. Table 3.6-1 of the SAR specifies an air overpressure criterion of 1 psi. Any structures, systems, and components important to safety will not be subjected to more than 1 psi air overpressure.

Accidental offsite explosions may occur at the Tekoi Rocket Engine Test Facility, Dugway Proving Ground, or in the Tooele Army Depot. Additionally, a rocket engine on transit to the Tekoi Rocket Engine Test Facility through the access road or on Skull Valley Road may explode. In addition, an aircraft and/or ordnance crash in the vicinity of the site could result in overpressurization. An evaluation of these potential offsite explosion hazards has been presented in Chapter 15, Accident Analysis, of this SER. This evaluation concluded that these potential offsite explosions do not pose a credible hazard to the proposed Facility.

Summary of Review

The staff reviewed the information provided regarding onsite and offsite explosion potential. The staff found it acceptable because:

- Descriptions of potential explosion sources are adequate.
- As determined in Chapter 15 of this SER, the potential explosion sources are at sufficient distances away to produce air overpressure insufficient to cause any damage to important to safety structures, systems, and components, including the storage cask.

The information presented is also acceptable for use in other sections of the SAR to perform additional safety analysis. Based on the information presented, the staff concludes with reasonable assurance that the requirements of 10 CFR 72.122(c) for explosion protection design of the proposed Facility have been satisfied.

6.2 Evaluation Findings

Thermal evaluation of the PFS Facility, as presented in the SAR, is based on the assumption that the HI-STORM 100 Cask System will be used for storage and only casks approved under 10 CFR Part 72 will be used for transporting the spent fuel to the Facility. Findings based on this review follow.

Reliable data sources have been used to present temperatures and solar insolation at nearby sites. Data recorded during the onsite measurement program have also been presented. These short-term data correlate well with data recorded at nearby sites for a significantly longer duration. Therefore, the SAR shows that information on temperatures and solar insolation at the proposed site is acceptable and in compliance with 10 CFR 72.92(a). The temperatures and solar loads at the site are bounded by the HI-STORM 100 Cask System design parameters.

The SAR adequately describes the design of the Facility for fire detection, alarm, and suppression systems to be installed. These systems will be designed based on acceptable codes and standards. Through design and Facility administrative procedures, sources of ignition will be kept out of the canister transfer cells when a canister is outside a storage or transfer cask. Locomotives will be kept out of the Canister Transfer Building through Facility administrative procedures. Fire barriers with adequate width will be placed around the restricted area of the Facility to prevent any fire hazard from wildfires. Adequate descriptions of potential sources of accidental onsite and offsite explosions have been presented. Consequently, the SAR shows that the fire and explosion hazards at the site are acceptable and in compliance with the requirements of 10 CFR 72.122(c). Based on the assessment of the fire protection measures and the potential fire and explosion hazards at the site, there is reasonable assurance that the HI-STORM 100 Cask System will not be exposed to fires or explosions that are beyond the design basis for the cask system.

6.3 References

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7 SHIELDING EVALUATION

7.1 Conduct of Review

The shielding evaluation includes a review of the information in Chapter 7 (Private Fuel Storage Limited Liability Company, 2000) of the SAR. Chapter 7 of the SAR describes the radiation protection features of the Facility that ensure that radiation exposures to workers and to the public meet NRC regulatory criteria and are maintained ALARA. This chapter also evaluates radiation doses to the public and to workers from the operation of the Facility. Relevant information in Chapter 3, Principal Design Criteria, and Chapter 4, Facility Design, of the SAR was also considered.

The shielding evaluation of the PFS Facility is based on the use of the HI-STORM 100 Cask System, which has been approved by the NRC for use under the general license provisions of 10 CFR Part 72 (Nuclear Regulatory Commission, 2000a). Cask-specific information presented in the HI-STORM 100 FSAR (Holtec International, 2000) was also reviewed as it pertained to the shielding and radiation protection aspects of the ISFSI. Cask-specific information already documented in the staff's HI-STORM 100 SER (Nuclear Regulatory Commission, 2000b) will not be repeated in this SER.

The shielding review considered how the information in the SAR addresses the following regulatory requirements:

- 10 CFR 72.24(b) requires that the SAR describe the ISFSI structures with special attention to design and operating characteristics.
- 10 CFR 72.24(c)(3) requires that the SAR describe all structures, systems, and components important to safety.
- 10 CFR 72.24(e) requires that SAR describe the means of controlling and limiting occupational radiation exposures, within the limits given in 10 CFR Part 20, to ALARA.
- 10 CFR 72.104(a) requires that the annual dose equivalent to any real individual located beyond the controlled area be limited to 25 mrem/yr to the whole body, 75 mrem/yr to the thyroid, or 25 mrem/yr to any other organ during normal operations and anticipated occurrences.
- 10 CFR 72.106(b) requires that the dose to any individual located beyond the controlled area be no greater than 5 rem to the whole body or any organ from any design basis accident.
- 10 CFR 72.126(a)(6) requires that structures, systems, and components be designed, fabricated, located, shielded, controlled, and tested to control external and internal radiation exposures to personnel.

- 10 CFR 72.128(a) requires that spent fuel storage systems be designed to ensure adequate safety under normal and accident conditions. These systems must be designed with suitable shielding under normal and accident conditions.

- 10 CFR 20.1101 requires that doses to workers and members of the public be reduced to levels that are ALARA

- 10 CFR 20.1201 requires licensees to control occupational dose such that radiation workers at the site receive a total effective dose equivalent of less than 5 rem/yr, a sum of the deep dose equivalent and committed dose equivalent to any individual organ or tissue other than the lens of the eye of less than 50 rem/yr, a lens dose equivalent of less than 15 rem, and a shallow dose equivalent to skin or any extremity of 50 rem.

- 10 CFR 20.1301(a) requires that each licensee conduct operations so that (1) the total effective dose equivalent to individual members of the public from licensed operations does not exceed 0.1 rem in a year, and (2) the dose in any unrestricted area from external sources does not exceed 0.002 rem in any one hour.

- 10 CFR 20.1302(b) contains requirements for how a licensee may demonstrate that the dose limit in 10 CFR 72.1301 will be met.

7.1.1 Contained Radiation Sources

The source of gamma and neutron radiation is the spent fuel stored in the HI-STORM 100 Cask System. As specified in Technical Specification 2.1.1, the spent fuel that will be stored in the storage casks is limited to the approved contents specified in Section 2.0 of Appendix B to NRC Certificate of Compliance No. 72-1014 for the HI-STORM 100 Cask System. The burnup and cooling time limits specified for intact PWR fuel in the Certificate of Compliance range from 33,300 MWD/MTU for 5 years to 44,700 MWD/MTU for 15 years. The burnup and cooling time limits specified for intact BWR in the Certificate of Compliance range from 29,900 MWD/MTU for 5 years to 41,000 MWD/MTU for 15 years. A detailed assessment of the radiation sources, specific radiological source terms, and calculation methods for this fuel are provided in Chapter 5 of the HI-STORM 100 FSAR, which the staff has previously reviewed and found acceptable in the HI-STORM 100 SER. The PFS Facility SAR references the HI-STORM 100 FSAR and provides a description of the radiation sources and calculation methods used in the site-specific dose calculations for the Facility.

The applicant evaluated discharged spent fuel inventory data gathered by the Department of Energy (U.S. Department of Energy, 1996) to determine appropriate cask-average burnup and cooling times for site-specific dose estimates at the Facility. The applicant evaluated the data and calculated weighted-average burnups of 32,400 MWD/MTU for PWR spent fuel and 23,800 MWD/MTU for BWR spent fuel which will be stored at the Facility. The applicant calculated a weighted-average cooling time of 23 years for all spent fuel that is stored at the Facility, assuming that 200 casks are loaded each year at the Facility. Based on this information, the applicant used PWR fuel (in the MPC-24 canister configuration) with a cask-average burnup and cooling time of 35,000 MWD/MTU for 20 years to calculate average on-site occupational

exposure estimates from the average fuel that is expected at the Facility. The applicant used a bounding, cask-average burnup and cooling time of 40,000 MWD/MTU for 10 years to calculate bounding off-site dose estimates to members of the public.

The staff finds the description of radiation sources and calculation methods to be consistent with the information approved in the HI-STORM 100 FSAR. The use of the PWR design basis fuel in the MPC-24 canister configuration for generic dose calculations is acceptable because external dose rates from the PWR and BWR fuel in their respective canister designs are similar, as shown in the HI-STORM 100 FSAR. Based on the fuel inventory analysis presented by the applicant, the average and bounding burnup and cooling times used for subsequent calculations of average on-site occupational exposures and bounding off-site dose rates are acceptable. Actual dose rates during operation of the Facility will be measured by active and passive radiation monitoring in order to verify compliance with the radiological limits in 10 CFR Parts 20 and 72. The applicant will also operate the Facility under a Radiation Protection Program as required in Technical Specification 5.5.3 to assure that radiation fields are continually monitored and radiation doses to workers and members of the public are maintained ALARA, as actual dose information is gathered during operations. Radiation monitoring at the Facility and the Radiation Protection Program are evaluated in Sections 11.1.2 and 11.1.4 of this SER.

7.1.2 Storage and Transfer Systems

7.1.2.1 Design Criteria

The shielding design criteria for the Facility are described in Sections 3.3.5.2, 4.2.1.5.3, and 7.3 of the SAR; these sections reference the HI-STORM 100 FSAR. The HI-STORM 100 storage cask is designed to limit the average external contact dose rates (gamma and neutron) to 40 mrem/hr on the sides, 10 mrem/hr on top, and 60 mrem/hr at the air inlets and outlets, based on design basis fuel. The transfer cask is designed to reduce dose rates from a loaded canister to ALARA levels. The staff finds the use of these design criteria for the Facility to be appropriate. These design criteria provide reasonable assurance that the Facility will meet the dose limits specified in 10 CFR 72.104(a) and 10 CFR 72.106(b). Additionally, these design criteria provide reasonable assurance that the Facility will provide adequate safety based on the use of sufficient shielding in accordance with 10 CFR 72.128(a)(2).

7.1.2.2 Design Features

The storage and transfer system shielding design features are described in Section 7.3 of the SAR. Facility design features that ensure that dose rates are ALARA and within regulatory limits include:

- The only source of radiation, that can lead to significant exposure to workers or members of the public at the Facility, are the sealed canisters containing spent fuel assemblies, which will always be shielded by a shipping, storage, or transfer cask.

- The shipping, transfer, and storage casks are heavily shielded to minimize external dose rates. The storage cask consists of thick layers of steel and

concrete. The transfer cask is composed of layers of steel, lead, and neutron-shielding materials.

- The PFS Facility site layout provides substantial distance (at least 2,130 ft) between the cask storage area and the restricted area boundary, thereby minimizing radiation exposures to members of the public located beyond the restricted area boundary.

- The Canister Transfer Building is located inside the restricted area, which minimizes the route between the handling facility and storage pad and maintains substantial distance (500 m or 1,650 ft) from the controlled area boundary.

The description of the Facility storage and transfer systems satisfies the requirements of 10 CFR 72.24(b), (c)(3), and (e) because the design of the shielding components important to safety and the means for controlling and limiting occupational radiation exposures and for meeting ALARA goals are sufficiently described. The Facility shielding design features satisfy the requirements of 10 CFR 72.126(a)(6) because they include heavy shielding to minimize personnel radiation exposure. The SAR provides sufficient information to determine whether the requirements of 10 CFR 72.128(a)(2) are met; this information includes a description of the shielding materials that will provide radiation protection under normal and accident conditions.

The staff finds the description of the storage and transfer system shielding design features to be sufficient. Based on this description, the effectiveness of the shielding design features in limiting dose rates around the Facility to the values specified in 10 CFR Parts 20 and 72 can be evaluated. This evaluation is provided in Section 7.1.4.2 and Chapter 11 of this SER.

7.1.3 Shielding Composition and Details

7.1.3.1 Composition and Material Properties

The shielding composition and material properties are described in Sections 4.2.1.5.3 and 7.3.3 of the SAR. These sections reference the HI-STORM 100 FSAR. The primary shielding during storage will be from the concrete and steel in the HI-STORM 100 storage cask. The primary shielding during transfer operations will be from the steel, lead, and neutron shield in the HI-TRAC transfer cask. The detailed properties of the materials included in the shielding model are given in the HI-STORM 100 FSAR.

The staff finds that the description of the shielding composition and details are sufficient to meet the requirements of 10 CFR 72.24(b) and 10 CFR 72.24(c)(3) by describing the shielding components important to safety. The description of material composition, density, and geometry are described in sufficient detail to evaluate the effectiveness of the shielding in reducing the dose rates around the Facility to within regulatory limits.

7.1.3.2 Shielding Details

The details of the shielding are described in Section 7.3.3 of the SAR. The ISFSI will consist of approximately 4,000 storage casks containing a total of up to 40,000 MTU of spent nuclear fuel. The storage casks will be placed on a concrete storage pad in a 2 × 4 array (up to eight storage

casks per pad). The concrete pads will be arranged in a 20 x 25 array. This arrangement maximizes the amount of shielding that the outermost casks provide to the casks in the middle of the array.

The HI-STORM 100 FSAR provides a detailed description of the HI-STORM 100 storage cask and HI-TRAC transfer cask. Radially, the HI-STORM 100 storage cask provides 26-3/4 inches of concrete shielding and 2-3/4 inches of steel shielding. Axially, the storage cask provides 10-1/2 inches of concrete shielding and 5-1/4 inches of steel shielding in the storage cask lid. The storage cask vent ports are designed with sharp bends to avoid radiation streaming out of these ports. In calculating the dose rates around the Facility, the vent ports have been explicitly modeled. The HI-TRAC transfer cask is equipped with heavy neutron and gamma shielding to reduce the dose rates from a loaded canister to ALARA levels.

The description of the shielding composition and details satisfy the requirements of 10 CFR 72.126(a)(6). The radiation protection systems that will shield onsite personnel from radiation exposure have been sufficiently described.

7.1.4 Analysis of Shielding Effectiveness

7.1.4.1 Computational Methods and Data

The computational methods and data used to analyze the effectiveness of the shielding at the Facility are described in Section 7.3.3.2 of the SAR and in the HI-STORM 100 FSAR. Analyses were conducted to determine the dose rates close to the transfer cask and the dose rates both close to and far from the storage casks.

The shielding analysis of the HI-STORM casks was performed using the computer code MCNP 4A (Los Alamos National Laboratory, 1995). MCNP 4A is a three-dimensional transport code that uses Monte Carlo techniques with a combinatorial geometry modeling capability able to model the complex surfaces associated with the storage casks. Continuous energy cross-sectional data from ENDF/B-V (Los Alamos National Laboratory, 1994) were used by the computer code to determine gamma and neutron cross sections. The gamma flux-to-dose conversion factors used in the PFS Facility SAR were from ANSI/ANS-6.1.1 (American Nuclear Society Standards Committee Working Group, 1977).

The computer codes employed by the applicant and cask vendor (Holtec International) are widely used for shielding analyses and are considered acceptable by the staff for use in modeling the shielding configurations and materials at the Facility. The ANSI/ANS-6.1.1 flux-to-dose conversion factors are acceptable values for use in the shielding evaluations.

7.1.4.2 Dose Rate Estimates

The estimates of dose rates at various locations on the site and beyond the edge of the restricted area site are described in Sections 7.3.3.3, 7.3.3.4, and 7.3.3.5 of the SAR.

The HI-TRAC transfer cask is designed to reduce dose rates from a loaded canister to ALARA levels. All transfers of spent fuel canisters will occur remotely within the heavily shielded

Canister Transfer Building, which will cause offsite doses from these operations to be negligible.

The applicant calculated off-site dose rates for the HI-STORM 100 based on PWR design basis fuel source terms with a burnup and cooling time of 40,000 MWD/MTU for 10 years as discussed in Section 7.1.1 of this SER. The applicant calculated average contact surface dose rates for the HI-STORM 100 storage cask to be approximately 10 mrem/hr at the sides, 3 mrem/hr on top, and 6 mrem/hour at the vents. Based on these values, the applicant calculated a site boundary dose rate of 0.0029 mrem/hr for 4,000 casks from direct and scattered radiation exposure. As discussed in Chapter 9 of this SER, no release of radioactive material in effluent is expected during normal operations; therefore, the dose due to effluents is not considered. The applicant extrapolated the site boundary dose rate out to a distance of two miles and calculated an annual dose of 0.0356 mrem to the nearest resident, assuming the resident is continually present for 8,760 hr/yr. The applicant also calculated an annual dose of 5.85 mrem for a hypothetical person at the site boundary (e.g., non-Facility worker), assuming the person is at the site boundary for 2,000 hr/yr which is approximately equal to 40 hr/week. These dose rates are less than the 10 CFR 72.104(a) dose limit of 25 mrem/yr to the whole body to a member of the public.

No accidents were identified in the SAR that could cause significant loss of shielding or complete penetration of the storage casks. The worst accident identified was the tornado missile impact. This design basis accident could cause localized thinning of the storage cask shielding. However, only a small number of casks would be damaged and much of the shielding would remain intact. Therefore, the offsite dose due to direct and scattered radiation would not increase significantly as a result of this accident. Further, in Chapter 9 of this SER, the dose due to a hypothetical release from a single canister is estimated to be 2.68 mrem (to an individual continuously present at the owner controlled area boundary for 30 days). Thus, there is reasonable assurance that the total dose (i.e., dose due to direct and scattered radiation and to a hypothetical release) from any design basis accident will be less than the 10 CFR 72.106(b) limit of 5 rem.

Based on the results of the applicant's shielding analysis, the staff finds that the dose rates, at locations onsite and offsite, are below the limits specified in 10 CFR 20.1201, 20.1301, 20.1302, and 72.104(a). As discussed in Section 7.1.1 of this SER, the burnup and cooling parameters are acceptable for demonstrating the shielding capability of the Facility. The applicant's shielding analysis satisfies the requirements of 10 CFR 72.24, 72.126, and 72.128 by demonstrating that radiation exposures to workers and members of the public will be adequately limited through the use of shielding at the Facility. The shielding analysis also demonstrates that no credible accident will significantly increase the dose rates. Therefore, there is reasonable assurance that under accident conditions, dose rates will remain below the limits specified in 10 CFR 72.106(b).

Chapter 11 of the SER evaluates the combined radiological exposure from the direct and scattered radiation doses and any potential radioactive materials in effluents. The purpose of the evaluation is to verify the radiation protection design of the Facility and that its radiological protection program satisfies the occupational exposure and public dose requirements in 10 CFR 20.1201, 20.1301, 20.1302, 72.104(a), and 72.106(b).

7.1.5 Confirmatory Calculations

The staff performed independent calculations of the dose rates that could be expected around the storage casks and at the edge of the PFS Facility controlled area. The staff used the MCNP 4A code (Los Alamos National Laboratory, 1995), ENDF/B-VI cross section data (Los Alamos National Laboratory, 1994), and gamma flux-to-dose conversion factors from ANSI/ANS 6.1.1 (American Nuclear Society Standards Committee Working Group, 1977). These calculations confirmed the onsite dose rates calculated by the applicant and also confirmed that the offsite dose rate would be less than the 25 mrem/yr whole body dose allowable to a member of the public as required by 10 CFR 72.104. Based on its confirmatory calculations, the staff finds the applicant's shielding analysis to be acceptable.

7.2 Evaluation Findings

Evaluation of shielding at the Facility assumed that only the HI-STORM 100 Cask System will be used. Based on the staff's review of the SAR, the staff finds that the requirements of 10 CFR 72.104(a) and 72.106(b) are met with respect to dose rates due to direct and scattered radiation. The staff also finds that the other applicable requirements of 10 CFR Parts 20 and 72, as identified in Section 7.1 of this SER, have been satisfied. The staff found that: (1) the description of the contained radiation sources of the Facility is sufficient to determine that the Facility will satisfy all radiological protection criteria; (2) the design of the storage and transfer systems at the Facility is adequate to maintain exposures to ALARA and within applicable regulatory dose limits; and (3) the shielding features of the Facility are adequate to ensure that radiation exposures to workers and to the public are within the applicable regulatory limits under normal and accident conditions.

7.3 References

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8 CRITICALITY EVALUATION

8.1 Conduct of Review

The review of the criticality analysis included Chapter 3, Principal Design Criteria, and Chapter 4, Installation Design, of the SAR (Private Fuel Storage Limited Liability Company, 2000). Chapter 3 of the SAR describes the design criteria and features of the proposed ISFSI that ensure the spent nuclear fuel stored at the site will remain subcritical. Chapter 4, which describes the installation design, discusses the proposed cask system and the criticality analysis performed for the cask. The objective of the criticality review is to ensure that the stored materials remain subcritical under normal, off-normal, and accident conditions during all operations, transfers, and storage at the proposed PFS Facility. This review considered how the information in the SAR addresses the following regulatory requirements:

- 10 CFR 72.40(a)(13) requires that there is reasonable assurance the proposed activities can be conducted without endangering the health and safety of the public.

- 10 CFR 72.124(a) requires that the proposed ISFSI handling, packaging, transfer, and storage systems for the radioactive materials be designed to be maintained subcritical and that, before a nuclear criticality accident is possible, at least two unlikely, independent, and concurrent or sequential changes must occur in the conditions essential to nuclear criticality safety.

- 10 CFR 72.124(b) requires that, when practicable, the design of the ISFSI be based on favorable geometry, permanently fixed neutron poisons, or both and that the design provide for positive means to verify the continued efficacy of any neutron poisons.

- 10 CFR 72.124 (c) requires that each area, except underwater, where special nuclear material is handled, used, or stored have a criticality monitoring system. Monitoring systems of dry storage areas are not required if the special nuclear material is packaged in its stored configuration under a 10 CFR Part 72 license.

The applicant proposes to use the HI-STORM 100 Cask System, which has been reviewed and approved by the NRC under the general license provision of 10 CFR Part 72. There were no site-specific conditions identified in the PFS Facility SAR that impact the criticality safety of the cask.

8.1.1 Criticality Design Criteria and Features

This section evaluates whether the proposed criticality safety design criteria and features will maintain the stored materials in a subcritical configuration. The Facility design criteria and features are described in SAR Sections 3.3.4, Nuclear Criticality Safety and 3.6, Summary of Design. Section 4.2.1.5.4, Criticality Design, discusses the HI-STORM 100 cask design with respect to criticality safety. The applicant did not rely on the use of burnup credit, burnable neutron absorbers, or fixed neutron absorbers for the criticality safety analysis.

8.1.1.1 Criticality Design Criteria

The applicant described the criticality safety design criterion in SAR Section 3.3.4, Nuclear Criticality Safety. The casks are designed such that at least two unlikely, independent, and concurrent or sequential changes to the conditions essential to criticality safety must occur before an accidental criticality is possible. The design criterion for criticality safety is that the effective multiplication factor, k_{eff}, including statistical biases and uncertainties shall not exceed 0.95 under all credible normal, off-normal, and accident conditions and events. The proposed cask system, the HI-STORM 100, meets this design criterion.

The staff reviewed the proposed design criteria for the Facility. The staff also reviewed the cask design criteria to ensure consistency with the Facility. The staff finds acceptable the proposed design criteria because the material will be stored such that subcriticality is maintained with k_{eff} not to exceed 0.95 for all normal, off-normal and accident conditions and will, therefore, meet the requirements of 10 CFR 72.124(a). The staff also finds that the use of an NRC-certified cask will ensure that the activities at the proposed ISFSI will be performed without endangering the health and safety of the public, in accordance with 10 CFR 72.40(a)(13). The Facility conditions for criticality safety are based upon the acceptance criteria in Section 8 of NUREG-1567 (Nuclear Regulatory Commission, 2000a), such as maintaining $k_{eff} \leq 0.95$ for all conditions and no credit for burnup, burnable neutron poisons, or neutron poisons in the cask.

8.1.1.2 Features

The PFS Facility criticality safety design features are described in SAR Section 3.3.4.1, Control Methods for Prevention of Criticality. The proposed cask system, the HI-STORM 100 Cask System, maintains the spent fuel in a subcritical configuration independent of the Facility. The cask design feature relied upon to prevent criticality is the canister geometry, which establishes sufficient fuel assembly separation and is described in Chapter 6 of the HI-STORM 100 FSAR (Holtec International, 2000). All canisters will arrive at the Facility in a dry condition and will not be opened at the Facility. The canisters are transferred to the HI-STORM 100 storage cask which is designed such that there is no credible mechanism to allow water to enter the canister during storage.

The canisters also employ fixed neutron poisons and flux traps, but these are only necessary when the canisters are filled with fresh water (i.e., during fuel loading/unloading at a utility), or to meet the requirements of 10 CFR Part 71 during offsite transportation. For offsite transportation, 10 CFR 71.55(b) requires that spent fuel transportation casks are designed to be subcritical if water were to enter the canister.

Per 10 CFR 72.124(c), a criticality monitoring system is not required because the material is packaged in its stored configuration under a Part 72 license. As stated above, the canisters are not opened at the Facility.

The staff verified that the design features important to criticality safety are clearly identified and adequately described. The staff finds that the design features are based on favorable geometry and therefore meet the requirements of 10 CFR 72.124(b). The staff also finds that the stored material will be maintained in a subcritical configuration; therefore, the design

provides reasonable assurance that the activities authorized by the license can be conducted without endangering the health and safety of the public as required by 10 CFR 72.40(a)(13).

8.1.2 Stored Material Specifications

This section of the SER evaluates the description of the stored material specifications used by the applicant to ensure that the spent nuclear fuel stored on the site will be maintained in a subcritical configuration. The proposed stored material specifications are discussed in Section 3.1.1 of the SAR. The materials will consist of the PWR and BWR spent fuel assemblies approved for storage in the HI-STORM 100. The approved contents for the HI-STORM 100 Cask System are given in Appendix B of Certificate of Compliance No. 1014. The proposed Technical Specifications for the PFS Facility specify that the spent nuclear fuel to be stored in at the Facility shall meet the requirements given in Section 2.0 of Appendix B to Certificate of Compliance No. 72-1014.

The staff reviewed the proposed fuel specifications given in the SAR to ensure that they are bounded by the approved contents for the HI-STORM 100 Cask System. The staff finds that the proposed material specifications are adequate to ensure that the contents will be maintained subcritical and that, before a nuclear criticality accident is possible, at least two unlikely, independent, and concurrent or sequential changes must occur in the conditions essential to nuclear criticality safety, in compliance with 10 CFR 72.124(a).

8.1.3 Analytical Means

This section of the SER evaluates the analytical means used by the applicant to show that the spent nuclear fuel stored at the Facility will remain subcritical. Relevant information concerning the HI-STORM 100 Cask System is contained in SAR Section Chapter 4.2.1.5.4, Criticality Design.

8.1.3.1 Model Configuration

The individual cask model configuration was reviewed and approved by the staff during the certification process of the cask and is discussed in the staff's HI-STORM 100 SER. The HI-STORM 100 cask analysis assumed fresh fuel at the maximum allowed enrichments, worst case configuration, and flooding with fresh water at various densities. The analysis also considered a single cask and an array of casks. There were no site-specific conditions that impacted the criticality safety analysis of the cask; therefore, no additional modeling by the applicant was necessary for the Facility.

8.1.3.2 Material Properties

The material properties were reviewed and approved by the staff during the certification process of the HI-STORM 100 Cask System, and is discussed in the staff's HI-STORM 100 SER.

8.1.4 Applicant Criticality Analysis

This section of the SER evaluates whether the applicant addressed the most reactive conditions and whether the computer programs used were appropriate for this system. The error contingency criteria and verification analysis are given in SAR Sections 3.3.4.2 and 3.3.4.3. A synopsis of the HI-STORM 100 criticality analysis is found in SAR Section 4.2.1.5.4, Criticality Design.

8.1.4.1 Computer Program

The computer program used to perform the HI-STORM 100 criticality analysis is described in the HI-STORM 100 FSAR and in the staff's related SER. No additional criticality codes or calculations are necessary for the Facility.

8.1.4.2 Multiplication Factor

Results of the HI-STORM 100 criticality analysis show that k_{eff} of the HI-STORM 100 Cask System will not exceed 0.95 for all allowed fuel loadings under all normal, off-normal, and accident conditions. This meets the design criterion for the Facility. The calculated k_{eff} values were reviewed by the staff during the certification process of the cask and are discussed in the staff's HI-STORM 100 SER. No additional calculations were performed for the Facility.

8.1.4.3 Benchmark Comparisons

SAR Section 3.3.4.3, Verification Analysis, requires benchmark comparisons for any criticality calculations not previously approved by the NRC. No additional calculations are necessary for the Facility as there were no site-specific conditions that affect the criticality safety analysis. The benchmark comparisons used in the HI-STORM 100 analysis were reviewed during the certification process of the cask and are discussed in the staff's HI-STORM 100 SER (Nuclear Regulatory Commission, 2000b,c).

8.1.4.4 Independent Criticality Analysis

No additional criticality calculations are necessary for the Facility; thus no confirmatory calculations were performed.

8.2 Evaluation Findings

Based on a review of the SAR, the staff has determined that:

- The design, procedures, and materials to be stored for the proposed PFS Facility provide reasonable assurance that the activities authorized by the license can be conducted without endangering the health and safety of the public, in compliance with 10 CFR 72.40(a)(13).

- The design and proposed use of the PFS Facility handling, packaging, transfer, and storage systems for the radioactive materials to be stored reasonably ensure that the materials will remain subcritical and that, before a nuclear

criticality accident is possible, at least two unlikely, independent, and concurrent or sequential changes must occur in the conditions essential to nuclear criticality safety. The SAR analyses and confirmatory analysis by the NRC adequately show that acceptable margins of safety will be maintained in the nuclear criticality parameters commensurate with uncertainties in the data and methods used in calculations, and demonstrated safety for the handling, packaging, transfer and storage under normal, off-normal, and accident conditions in compliance with 10 CFR 72.124(a) and (b).

- A criticality monitoring system is not required at the PFS Facility since the special nuclear material is packaged in its stored configuration under a 10 CFR Part 72 license, in compliance with 10 CFR 72.124 (c).

8.3 References

Holtec International. 2000. *Final Safety Analysis Report for the Holtec International Storage and Transfer Operation Reinforced Module Cask System (HI-STORM 100 Cask System).* Volumes I and II. HI–2002444. Docket 72-1014. Marlton, NJ: Holtec International.

Nuclear Regulatory Commission. 2000a. *Standard Review Plan for Spent Fuel Dry Storage Facilities.* NUREG–1567. Washington, DC: Nuclear Regulatory Commission.

Nuclear Regulatory Commission. 2000b. 10 CFR Part 72 *Certificate of Compliance No. 1014, Amendment 0, for the HI-STORM 100 Cask System.* Docket No. 72-1014. May 31.

Nuclear Regulatory Commission. 2000c. *Holtec International HI-STORM 100 Cask System Safety Evaluation Report.* Docket No. 72-1014. May.

Private Fuel Storage Limited Liability Company. 2000. *Safety Analysis Report for Private Fuel Storage Facility.* Revision 18. Docket No. 72–22. La Crosse, WI: Private Fuel Storage Limited Liability Company.

9 CONFINEMENT EVALUATION

9.1 Conduct of Review

The staff reviewed the confinement evaluation presented in the SAR (Private Fuel Storage Limited Liability Company, 2000). The staff reviewed Sections 3.3.2.1, 3.4, 4.2.1:2.2, 4.2.1.5.5, and 8.2.7 of the SAR. The staff also reviewed the applicable sections of the proposed Technical Specifications.

The Facility will use the HI-STORM 100 Cask System, which has been approved by NRC for use under the general license provisions of 10 CFR Part 72. The design specifications for the confinement function of the HI-STORM 100 cask are addressed in Chapter 7 of the HI-STORM 100 Cask System FSAR (Holtec International, 2000).

Based on the statements in the HI-STORM 100 FSAR, the applicant conducted an analysis of a hypothetical radiological release. The confinement evaluation submitted by the applicant relies on the analyses performed by Holtec International to demonstrate compliance with 10 CFR Part 72, and includes a discussion of radiological release calculations and an evaluation of stored material degradation. The applicant provided no information in the PFS Facility SAR on chemical composition and mechanical properties of materials for construction of critical cask components; this information is provided in the referenced HI-STORM 100 Cask System FSAR.

This review was conducted in accordance with the guidance presented in Chapter 9 of NUREG–1567, "Standard Review Plan for Spent Fuel Dry Storage Facilities."

9.1.1 Radionuclide Confinement Analysis

The application was reviewed to identify the quantity of radionuclides that hypothetically could be released during normal, off-normal, and accident conditions, including design basis accidents. The information presented has been reviewed for conformance with the following regulatory requirements:

- 10 CFR 72.24(l) requires that the licensee provide a description of the equipment to be installed to maintain control over radioactive materials in gaseous and liquid effluents, including the means to maintain radioactive materials in effluents as low as is reasonably achievable.
- 10 CFR72.44(c) requires that the licensee define Technical Specifications for design features that would have a significant effect on safety if altered or modified.
- 10 CFR 72.122(a) requires that structures, systems, and components important to safety be designed, fabricated, erected, and tested to quality standards commensurate with the importance to safety of the function to be performed.
- 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.

- 10 CFR 72.126(d) requires that the ISFSI be designed to limit to ALARA levels the release of radioactive materials in effluents during normal operations; and control the release of radioactive materials under accident conditions.

- 10 CFR 72.128(a)(3) requires that spent fuel storage systems be designed with confinement structures and systems.

The HI-STORM 100 Cask System is designed for long-term confinement and dry storage of PWR and BWR spent nuclear fuel. The design of the HI-STORM 100 Cask System is discussed in detail in Section 4.2.1.4 of the SAR. The major components of the HI-STORM 100 Cask System that are classified as important to safety include the sealed MPC and the storage cask. The MPC is designed to maintain a confinement barrier under all normal, off-normal, and accident conditions.

The confinement boundary for the Holtec HI-STORM 100 includes the MPC shell, the bottom baseplate, the MPC lid (including the vent and drain port cover plates), the MPC closure ring, and the associated welds. The welds forming the confinement boundary are described in detail in Section 7.1.3 of the HI-STORM 100 FSAR. The MPC is designed, fabricated, and tested in accordance with the applicable requirements of ASME code, Section III, Subsection NB, to the maximum extent practicable. The MPC lid and MPC closure ring seal welds are designed to maintain confinement under normal and design basis accident conditions.

In Section 6.5 of the SAR, the applicant states that no releases of any type of radioactive material will occur during normal operations. This statement is in agreement with the statements in Section 7.2.3 of the HI-STORM 100 FSAR. Thus, leakage of the MPCs under normal conditions was not considered.

In Section 8.2.7 of the SAR, leakage from the HI-STORM cask under hypothetical accident conditions was evaluated. Following the methodology in the HI-STORM 100 FSAR, and in accordance with Interim Staff Guidance Number 5, the applicant calculated the dose to an individual continuously present for 30 days at the location nearest to the Canister Transfer Building on the owner controlled area boundary. This hypothetical, worst-case calculation yielded a total effective dose equivalent of 2.68 mrem from a single leaking MPC. The accident dose rates (dose due to direct and scattered radiation and to a hypothetical release) for the HI-STORM 100 Cask System do not exceed limits specified in 10 CFR 72.106(b).

While a hypothetical accident condition leakage calculation was performed for the HI-STORM 100 cask, the applicant expects that there will be no release of radioactive materials in effluents during normal and all credible accident conditions. This is supported by the applicant's analyses which demonstrate that the MPC would maintain its confinement integrity under the design basis normal, off-normal, and accident conditions (including earthquake, tornado, flood, explosions, fire, and cask tipover). Based on the results of the applicant's analyses, the staff agrees that the MPC confinement integrity would be maintained under the design basis normal, off-normal, and accident conditions. The staff further investigated the acceptability of the applicant's conclusion that the HI-STORM 100 cask would not leak under normal, off-normal, and accident conditions by performing a risk assessment of the MPC confinement system welds. The probabilities of a leak through the various welds in the MPC are summarized in Table 9-1.

Table 9-1: Number of Through-Wall Flaws per Weld for each weld used to construct and seal the HI-STORM MPC

Weld	Probability of a Through-Wall Flaw
MPC Lid to Shell Weld	5.8×10^{-9} (P1)
Closure Ring to Shell	3.6×10^{-4} (P2)
Closure Ring to Lid Weld	2.8×10^{-4} (P3)
Vent and Drain Cover Plate Weld	1.2×10^{-6} (P4)
Circumferential and Axial Seam Weld	7.1×10^{-6} (P5)
Shell to Baseplate Weld	2.6×10^{-6} (P6)

Applying the probabilities outlined in Table 9-1, the probability of forming a leak in the cask lid seal welds is:

$$(P1 \times P2) + (P1 \times P3) + [(P4 \times P2) + (P4 \times P3)] \times 2 =$$

$$[P1 + (2 \times P4)] \times (P2 + P3) = 1.5 \times 10^{-9}$$

The total probability of a leak in the MPC is calculated as the probability from the shell to base plate weld (P6), plus the probability of the seam weld (P5), plus the probability of the cask lid welds. The total probability is:

$$P6 + P6 + [(P1 + P4) \times (P2 + P3)] = 9.7 \times 10^{-6}$$

The probabilities above do not credit helium leak testing performed by the MPC manufacturer. Consequently, the probabilities are expected to be lower.

Using the assumption that the cask leaks at a flow rate and with an effluent source term analyzed in the HI-STORM 100 FSAR (maximum leak rate permitted by the HI-STORM 100 Technical Specifications and validated not to be exceeded by the Technical Specification leak tests), the radiological consequence at 100 meters from the cask, assuming plateout and settling of radionuclides, is 8.08×10^{-5} mrem/year. The risk from one cask is the product of the probability of a leak times the consequences (mrem/year). The risk from one HI-STORM leaking cask is 8×10^{-10} mrem/year.

Assuming a population of 4000 casks, Table 9-2 identifies the probability of one or more (k) casks leaking and the associated risk, in mrem/year.

Table 9-2: Risk at PFS from Leaking Cask(s) at 100 meters

Number of Casks (k) Leaking From a Population of 4000 Casks	Probability of k Casks Leaking at the Same Time	Risk from k Casks Leaking (mrem/year)
1	4×10^{-2}	3×10^{-6}
2	8×10^{-4}	1×10^{-7}
3	1×10^{-5}	2×10^{-9}
4	9×10^{-8}	3×10^{-11}
5	7×10^{-10}	3×10^{-13}

The staff's analysis shows that more than 3 out of 4000 casks leaking at PFS is not a credible event (i.e., probability less than 1×10^{-6}). At 100 meters from the cask, the radiological consequence of radioactive effluents leaking from three casks is negligible (i.e., 2×10^{-4} mrem/year, which is orders of magnitude below normal background radiation).

The staff, therefore, concludes that:

- The risk to the public from radioactive effluent released at the Facility with 4000 casks is 3×10^{-6} mrem/year (calculated by summing the risks in the third column of Table 9-2).
- Should three in 4000 casks leak, the consequence at 100 meters is insignificant (2×10^{-4} mrem/year).
- Reasonable assurance exists that the risk from radioactive effluents released to the general public from storing 4000 HI-STORM casks at the Facility is insignificant.
- Stainless steel welded casks (with redundant welds in the lid enclosure of the cask) manufactured and inspected according to ASME code, as approved by the staff, are not expected to release radioactive effluents.

In addition to the investigation described above, the staff reviewed the applicable chapters of the SAR, and found that the conclusions that were made by the applicant were in agreement with the Holtec HI-STORM 100 FSAR, and are acceptable. The staff also reviewed the site Technical Specifications proposed by the applicant, and found those portions related to the confinement integrity of the HI-STORM cask to be acceptable.

9.1.2 Confinement Monitoring

The staff's review of this section focused on two areas. These areas included the continuous monitoring of closure seal effectiveness and the measure of radionuclides released to the environment during normal and accident conditions. These areas are discussed in Sections

4.3.1, 4.3.5, 4.3.8.4, and 4.3.11 of the SAR. The information presented has been reviewed for conformance with the following regulatory requirements:

- 10 CFR 72.24(l) requires that the licensee provide a description of the equipment to be installed to maintain control over radioactive materials in gaseous and liquid effluents, including the means to maintain radioactive materials in effluents as low as is reasonably achievable.

- 10 CFR 72.44(c) requires that the licensee define Technical Specifications for surveillance requirements to ensure that the conditions for safe storage be met.

- 10 CFR 72.122(h)(4) requires that storage confinement systems have the capability for continuous monitoring in a manner such that the licensee will be able to determine when corrective action needs to be taken to maintain safe storage conditions.

- 10 CFR 72.126(c)(1) requires that means for measuring the amount of radionuclides in effluents during normal operations and under accident conditions be provided for storage and handling systems.

- 10 CFR 72.128(a)(3) requires that spent fuel storage systems be designed with a capability to test and monitor components important to safety.

The final seal welds and leak testing of the MPC will be performed at the originating nuclear power plant. Welder qualifications, welding procedures, nondestructive examination, and leak testing of the welds will be in accordance with NRC-accepted industry standards (i.e., the ASME Boiler and Pressure Vessel Code and ANSI-N14.5).

The MPCs are loaded into the shipping casks at the originating nuclear power plant and are not opened at the Facility. Each MPC will arrive on the PFS Facility site in a shipping cask, and is then transferred to the HI-STORM 100 storage cask in the canister transfer building, using the HI-TRAC transfer cask.

Based on the staff's assessment of welded cask enclosures, as stated in NUREG–1536, "Standard Review Plan for Dry Cask Storage Systems," Chapter 7, Section V.2, the MPC, which is the confinement system for the HI-STORM 100 Cask System, provides reasonable assurance that no effluents will be released and, therefore, requires no monitoring of the MPC for leakage. The seal weld will be inspected and tested in accordance with the requirements in Section 8.1.5 of the HI-STORM 100 FSAR. These requirements were reviewed during the certification of the HI-STORM 100 storage cask, and were found to be acceptable by the staff.

The staff finds the applicant's proposal to provide no monitoring of the confinement barrier for the HI-STORM 100 casks acceptable, because the casks will be loaded, welded, inspected, and tested in accordance with appropriate procedures.

9.1.3 Protection of Stored Materials from Degradation

Review of this section of the SAR was performed to establish that the fuel cladding would not experience significant degradation during the licensed storage period of 20 years. The information presented has been reviewed for conformance with the following regulatory requirements:

- 10 CFR 72.24(g) requires that the license application include an identification and justification for the selection of those subjects that will be probable license conditions and Technical Specifications.

- 10 CFR 72.122(h)(1) requires that the spent fuel cladding be protected during storage against degradation that leads to gross ruptures or be otherwise confined such that degradation of the fuel during storage does not pose operational safety problems with respect to its removal from storage.

Following the loading of the MPC, the main lid is welded and a helium leak test is performed on the seal weld. The MPC cavity is then vacuum dried and filled with helium fill gas. The vent and drain ports are then welded into place and a helium leak test is conducted on the vent and drain port covers. These steps are described in detail in the HI-STORM 100 FSAR. The helium back-fill procedure ensures that the presence of oxidizing gasses in the MPC cavity will be minimized.

The thermal analysis of the HI-STORM 100 cask indicates that the fuel cladding temperature will not exceed the limits established to prevent fuel clad degradation during storage.

The staff verified that the applicant's SAR was consistent with the information provided in the HI-STORM 100 FSAR. The staff reviewed the proposed Technical Specifications and found the portions related to the protection of stored materials from degradation in the HI-STORM cask to be acceptable.

9.2 Evaluation Findings

Evaluation of confinement of spent nuclear fuel stored at the Facility assumed that only the HI-Storm 100 cask will be used. Based upon the staff's review of the applicant's submittal and the applicable Technical Specifications the staff has made the following findings.

- The radionuclide confinement analysis for the Holtec HI-STORM 100 cask and the PFS site has met the requirements of 10 CFR 72.24(l) by providing a description of how radioactive materials in gaseous and liquid effluents will be controlled such that they are ALARA. The requirements of 10 CFR 72.44(c) have been met based on the staff's review of the Technical Specifications that have been submitted by the applicant. Because the MPC lid is welded and tested in accordance with ASME code and is not expected to leak under normal, off-normal, and accident conditions, the staff finds that the requirements of 10 CFR 72.122(h)(3), 10 CFR 72.126(d),10 CFR 72.128(a)(3), and 10 CFR 72.122(a) have been met.

- The staff concludes that the HI-STORM 100 cask, which contains the MPC which has been welded and tested in accordance with ASME code, is not expected to leak and therefore does not require confinement monitoring. Based on this finding, the requirements of 10 CFR72.44(c), 10 CFR 72.122(h)(4), 10 CFR 72.126(c)(1), and 10 CFR 72.128(a)(3) are met.

- The staff concludes that the proposed Technical Specifications are sufficient to protect the stored materials from degradation in accordance with 10 CFR 72.24(g). The staff also finds that the proposed methods to protect the stored materials from degradation are acceptable to protect the spent fuel cladding from gross ruptures in accordance with 10 CFR 72.122(h)(1).

9.3 References

American Society of Mechanical Engineers. ASME Boiler and Pressure Vessel Code, Section III. NY: American Society of Mechanical Engineers. 1998.

Holtec International. *Final Safety Analysis Report for the Holtec International Storage and Transfer Operation Reinforced Module Cask System (HI-STORM 100 Cask System). Volumes I and II.* HI–2002444. Docket 72-1014. Marlton, NJ: Holtec International. 2000

Nuclear Regulatory Commission. Atmospheric Dispersion Models for Potential Accident Consequence Assessments at Nuclear Power Plants. Regulatory Guide 1.145. Revision 1. Washington, DC: Nuclear Regulatory Commission. 1983.

Nuclear Regulatory Commission. *Standard Review Plan for Spent Fuel Dry Storage Facilities.* NUREG–1567. Washington, DC: Nuclear Regulatory Commission. 1998.

Private Fuel Storage Limited Liability Company. *Safety Analysis Report for Private Fuel Storage Facility.* Revision 18. Docket No. 72-22. La Crosse, WI: Private Fuel Storage Limited Liability Company. 2000.

Thadani, A.C., Office of Nuclear Regulatory Research, U.S. Nuclear Regulatory Commission, memorandum to Kane, W.F., Office of Nuclear Material Safety and Safeguards, U.S. Nuclear Regulatory Commission. August 22, 2000.

10 CONDUCT OF OPERATIONS EVALUATION

10.1 Conduct of Review

Chapter 9, Conduct of Operations, of the SAR, describes the organizational structure that will manage and operate the Facility, including the associated plans and procedures for preoperational testing and operations, training, normal operations, emergency planning, and decommissioning. The chapter includes descriptions of the responsibilities of key personnel, training program, standards and procedures that govern daily operations, and records generated as a result of those operations. The controls used to promote safety and ensure compliance with the license and the regulations applicable to the Facility are also included. The purpose of the review is to ensure that the infrastructure to manage, test, and operate the Facility, including provisions for effective training, is acceptable.

10.1.1 Organizational Structure

Section 9.1, Organizational Structure, of the SAR, describes the organizational structure to be used to manage and operate the Facility. The review considered how the information in the SAR addresses the following regulatory requirement:

- 10 CFR 72.40(a)(4) requires that the applicant be qualified to conduct the operation covered by 10 CFR Part 72.

10.1.1.1 Corporate Organization

Section 9.1, Organizational Structure, of the SAR describes the corporate organization that will be used to manage and operate the Facility.

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

The Board of Managers is responsible for:

- supervising the General Manager/Chief Operating Officer,
- long-range planning,
- preparation of the license application,
- ensuring establishment and effective implementation of the QA program, and
- ensuring compliance with the conditions of the license.

The Facility Safety Review Committee is responsible for reviewing and advising the Board of Managers on all matters relating to structures, systems, and components important to safety. Committee responsibilities include, but are not limited to, the review of:

- safety evaluations for procedures and changes thereto,

- changes to structures, systems, and components classified as important to safety,
- tests or experiments involving structures, systems, and components classified as important to safety,
- review of QA audits related to safety,
- proposed changes to the technical specifications of the license, and
- violations of codes, regulations, orders, license requirements, or internal procedures/instructions which pertain to structures, systems, and components classified as important to safety.

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

- Chairman - Facility General Manager/Chief Operation Officer,
- Quality Assurance,
- Radiation Protection,
- Nuclear Engineering, and
- Maintenance/Operations.

During construction, PFS will have a team of three persons available for oversight of Facility design, procurement, and construction. This staff will be led by the PFS Facility Project Manager and will include a construction engineer and a procurement specialist. They will ensure oversight of the Architect/Engineer, contractors, and vendors and will be assisted as needed (at the discretion of the PFS Facility Project Manager) by utility staff from the member utilities in a full range of specialties appropriate to the design, construction, startup, and operation of an ISFSI. These three persons will be available for initial training of the site staff prior to Facility operation.

At the completion of Facility design, construction, licensing, and testing, the responsibility for daily operation of the Facility will be turned over to the General Manager who reports to the Chairman of the Board. The General Manager will be responsible for the receipt, handling, storage, and consolidation of the spent fuel. The General Manager will also be responsible for the safe maintenance and operation of the Facility; reconfiguration of spent fuel storage areas, if required; and refurbishing degrading facilities to ensure safety and environmental compliance.

The staff review finds the corporate organizational structure acceptable because it defines the relationships between corporate organizations and delineates authority and responsibility. Responsibility is clear to specific individuals and parts of the organization and the functions of radiation protection and other safety agencies are provided organizationally separate lines of reporting from Facility operations. The staff has also determined that a Safety Review Committee will be formed and will be properly organized and staffed and therefore is acceptable.

10.1.1.2 Onsite Organization

Sections 9.1.2.1, Onsite Organization, and 9.1.2.2, Personnel Functions, Responsibilities, and Authorities, in the SAR present the onsite organization, including responsibilities and reporting relationships.

During the operational phase, the Board of Managers has overall responsibility for safe operation of the Facility and the authority to ensure continued safe operation. The General Manager will also function as the Chief Operating Officer during the operational phase and ensure safe and efficient operations and maintenance activities at the Facility. The functions represented by the PFS organization have the authority to control various aspects of the Facility including engineering and design, QA, fuel accountability, maintenance, radiation protection, training, operations, and decommissioning.

As discussed in Section 9.1.4, Liaison with Outside Organizations, of the SAR, the oversight of the outside organizations which manufacture canisters is provided by the General Manager/Chief Operating Officer and the Nuclear Engineering staff, who will conduct oversight activities in accordance with the QA program. Fabrication of canisters to appropriate standards and storage, transfer, and transportation technology are monitored by the nuclear engineering staff. The oversight of outside organizations is audited periodically by the QA staff.

The staff review finds the on-site organizational structure acceptable because it defines relationships between on-site organizations and liaisons with outside organizations, and delineates authority and responsibility. The position responsible for oversight of outside organizations that manufacture canisters is clearly defined.

10.1.1.3 Management and Administrative Controls

Section 9.4.1, Procedures, of the SAR commits to preparing and using administrative, radiation protection, maintenance and surveillance, QA, and training procedures that will be employed at the Facility. Use of these procedures encompasses preoperational testing as well as normal operations. These procedures and subsequent changes thereto will be reviewed and approved by the Health Physics and QA organizations, independent of the operating organizations. The applicant has committed that procedures will contain sufficient detail to allow qualified and trained personnel to perform the actions without incident or abnormal event.

Section 9.4.2, Records, of the SAR describes the procedures and requirements for maintaining records at the Facility. These procedures will be developed specifically for the Facility. The scope of the record keeping procedures includes records retention period; QA requirements; operating records that document principal maintenance, alterations, and additions to facilities; records of off-normal occurrences and events associated with radioactive releases; records for decommissioning; and environmental surveys. The record keeping function falls under the responsibility of the Administrative Assistant. Unless otherwise noted, records will be maintained until termination of the Facility license by the NRC.

The record keeping system discussed in Section 9.4.2, Records, of the SAR includes documentation of the receipt, inventory, location, and transfer of spent fuel. The time period for keeping the various records will be specified and duplicate records will be retained in both the Administration Building and the Security and Health Physics Building that will ensure both sets of records could not be destroyed by a single event.

The staff found that the management and administrative controls committed to in the SAR are adequate and, if fully implemented, provides reasonable assurance that the operations at the site will be properly controlled and documented. The applicant has described an organizational

system for the preparation and control of procedures, including changes to procedures, and for generating and maintaining adequate records. The staff finds this organizational system acceptable based on the descriptions and commitments given in the SAR.

10.1.2 Pre-Operational Testing and Startup Operations

10.1.2.1 Pre-Operational Testing Plan

Section 9.2, Pre-Operational Testing and Operation, of the SAR includes Subsections 9.2.1, Administrative Procedures for Conducting Test Programs; 9.2.2, Pre-Operational Test Plan; and 9.2.3, Operational Readiness Review Plan. The review considered how the information in the SAR addresses the following regulatory requirement:

- 10 CFR 72.40(a)(4) requires that the applicant be qualified to conduct the operation covered by 10 CFR Part 72.

Prior to receipt and storage of fuel at the Facility, a series of preoperational, startup, and performance tests will be developed and implemented. The scope of these tests will include construction testing, physical facilities testing, operational testing, and associated auxiliary equipment. The objective of the preoperational and startup testing program is to verify that the storage system components can operate safely and effectively.

Section 9.2.1, Administrative Procedures for Conducting Test Operations, of the SAR states that appropriate test procedures will be developed to support the preoperational testing and startup programs. These test procedures will be prepared, reviewed, modified, and controlled by a responsible line manager and the Operations Review Committee.

Section 9.2.2, Pre-Operational Test Plan, of the SAR provides a description of the test program and commits that the tests will simulate, as nearly as possible, the actual operations at the Facility. Testing will be performed for (i) construction, (ii) physical facilities, and (iii) operational procedures.

Construction testing will be performed on:

- cask storage pad construction,
- Canister Transfer Building construction, and
- Facility yard and yard infrastructure construction.

Physical facilities testing will be performed on:

- storage system transfer casks,
- canister downloader equipment,
- lifting yokes,
- Canister Transfer Building overhead 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, and
- fire truck and fire protection equipment.

Operational testing will include:

- removing the personnel barrier, impact limiters, and shipping cask from the heavy haul trailer or rail car using the canister transfer overhead bridge crane;

- up-righting 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-supplied 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-supplied canister lifting slings and equipment;

- ensuring that all steps throughout the transfer process are performed in an ALARA manner to minimize radiation doses;

- 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; and

- transferring the canister from the storage cask back to the shipping cask using the overhead crane as required when shipping fuel offsite.

Section 9.2.3, Operational Readiness Review Plan, of the SAR commits to an Operational Readiness Review to be performed by the Facility staff in order to verify the readiness of the Facility and personnel to begin full operations.

The Operational Readiness Review team 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 Operational Readiness Review team is expected to conduct internal meetings with the applicable organizations to ensure that all activities reviewed in the Operational Readiness Review are accomplished prior to operation. The Operational Readiness Review team will prepare and issue a report addressing the scope of the Operational Readiness Review and all conclusions, findings, and observations of each review item. The report will be signed off by the Operational Readiness Review Team Leader, Facility General Manager, and other appropriate managers.

The staff review found that the preoperational test plan includes the necessary tests and provides for proper evaluation, approval, and use of the test results. Appropriate administrative procedures will be developed to support the preoperational testing and startup programs, and a Facility staff review of operational readiness will be performed prior to operation.

10.1.2.2 Startup Plan

The SAR did not include a startup plan. Therefore, a license condition requires PFS to submit a startup plan to the NRC prior to receipt and storage of fuel at the Facility.

NUREG-1567 provides guidance on the elements that should be included in a startup plan. The operating startup plan should identify those specific operations involving the initial handling of radioactive material to be placed into storage. Although plant procedures to be used for normal operations or during steady-state conditions would not necessarily be included in the operating startup plan, the evaluation of the effectiveness of those procedures should be elements of the operating startup plan. For ALARA considerations, as many of the operating startup actions as feasible should be performed during preoperational testing (i.e., before sources of exposure are present).

The operating startup plan should include the following elements:

- tests and confirmation of procedures and exposure times involving actual radioactive sources (e.g., radiation monitoring, in-pool operations);
- direct radiation monitoring of casks and shielding for radiation dose rates, streaming, and surface "hot-spots";
- verification of effectiveness of heat removal features; and
- documentation of results of tests and evaluations.

10.1.3 Normal Operations

Section 9.4, Normal Operations, of the SAR includes Subsections 9.4.1, Procedures, and 9.4.2, Records. The review considered how the information in the SAR addresses the following regulatory requirement:

- 10 CFR 72.40(a)(4) requires that the applicant be qualified to conduct the operation covered by 10 CFR Part 72.

10.1.3.1 Procedures

Section 9.4.1, Procedures, of the SAR commits to preparing and using administrative, radiation protection, maintenance, surveillance, QA, and training procedures that will be employed at the Facility. Use of these procedures encompasses preoperational testing as well as normal operations. These procedures and changes thereto will be reviewed and approved by the Health Physics and QA organization, independent of the operating organization. The SAR states that procedures will contain sufficient detail to allow qualified and trained personnel to perform the actions without incident or abnormal event.

The staff review found that the control of procedures, including procedure changes, described in the SAR was adequate. Preparation of procedures and procedure changes will have the appropriate level of detail and safety review.

10.1.3.2 Records

Section 9.4.2, Records, of the SAR describes the procedures and requirements for maintaining records at the Facility. The procedures will be developed specifically for the Facility. The scope of the record keeping procedures includes record retention period; QA requirements; operating records that document principal maintenance, alterations, and additions to facilities; records of off-normal occurrences and events associated with radioactive releases; records for decommissioning; and environmental surveys. The record keeping function falls under the responsibility of the Administrative Assistant. Unless otherwise noted, records will be maintained until termination of the Facility license by the NRC.

The staff review found that the record keeping procedures committed to in the SAR are adequate to assure that records will be properly developed and maintained.

10.1.4 Personnel Selection, Training, and Certification

Section 9.1.2.2, Personnel Functions, Responsibilities, and Authorities, of the SAR defines the Management and Operating contractor positions that specify minimum qualifications and training for the operation of the Facility. Section 9.1.3, Personnel Qualification Requirements, of the SAR contains Subsections 9.1.3.1, Minimum Qualification Requirements, and 9.1.3.2, Qualifications of Personnel. Section 9.3, Training Program, of the SAR contains Subsections 9.3.1, Program Description; 9.3.2, Retraining Program; and 9.3.3, Administration and Records. The review considered how the SAR addresses the following regulatory requirements:

- 10 CFR 72.40(a)(4) requires that the applicant be qualified to conduct the operation covered by 10 CFR Part 72.

- 10 CFR 72.40(a)(9) requires that the personnel training program comply with Subpart I of 10 CFR Part 72. Subpart I, Training and Certification of Personnel, consists of 10 CFR 72.190, 72.192 and 72.194, summarized below.

- 10 CFR 72.190 requires that operators of equipment and controls that are important to safety must be trained and certified, or be under the direct visual supervision of such an individual. Supervisory personnel who direct such operations must also be certified.

- 10 CFR 72.192 requires that the applicant establish a program for training, proficiency testing, and certification of personnel, and that the program be submitted to the Commission for approval.

- 10 CFR 72.194 requires that the physical condition and general health of personnel certified for the operation of equipment and controls that are important to safety must not adversely affect safe operation of the Facility. For example, a condition that might cause impaired judgment or motor coordination must be considered in the selection of personnel.

10.1.4.1 Personnel Organization

Section 9.3, Training Program, of the SAR states that PFS commits to providing training using a systematic approach to training to support the Emergency Plan, Physical Security Plan, QA plan, and administrative and safety requirements. Section 9.3.4, Administration and Records, of the SAR assigns responsibility for the training program to the Emergency Preparedness Coordinator. This responsibility includes implementing the training program and maintaining up-to-date training records for trained personnel, new employees, and refresher or upgrading training. Records to be maintained in accordance with the record keeping program described in Section 9.4.2, Records, of the SAR will include written examinations, records of practical examinations that include delineation of operator strengths, weaknesses, and recommendations for additional training or retesting; training topics and hours for each operator; and job performance.

The staff review found that the personnel organization and systematic approach to training are acceptable. The personnel organization identifies the position that has responsibility for the training program, including implementing the program and maintaining training records.

10.1.4.2 Selection and Training of Operating Personnel

Section 9.1.3.1, Minimum Qualification Requirements, of the SAR defines the qualifications required for specific job assignments. Specific requirements are identified for the General Manager, the Radiation Protection Manager, Radiation Protection Technicians, Lead Mechanic/Operator, Mechanics, Lead Instrument and Electrical Technician, Lead QA Technician, QA Technician and Auditor, Lead Nuclear Engineer, Nuclear Engineers, Security Captain, Emergency Preparedness Coordinator, and the engineer positions on the Safety Review Committee. The qualifications listed in the SAR for these positions are consistent with those of similar positions for other nuclear facilities. Operation of equipment and controls is limited to trained and certified personnel, or is performed under their direct visual supervision.

In Section 9.1.3.2, Qualifications of Personnel, of the SAR, PFS commits to maintaining personnel having specific training requirements so that compliance with the minimum requirements can be demonstrated.

In Section 9.3, Training Program, of the SAR, PFS commits to use of the systematic approach for training personnel for Facility operations including the Emergency Plan, Physical Security Plan, QA plan, and administrative and safety requirements.

General Employee Training will be provided for all Facility operators and supervisory personnel. Topics will include applicable regulations and standards, the engineering principles of radiological shielding, basic health physics, fuel handling, the structural characteristics of the Facility, administrative procedures, and the Emergency Plan and procedures.

Detailed operator training will be provided for those individuals requiring it. The training will include:

- canister transfer system design and operations,
- canister transfer system normal and off-normal procedures,
- storage Facility normal and off-normal procedures,
- on-site transportation normal and off-normal procedures,
- maintenance,
- storage cask temperature monitoring system,
- radiation detection, monitoring, sampling, and survey instruments,
- layout and functions of the Facility,
- operator responsibility and authority,
- technical specifications,
- normal and emergency communications,
- on-site transportation, and
- topics covered in General Employee Training, addressed with specific emphasis on operations.

Section 9.3.3, Continuing Training, of the SAR commits to preparing procedures to implement retraining, proficiency testing, and requalification for ISFSI personnel, as required.

The staff review found that PFS's program for selection and training of operating personnel will provide an adequately trained operations and supervisory staff, acceptable documentation, and records of the training. The staff has reviewed the personnel qualification requirements and training program commitments described by the applicant in the SAR. On the basis of this review, the staff has determined that the described personnel training and certification program will comply with 10 CFR Part 72, Subpart I. The basis for this determination is as follows.

Pursuant to 10 CFR Part 72, Subpart I, a plan and program for training and certification must be defined in a license application at a level of detail that provides reasonable assurance that Facility personnel will be trained and qualified to perform spent fuel storage activities without undue risk to the health and safety of workers and the public. NUREG–1567 (Nuclear Regulatory Commission, 1998) provides guidance to the staff for the acceptable level of detail of descriptions of the training program, its administration, commitments for its implementation, and the principles to be applied in the development of the training and certification program. For example, NUREG–1567, Section 10.4.4.2, states that the type and level of training to be provided for each job description, including specific training provided to specific job description, must be listed. Alternately, the basis used to identify the type and level of training may be described. The applicant committed to conduct training using a systematic approach to training. The staff considers the five elements of a systematic approach to training (or

equivalent), as defined in 10 CFR 55.4 to be an acceptable method for training program implementation at an ISFSI. The proposed training plan commits to using the five elements, as defined in 10 CFR 55.4.

The staff reviewed the personnel qualification requirements specified in Section 9.1.3 of the SAR and compared those qualifications to the requirements of Regulatory Guide 1.8 (Nuclear Regulatory Commission, 1987) and associated American National Standards Institute/American National Society (ANSI/ANS) standards. Regulatory Guide 1.8 and the ANSI/ANS standards referenced in the regulatory guide address the qualification and training of personnel for nuclear power plants. For various positions, the Regulatory Guide and referenced ANSI/ANS standards specify particular qualifications, such as education, training, examination and experience. The regulatory guide and ANSI/ANS standards are applicable to the operating organization at a commercial nuclear power reactor. Because the PFS Facility is a passive Facility with significantly less complex operations than a commercial nuclear power reactor, there is a significant reduction in the size of the management staff proposed for the Facility as compared to a reactor facility. The staff has determined that the Facility operating organization and designation of responsibilities is acceptable, given the passive nature and operating requirements of an ISFSI.

The staff has determined that the SAR provides an acceptable level of detail with respect to operator experience, instruction and training courses, examination and testing requirements, and the criteria for qualifications or revocations. Qualifications for operators must include applicable training and experience, which may be at facilities other than dry storage facilities. The minimum personnel qualification requirements are comparable to similar positions at power reactor facilities described in Regulatory Guide 1.8 (Nuclear Regulatory Commission, 1987) and are generally equivalent to the qualification requirements that are in place at other ISFSIs, including the requirements for general managers and operators or Certified ISFSI Specialists. The staff concludes that the personnel qualification requirements stated in the SAR are equivalent to those specified for similar nuclear facilities and are therefore acceptable.

The applicant will evaluate certified operator trainee mastery of training objectives and provide pass/fail criteria. In the SAR, Section 9.4.1.1, the applicant committed to evaluate the physical condition and general health of personnel who are certified for operations that are important to safety. These personnel will be evaluated according to NRC Form 396, which is used to evaluate licensed operators at commercial nuclear reactors. The staff concludes that these commitments are acceptable.

In summary, the staff has determined that the applicant has provided sufficient details concerning its personnel training and qualifications to provide reasonable assurance that its training and certification program will satisfy the requirements of 10 CFR Part 72, Subpart I. Certain operations will be performed only by trained and certified operators, and the physical condition and general health of operators will be considered in the qualification of operators, as required by 10 CFR 72.192 and 72.194 of Subpart I. The qualifications and certifications of the operators will be inspected and evaluated following the issuance of a license to ensure regulatory compliance prior to the conduct of licensed operations at the Facility.

As described in the previous text, the staff has determined that the Facility training program, including the commitments made by the applicant, provide reasonable assurance of compliance with the standards in 10 CFR Part 72, Subpart I, and are consistent with the applicable

regulatory guidance. This training program includes specific training in ALARA principles. Based on the Facility description of its training program, the staff concludes that the training commitments are consistent with Regulatory Guide 8.8 (Nuclear Regulatory Commission, 1978), which provides guidance in training and instruction in ALARA principles for nuclear power plant personnel, and provide reasonable assurance that NRC requirements related to radiation protection training and ALARA principles will be satisfied.

10.1.4.3 Selection and Training of Security Guards

The requirements for the security organization is addressed in Chapter 18 of this SER.

10.1.5 Emergency Planning

The Emergency Plan is addressed in Chapter 16 of this SER.

10.1.6 Physical Security and Safeguards Contingency Plans

Physical Security is addressed in Chapter 18 of this SER.

10.2 Evaluation Findings

The staff has reviewed the SAR and has determined that PFS has established an acceptable plan to conduct the operations of the Facility. The staff has determined that:

- The conduct of operations described for the Facility meets the requirements of 10 CFR 72.40(a)(4) in that PFS will be qualified by training and experience to conduct the operations included in the license.

- The conduct of operations described for the Facility meets the requirements of 10 CFR 72.40(a)(9), 72.190, 72.192, and 72.194 in that PFS has provided a description of the procedures and policies that assure that operation of equipment and controls that are important to safety is limited to trained and certified personnel; has provided an adequate operator training and certification program; and has operator qualifications that assure that the physical condition and general health of operators will not cause operational errors that could endanger other workers or the health and safety of the public.

License Condition

LC10-1 PFS must submit a startup plan to the NRC prior to receipt and storage of fuel at the Facility.

10.3 References

Nuclear Regulatory Commission. 1978. *Information Relevant to Ensuring That Occupational Radiation Exposures at Nuclear Power Stations Will Be ALARA.* Regulatory Guide 8.8. Revision 3. Washington, DC: Nuclear Regulatory Commission.

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Nuclear Regulatory Commission. 1989. *Standard Format and Content for the Safety Analysis Report for an Independent Spent Fuel Storage Installation or Monitored Retrievable Storage Installation (Dry Storage).* Regulatory Guide 3.45. Revision 01. Washington, DC: Nuclear Regulatory Commission.

Nuclear Regulatory Commission. 1998. *Standard Review Plan for Spent Fuel Dry Storage Facilities.* NUREG–1567. Washington, DC: Nuclear Regulatory Commission.

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