72-22-ISFSI 6/19/00 STAFF-A

EXHIBIT

STAFF "A"

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

DOCKETED

RAS 2050

December 15, 1999

JANUARY 4, 2000 - REVISED IN ITS ENTIRETY TO CORECT CHAPTER 17

'00 AUG -7

Mr. John D. Parkyn
Chairman of the Board
Private Fuel Storage, L.L.C.
P.O. Box C4010
La Crosse, WI 54602-4010

SUBJECT: SAFETY EVALUATION REPORT FOR SYSTEMS NOT DIRECTLY ASSOCIATED WITH STORAGE CASKS (TAC NO. L22462)

Dear Mr. Parkyn:

By application dated June 20, 1997, Private Fuel Storage, L.L.C. (PFS), submitted an application to the Nuclear Regulatory Commission (NRC) for a license to operate an away-from-reactor independent spent fuel storage installation (ISFSI) on the reservation of the Skull Valley Band of Goshute Indians, a federally-recognized Indian Tribe. Subsequently, the NRC staff decided that its safety evaluation of the application would be completed in two segments: first, the staff would complete a safety evaluation report (SER) for systems not directly associated with the storage cask systems proposed by PFS for use at the PFS Facility ISFSI (PFSF); and second, the staff would complete an SER which included at least one of the cask systems proposed for use at the PFSF. It was determined that because of the length of time necessary to complete the review, evaluation, and rulemaking associated with cask certification, it might be useful to follow this approach.

Enclosed is the SER for the systems not directly associated with the storage casks chosen for use at the proposed PFSF. As the staff completed its review, it found several areas where PFS had not provided sufficient information for the staff to determine regulatory compliance. The major areas where sufficient information was not provided are (a) the site soil composition and the related issue of demonstrating the stability of the storage pads and foundation of the cask transfer building; and (b) analyses associated with the probability of military airplane crashes at the PFSF. Several meetings were held with PFS to discuss the information that was needed for the staff to complete its evaluation.

The staff's review of the probabilistic seismic hazard analysis (PSHA) submitted by PFS determined that the analysis needed to be revised to reflect the 2000-year return period earthquake. The applicant has submitted a revised analysis. However, the revised analysis was provided to the staff too late for the information to be reviewed in this SER. Therefore, the final conclusion on the PFS seismic analyses is not addressed in this SER. Similarly, a significant amount of information has been provided to the staff by PFS regarding the open items identified at the end of some of the chapters of this SER. However, any such information provided in commitment letters and in the SAR revisions after Revision 3 have not been analyzed for this SER and will be considered in a supplemental SER which will be issued in the Spring of 2000. Provision of additional information to address the open items identified by the staff is, of course, in support of the demonstration of compliance with all applicable regulatory requirements. Such demonstration is the basis for a licensing decision.

Template = SECY-028 SECY-02

NUCLEAR REGULATORY COMMISSION

Docket No. 72-22 Official Exh. No. A

In the matter of Private Fuel ~~Corp~~ Storage

Staff X IDENTIFIED X

Applicant RECEIVED X

Intervenor REJECTED

Cont'g Off'r

Contractor DATE 19 June 00

Other Witness: PAUL W. LAIN, RANDOLPH L. SULLIVAN

Reporter Vicky McDaniel

As a convenience for the reader, this SER contains chapter headings for all aspects of the licensing review, including the cask-specific chapters not addressed at this time. Because 10 CFR Part 72 is a systems-based and performance-oriented regulation, both cask-specific issues and other issues are sometimes addressed in a single chapter. This reflects a systems approach where the cask directly interfaces with another subsystem or where cask performance and performance of other aspects of the site or facility are related. Chapters in which a majority of the information is cask specific are not addressed at this time. The staff found that making a few findings in a given chapter, while deferring others, could lead to confusion for you and other readers of this SER. PFS must address all identified open items, as well as all applicable regulatory requirements, for the staff to complete its review and issue a supplemental SER on the systems not directly associated with storage casks. As noted above, information related to many of the open items has been provided to the staff and is currently under evaluation. The NRC staff will be available to meet with PFS to discuss these open items, if PFS would like further clarification or discussion.

PFS must provide an update to its Safety Analysis Report (SAR) to include all of the above information. In addition, it must include all responses to the staff's requests for additional information (RAIs). All commitment letters sent to the staff subsequent to the final RAI responses should also be included as an appendix to the revised SAR. In order for the staff to be able to complete its review and issue the supplemental SER in the Spring of 2000, this revised SAR must be submitted by January 18, 2000. Another SER covering the staff's review of systems directly associated with storage casks will be prepared after an amendment to the PFS application reflecting a certified cask system is received.

If you have any questions regarding this letter or the enclosure, please feel free to contact me at (301) 415-8518.

Sincerely,
ORIGINAL SIGNED BY /s/
Mark S. Delligatti, Senior Project Manager
Licensing Section
Spent Fuel Project Office
Office of Nuclear Material Safety
and Safeguards

Docket: 72-22

Enclosure: Safety Evaluation Report
cc: PFS Service Lists (Distribution Attached)

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SAFETY EVALUATION REPORT

OF THE SITE-RELATED ASPECTS OF

THE PRIVATE FUEL STORAGE FACILITY

INDEPENDENT SPENT FUEL STORAGE INSTALLATION

Docket No. 72-0022

TABLE OF CONTENTS

ACRONYMS

ALARA	As Low As Reasonably Achievable
ANS	American Nuclear Society
ANSI	American National Standards Institute
DOE	U.S. Department of Energy
DSHA	Deterministic Seismic Hazard Analysis
HMR	Hydrometeorological Report
ISFSI	Independent Spent Fuel Storage Installation
LES	Louisiana Energy Services
LLEA	Local Law Enforcement Agency
MTU	Metric Tons of Uranium
NRC	U. S. Nuclear Regulatory Commission
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
QA	Quality Assurance
RAI	Requests for Additional Information
SAR	Safety Analysis Report
SER	Safety Evaluation Report
TLD	Thermoluminescent Dosimeters
TMI-2	Three Mile Island Unit-2
USDA	U.S. Department of Agriculture

Safety Evaluation Report
of the Site-Related Aspects of
the Private Fuel Storage Facility
Independent Spent Fuel Storage Installation

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 CFR72.30.

(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; and

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

The applicant also submitted 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 site-related design, operational, and other safety aspects of the Facility, as described in the above submittals except for the Environmental Report. The Environmental Report will be the subject of a separate Environmental Impact Statement.

This SER addresses only those matters related to the site; it does not include the evaluation of the cask-specific or cask-dependent aspects of the Facility. The staff's assessment in this SER is based on whether the Facility meets the requirements of 10 CFR Part 72, but only to the extent that such determinations can be made independent of a dry cask storage system design.

The dry cask storage systems proposed for the Facility are the HI-STORM 100 Cask System (Docket No. 72-1014) and the TranStor™ Storage Cask System (Docket No. 72-1023). The U.S. Nuclear Regulatory Commission (NRC) is currently reviewing these cask systems for use under the general license provisions of 10 CFR Part 72, Subpart K. To minimize duplication, review of either cask system for site-specific use at the Facility is being delayed until one of the cask systems is approved and a certificate of compliance is issued. Before a license for the Facility is issued, the applicant needs to demonstrate that one of the proposed cask systems is acceptable for use at the Facility under the site-specific license provisions of 10 CFR Part 72. The final SER and licensing basis for the Facility will include consideration of at least one of the cask systems.

References

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, and May 19, August 10, August 27, and September 8, and September 21, 1999.

Private Fuel Storage Limited Liability Company. *Safety Analysis Report for the Private Fuel Storage Facility, Revision 3.* Docket Number 72-22. May 19, 1999.

Private Fuel Storage Limited Liability Company. *Emergency Plan for the Private Fuel Storage Facility, Revision 6.* Docket Number 72-22. September 8, 1999.

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.

1 GENERAL DESCRIPTION

1.1 Conduct of Review

Chapter 1 of the SAR provides a general, non-proprietary 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 staff and other interested parties with the pertinent features of the installation.

1.1.1 Introduction

The Facility is an ISFSI that uses dry cask storage technology. In accordance with 10 CFR 72.42, the Facility will be initially licensed for 20 years. Before the end of this license term, the applicant may submit an application to renew the license for another 20 years. All spent fuel will be transferred offsite and the Facility will be ready for decommissioning by the end of the second 20-year 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 that are approved under 10 CFR Part 71 will be used to transport the spent 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 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 construction of the Facility in September 2000 and plans 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 concrete storage cask. The storage system, consisting of approximately 4,000 storage casks, is passive and relies on natural convection for cooling.

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 is located between the two fences. The cask storage area, consisting of 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 × 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 systems proposed for the Facility are Holtec International's HI-STORM 100 Cask System and BNFL Fuel Solutions' (formerly Sierra Nuclear Corporation) TranStor™ Storage Cask System. These cask systems are described in detail in the Topical Safety Analysis Report for the HI-STORM 100 Cask System (Holtec International, 1997, Docket No. 72-1014)) and the SAR for the TranStor™ Storage Cask System (Sierra Nuclear Corporation, 1997, Docket No. 72-1023). These cask systems are canister-based and store spent fuel in a vertical orientation. Generally, the proposed cask systems consist of three components: the dual-purpose (storage and transportation) canister, the transfer cask, and the storage cask. The canister has a welded closure and is the confinement system for the stored fuel. The transfer cask provides radiation shielding and structural protection of the canister during transfer operations. The storage cask provides radiation shielding and structural protection of the canister during storage. The cask systems are passive and do not rely on any active cooling systems to remove spent fuel decay heat.

The spent fuel is loaded into the dual-purpose canister at the originating nuclear power plant. Before transport, the canister's lid is welded in place and the canister is drained, vacuum dried, filled with an inert gas, sealed, and leak tested. Shipping casks, approved under 10 CFR Part 71, are used to transport the canisters 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 canister is transferred from the shipping cask to the transfer cask, then from the transfer cask into the concrete storage cask. The concrete storage cask, loaded with the canister, is then closed, and moved to the storage area using a cask transporter and placed on a concrete pad in a vertical orientation.

The HI-STORM 100 and the TranStor™ cask systems are currently under NRC review for use under the general license provisions of 10 CFR Part 72, Subpart K. To minimize duplication, review of either cask system for site-specific use at the Facility is being delayed until one of the cask systems is approved and a certificate of compliance is issued. Before a license for the Facility is issued, the applicant needs to demonstrate that one of the proposed cask systems is acceptable for use at the Facility under the site-specific license provisions of 10 CFR Part 72. Thus, this SER does not include cask-specific or cask-dependent evaluations and findings at this time. The SER will be amended before the license for the Facility is issued to include an evaluation of whether the selected cask system is suitable for use at the Facility.

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 and BNFL Fuel Solutions are responsible for the design of the of the HI-STORM 100 and TranStor™ cask systems, respectively. 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 Chapter 1 of the SAR have sufficient detail to allow familiarization with the pertinent site-related features of the proposed ISFSI.

This SER does not include cask-specific or cask-dependent evaluations at this time; therefore, no findings have been made in this SER regarding the adequacy of the cask system information.

Open Items

1-1 The dry cask storage systems proposed for the Facility are currently under NRC review for use under the general license provisions of 10 CFR Part 72, Subpart K. Review of the cask systems for site-specific use at the Facility will be conducted when one of the cask systems is approved for use under the general license. Before a license for the Facility is issued, the applicant should demonstrate that the cask system is acceptable for use at the Facility under the site-specific license provisions of 10 CFR Part 72. Further, the applicant should ensure that cask information in the Facility SAR is consistent with the Final Safety Analysis Report for the specific dry cask storage system. The final SER will include consideration of the cask system.

1-2 The SAR should be updated to incorporate all information that was used as the basis for demonstrating compliance with 10 CFR Part 72, including information and commitments provided in the applicant's responses to the NRC's requests for additional information (RAIs).

1.3 References

Donnell, J.L. 1998. *Supplemental Response to RAIs*. Letter (June 15) to Director, Office of Nuclear Material Safety and Safeguards, Nuclear Regulatory Commission. La Crosse, WI: Private Fuel Storage Limited Liability Company.

Holtec International. 1997. *Topical 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–951312. Revision 1. Docket No. 72-1014. Marlton, NJ: Holtec International.

Parkyn, J.D. 1998. *Response to Request for Additional Information*. Letter (May 19) to Director, Office of Nuclear Material Safety and Safeguards, Nuclear Regulatory Commission. La Crosse, WI: Private Fuel Storage Limited Liability Company.

Parkyn, J.D. 1999. *Response to Request for Additional Information*. Letter (February 10) to Director, Office of Nuclear Material Safety and Safeguards, Nuclear Regulatory Commission. La Crosse, WI: Private Fuel Storage Limited Liability Company.

Private Fuel Storage Limited Liability Company. 1999. *Safety Analysis Report for Private Fuel Storage Facility*. Revision 3. Docket No. 72-22. La Crosse, WI: Private Fuel Storage Limited Liability Company.

Sierra Nuclear Corporation. 1997. Safety Analysis Report for the TranStor™ Storage Cask System. SNC–96–72SAR. Revision B. Docket No. 72-1023. Scotts Valley, CA: Sierra Nuclear Corporation.

2 SITE CHARACTERISTICS

2.1 Conduct of Review

Chapter 2 of the SAR discusses the geographical location of the Facility and meteorological, hydrological, seismological, geological, and volcanological characteristics of the site and surrounding area. It describes the population distribution within and around the Reservation, land and water uses, and associated site activities. It 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 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. However, the applicant needs to demonstrate that the Facility is designed adequately to withstand a 2000-year return period earthquake.

The information and analyses in 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. Also, the staff requires additional information in some review areas of Chapter 2 in order to determine regulatory compliance. This additional information is listed as Open Items in the Section 2.2 of this SER.

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 ISFS; 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 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.

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-ft 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 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 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 and the leased operation (since 1975) of the Tekoi Rocket Engine Test Facility on the south side of Hickman Knolls by Allied Techsystems. 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 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 5,300 acre-feet of water per year. The applicant estimates water needs at 8,500 gallons per day during construction and 3,600 gallons per day during operation. Assuming a conservative 365 days in a year, the water usage is estimated to be 9.52 acre feet per year during construction activities and 4.03 acre feet per year for operational activities. Therefore, the projected amount of water used during construction activities is about 0.18 percent (9.52 acre-feet divided by 5,300 acre feet used per year) of current total water production estimated in Skull Valley and 0.076 per cent (4.03 acre feet divided by 5,300 acre feet used per year) for operations. These water-use amounts attributed to the Facility are very small when compared to 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 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 on-site monitoring well, the applicant determined that operation of the Facility water well will have no measurable off-site effects on existing groundwater quality or levels (Stone & Webster Engineering Corporation, 1999b). 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 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 requirement 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 man-made 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. It 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 ft) lie between this site and the PFS Facility. The Tekoi Rocket Engine Test Facility is a commercial facility that performs activities including high explosive and rocket motor testing. It 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 ft). The Utah Test and Training Range is a federal site with activities that include testing air-to-ground and air-to-air munitions. The Tooele North Army Depot Area is a federal site that is 17 miles east-northeast of the PFS Facility. The Tooele South Army Depot Area is a federal site that performs activities that include incineration of retired nerve agents and is about 22 miles east-southeast of the PFS Facility. The Stansbury Mountains (elevation greater than 8,000 ft) 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 that have a potential to crash at the PFS 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 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 reviewed the information and is unable to verify the cumulative probability of aircraft crashes onto the proposed site taking into account all possible credible scenarios. The staff has determined that additional information is needed to assess the potential hazards from military aircraft flying through Skull Valley and helicopter flights over the Utah Test and Training Range and Skull Valley. The staff's evaluation is ongoing, and the consequence of potential aircraft crashes remains an open item in Section 2.2 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, lighting, 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 (NWS) (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 NWS, 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 ft above mean sea level), as a part of the regional climatology information, based on the long-term meteorological data collected by the NWS at the Salt Lake City International Airport (National Oceanic and Atmospheric Administration, 1992). Because the Stansbury and Oquirrh Mountains, with elevations exceeding 10,000 ft 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 provides the temperature data recorded at Dugway (approximately 12 miles south of the site at an elevation of 4,340 ft above mean sea level) and Iosepa South Ranch (about 12 miles north of the site at an elevation of 4,415 ft 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 provided annual average and average daily maximum temperature for the month of July recorded at the proposed site meteorological tower during 1997 and 1998 (RAI 4-1, Donnell 1999g).

The temperatures recorded at different sites are summarized in Table 2-1 of this SER. The highest 24-hr average temperature recorded in July 1999 is 83.8 °F. The average daily maximum temperature is defined as the average of peak temperatures for a month. The highest average daily maximum temperature recorded at the site for July 1997 is 92.6 °F.
Table 1 in (Parkyn, 1998) gives the measured solar radiation at the proposed site from December 1996 through March 1998. Solar radiation data before December 1996 was not recorded at the site. The 1997 annual maximum and average solar radiation are 988 W/m^2 and

189.4 W/m^2, respectively. Based on a 12-hr average, the site meteorological measurements show that the solar radiation at the site on several days exceeds 600 W/m^2. The maximum 12-hr average recorded in 1997 is 676 W/m^2, based on the measurements taken 8am–7pm on June 28, 1997. The maximum 24-hr average solar insolation measured on June 24, 1997, at the site is 355 W/m^2. Table 1 in (Parkyn, 1998) gives the measured humidity at the proposed site from December 1996 through March 1998. 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 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 it acceptable because reliable data sources, such as the NWS, were used and the data from December 1996 to March 1998 is appropriately summarized. However, the staff notes that some of this data is limited to a few years or is not presented because bounding values (e.g. atmospheric diffusion parameters) are assumed by the applicant. The adequacy of these values for use in other SAR analyses (e.g., the appropriate solar radiation value for the thermal analysis) are dependent on cask-specific design parameters that are not currently present in the SAR. Therefore, the staff will require additional information and analyses from the applicant in order to determine appropriate meteorological values for other analyses within the SAR. These are considered open items in Section 2.2 of this SER.

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 ft 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). However, as noted above, additional cask-specific information is necessary for the staff to determine compliance with the applicable regulatory requirements.

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†	86*	
July	92.2†/93*	94.4†	95*	92.6‡
August	89.4†/90*	91.3†	93*	
Average Daily Minimum (°F)				
June	55.4†/53*	53.2†	53*	NA
July	63.7†/62*	61.9†	62*	NA
August	61.8†/59*	59.3†	59*	NA
Annual Average (°F)				
June	69.1†/69*	69.0†	65.5*	67.2
July	77.9†/78.2*	78.2†	73.5*	71.7
August	75.6†/75.3*	75.3†	72.9*	75.3
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	0.1
February	-14†	-29†	NA	2.9

* Private Fuel Storage Limited Liability Company, 1999

† Ashcroft et al., 1992

‡ Highest 24-hr average temperature recorded in July at Private Fuel Storage Facility Meteorological Tower = 83.8 °F

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 the 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, man-made 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, lighting, 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 square miles, 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 ft above mean sea level at the southwest corner, falling to elevation 4,462 ft 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 its 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 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 site, and an evaluation of how the proposed structures affect site safety. Based upon 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 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 the 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 in. for a 72-hr General Storm and 10.2 in. for a 6-hr 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 hr in Basin A and 4.2 hr 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 hr and Basin B yielded a time of concentration of approximately 4 hr. In Basin A, the applicant's 11 hr estimate will yield a more conservative peak flood discharge (larger) than the 12 hr value derived by the staff. In Basin B, the difference between 4 hr and 4.2 hr 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 ft above mean sea level. The earth berm with top elevation of 4,507.5 ft above mean sea level contains the flood flow. The PMF overtops the low point of the access road embankment (elevation 4,502 ft) by approximately 4.5

ft. The cask storage pads are located downstream from the embankment where the flood water surface elevation is approximately 4.5 ft lower than the pad elevation. The PMF water surface elevation adjacent to the northeast corner of the Facility is approximately 4,456.8 ft above mean sea level. Therefore, the cask storage area is approximately 5 ft 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 ft downstream of the roadway embankment, 4456.7 ft adjacent to the roadway embankment, and 4,477.4 ft upstream of the roadway embankment above mean sea level. The SAR presented water surface elevations of 4,444.2 ft, 4,456.7, and 4477.4 ft 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 ft above mean sea level) by approximately 3.2 ft at an elevation of 4,478.2 ft. above mean sea level. The berm constructed immediately upstream of the Facility will have a top elevation of 4,480 ft above mean sea level and extend above the PMF water surface elevation approximately 1.8 ft (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 ft above mean sea level. The cask storage pad elevation is 4,462 ft above mean sea level, indicating the pad is approximately 4 ft 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 breech 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 ft 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 thee west component of the Facility berm. Should the berm fail, the flood water surface elevation could potentially rise approximately 3 ft. However, there will be a minimum 1 ft 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 breech will result, thereby funneling the flood flows through a concentrated area. The breech represents an increase in flood flow cross-sectional area, thereby reducing the water surface elevation in the vicinity of the breech. 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 ft 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 ft. However, at least a 2-ft 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 ft above the lake. A wave over 400 ft high would have to be generated at the Great Salt Lake and travel 24 miles to reach the site. Therefore, surge or seiche flooding should not impact the site.

The staff reviewed the discussion on probable maximum surge and seiche flooding and found it acceptable because this phenomena 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 surge and seiche flooding and found it acceptable because this phenomena 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 surge and seiche flooding and found it acceptable because this phenomena 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 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, lighting, 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-ft. The average annual groundwater discharge and recharge is between 30,000 and 50,000 acre-ft, with evapotranspiration accounting for 80 to 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 ft. 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-ft with a transmissivity of 2,675 ft/day^2. Annual discharge from pumping is estimated at 5,000 acre-ft 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 ft 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 ft but maintain a static water depth of 100 ft 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 has 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 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-ft 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 ft, did not intercept the water table. The groundwater table is greater than 100 ft 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 has 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 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 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 also includes responses to Round 1 RAIs 2-5 and 2-7 (Parkyn, 1999a; Donnell, 1999d) including two reports entitled "Fault Evaluation Study and Seismic Hazard Assessment Study—Final Report" (Geomatrix Consultants, Inc., 1999a) and "High-Resolution Seismic Shear Wave Reflection Profiling for the Identification of Faults at the Private Fuel Storage Facility, Skull Valley, Utah" (Bay Geophysical Associates, Inc., 1999). 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 than 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 (DE) 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 DE must be equivalent to the safe shutdown earthquake (SSE) for a nuclear power plant. (2) Regardless of the results of the investigations anywhere in the continental U.S., the DE 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 be designed to withstand the effects of natural phenomena such as earthquakes, tornadoes, lighting, 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 has reviewed information presented in Section 2.6 of the SAR, Geology and Seismology. The staff also reviewed relevant literature cited in the SAR and the revised "Fault Evaluation Study and Seismic Hazard Assessment Study—Final Report." In Revision 2 of the SAR (Private Fuel Storage Limited Liability Company, 1998), geologic and seismic information included (i) review of published and unpublished literature; (ii) reconnaissance geological mapping of the valley; (iii) 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) DSHA performed by Geomatrix Consultants, Inc. (Appendix 2D); (vi) consulting report on surface geomorphology features prepared by D. Curry of the University of Utah, at the request of Stone & Webster (Appendix 2C); and (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).

The applicant response to Round 1 RAIs 2-5 and 2-7 (Parkyn, 1999a; Donnell, 1999d; Geomatrix Consultants, Inc., 1999a; Bay Geophysical Associates, Inc., 1999), provides 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) PSHA; and (xiv) 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 staff has reviewed the information presented in Section 2.6 , Geology and Seismology, and accompanying Appendixes 2B, 2C, 2D, and 2E, of the SAR (Parkyn, 1999a,b; Donnell, 1999d,g,h; Geomatrix Consultants, Inc., 1999a; Bay Geophysical Associates, Inc., 1999) 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), 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 subsequent 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 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 northwestern 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 ft 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 ft deep and soil ridges up to 4 ft 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) 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 ft 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 ft high and

100 ft wide, were originally mapped as possible fault traces by Sack (1993). In the Revision 2 of the SAR (Private Fuel Storage Limited Liability Company, 1998), 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 to 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., 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).

Surface mapping and related field investigations by the applicant (1999) and 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, it appears 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, 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 coseismic 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), suggested in the Revision 2 of the SAR (Private Fuel Storage Limited Liability Company, 1998, Appendix 2A, p 2), 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, at least 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 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 it is a time-independent statement (i.e., it does not take into consideration the planned operating period of the Facility). Also, 10 CFR 72.102(f)(1) requires that analyses using the

Appendix A methodology should use a design peak horizontal acceleration equivalent to that of the safe shutdown earthquake for a nuclear power reactor. Furthermore, NUREG–0800 (Nuclear Regulatory Commission, 1997b, Section 2.5.2.6) states the NRC preference that the 84th-percentile value of the ground motion spectrum be used to calculate the peak horizontal acceleration for the reactor safe shutdown earthquake (Nuclear Regulatory Commission, 1997b).

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, the peak horizontal acceleration and peak vertical acceleration values from the seismic event would be 0.72 and 0.80g, respectively (Geomatrix Consultants, Inc., 1999b). 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 proposes to design the Facility to the ground motions produced by 1,000-year return period earthquakes. These design-ground motions 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).

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 SAR and the result of the staff's independent investigation. 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,

1998, 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, 1998, 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 is 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 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 and complete. 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 an aseismic 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. The staff evaluated the use of probabilistic assessment with this alternative interpretation and found that the Geomatrix Consultants, Inc. (1999a) model is acceptable. It is acceptable because it is conservative and that of the possible interpretations, the applicant's hazard curve bounds what the staff's independent analyses considers anticipated ground motions from potential seismic sources at the site in and around Skull Valley, Utah (Stamatakos,1999).

In summary, the staff found that the applicant's considerations of seismic source characteristics and associated uncertainties provide reasonable assurance that all significant source and capable faults have been identified and their characteristics and associated uncertainties are adequately 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.

Estimate of Ground Motion Attenuation

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 because (i) it represents the state-of-the-art knowledge and (ii) both the PFS site and Yucca Mountain have seismotectonic characteristics of the Basin and Range.

For purposes of the Facility PSHA, 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. 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 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.

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 estimated.

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. PSHA calculations include seismic hazard from each individual source and 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. The mean peak horizontal acceleration is 0.38 and 0.49g for 1,000- and 2,000-year return periods, respectively. The mean peak vertical acceleration is 0.36 and 0.54g for 1,000- and 2,000-year return periods, respectively. Equal-hazard response spectra for return periods of 1,000 and 2,000 years (mean annual probabilities of exceedance of 1×10^{-3} and 5×10^{-4}, respectively) were calculated and presented in Geomatrix Consultants, Inc. (1999c).

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 are 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 show that the ground motion from the East fault generally envelops those from the other sources. The 84th-percentile peak ground accelerations for the East fault are 0.72g in the horizontal direction and 0.80g in the vertical direction. When compared with the PSHA results, the controlling deterministic spectra generally are between the 5,000 and 10,000-year 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. (1999c) based on its site-specific PSHA results as reviewed in Stamatakos et al. (1999) and documented in detail in Geomatrix Consultants, Inc. (1999a). 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. (1999c). 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

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. (1999c). These studies resulted in design ground motion peak horizontal acceleration of 0.40g and peak vertical acceleration of 0.39g for 1,000-year return period earthquakes and a peak horizontal acceleration and peak vertical acceleration of 0.53g for a 2,000-year return period.

The applicant's exemption request specifies a 1,000-year return period to calculate design basis ground motions with the PSHA methodology. The applicant (Parkyn, 1999b) states (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 and TranStor™ systems 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.

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:

- 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 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 amended 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). Technically, PSHA has many advantages over DSHA. For example, 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. Most importantly, 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 level of ground motion that a particular structure, system, and component is designed to 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 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 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 and TranStor™ 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 DOE standard, DOE–STD–1020–94 (U.S. Department of Energy, 1994), 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 mean ground motion with a 2,000-year return period. Category-3 facilities in the DOE standard have potential accident consequences similar to a dry spent fuel storage facility.

- The Uniform Building Code and the National Earthquake Hazards Reduction Program (International Conference of Building Officials, 1994; Building Seismic Safety Council, 1995) both recommend using peak ground motion values that have a 90-percent probability of not being exceeded in 50 years for the seismic design of structures. Considering the radiological safety aspects of a dry spent fuel storage facility, conservative peak ground motion values that have a 99 percent likelihood of not being exceeded in the 20-year licensing period of the Facility are considered adequate for its seismic design. This exceedance probability corresponds to a return period of 2,000 years.

- The NRC has accepted a design seismic value that envelops the 50^{th}-percentile deterministic ground motion value and the 2,000-year return period probabilistic ground motion value for the TMI-2 ISFSI facility license.(Nuclear Regulatory Commission, 1998b; Chen and Chowdhury, 1998). The TMI-2 is designed to store spent nuclear fuel in dry storage casks. The applicant's 2,000-year PSHA response spectra generally envelops the 50^{th}-percentile updated DSHA response spectra (Stamatakos et. al., 1999). A lower design value of 50^{th}-percentile design earthquake is adequate because the passive design of the dry cask storage facility is inherently less hazardous and less vulnerable to earthquake-initiated accidents than an operating nuclear power reactor, which requires a 84^{th}-percentile design earthquake (Hossain et al., 1997).

- In its Fault Evaluation Study and Seismic Hazard Assessment Study—Final Report for the site, Geomatrix Consultants, Inc. (1999a) concluded that an appropriate design probability level for both vibratory ground motion and fault displacement for the site is 5×10^{-4} (or a 2,000-year return period).

Therefore, the staff concludes that additional analyses are needed to assess ground vibrations of the Facility and to approve the applicant's request for an exemption to 10 CFR 72.102(f)(1). The staff agrees that the use of the PSHA methodology is acceptable, however, the SAR analyses need to be revised to consider a 2,000-year return period, rather than a 1,000-year return period. These analyses are required to verify compliance with the applicable requirements of 10 CFR Part 72, Subpart E.

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. They 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 ft in the last 60 ka and 2 ft 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 in., 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 in. 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 in., 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 in. of cumulative displacement across 88 m of exposure in Trench T-2, with a fracture spacing between 3 and 5 ft. This suggests vertical displacement of less than 1 m accumulated across the entire width of the proposed site (approximately 5,000 ft) over the last several million years.

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 done and materials presented by the applicant provide reasonable assurance that the displacements due to faulting are smaller than 0.04 in. 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 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 of the SAR, Stability of Subsurface Materials, which refers to the following sections of the SAR for details: 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. The staff also reviewed information presented in Appendix 2A, Geotechnical Data Report, of the SAR and other data and analyses presented by the applicant (Parkyn, 1998, 1999a; Donnell, 1998; Donnell, 1999a,d,g,h,i,j; ConeTec, Inc., 1999; Parkyn, 1997).

Geotechnical characterization of the site was performed through a combination of field and laboratory testing. The first set of testing (Appendix 2A of the SAR) consisted of 22 borings: 16 borings located on a 4 × 4 grid of ~650-ft spacing (between grid points) within the proposed storage-pad area, two borings in the proposed waste-handling area, two borings along the access road, and two borings outside of the controlled area to the south. These borings were used mainly for conducting standard penetration tests, and several split-spoon samples were obtained along with the standard penetration tests. Also, nine undisturbed (shelby-tube) samples were obtained from seven boreholes. The undisturbed samples were used for laboratory triaxial and odometer testing to obtain strength and compressibility data. The split-spoon samples were used for laboratory index testing, such as Atterberg limits and percentage of fine fraction. The second set of testing (Donnell, 1999j; ConeTec, Inc., 1999) consisted of 39 cone penetrometer tests and 16 dilatometer tests. No additional samples were obtained through the cone penetrometer tests and dilatometer tests because the testing techniques do not provide for sample recovery, but the tests provided continuous profiles of soil characteristics useful for classification and stability evaluation of the subsurface materials.

Soil Classification

Soil classification was performed using 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 cone penetrometer test logs. Based on information from these sources, the subsurface materials at the site have been classified by the applicant as consisting of a relatively compressible top layer, 25–30 ft thick, underlain by much denser and stiffer material. The underlying layer is classified as dense sand and silt. The top layer was initially described in the SAR as a mixture of silt, silty clay, and clayey silt, but the description was revised after the examination of the cone penetrometer test logs (Donnell, 1999a,j; ConeTec, Inc., 1999) to reflect the preponderance of cohesionless silt (with varying amounts of sand and clay). The water table is at a depth of about 125 ft below the ground surface, (i.e., at about elevation 4,350 ft above mean sea level).

The staff reviewed the applicant's soil classification and found the division of the subsurface materials at the site into two layers is an oversimplification of the soil profile. Although both the standard penetration test and cone penetrometer test data support the occurrence of a stiff layer below a depth of 25–30 ft, the cone penetrometer test data also clearly indicate the occurrence of three different layers above the stiff layer: (i) a top layer, about 10 ft thick, in which the value of cone resistance (Q_t) lies between 12–17 tsf; (ii) a middle layer, also about 10 ft thick, in which Q_t is about 30–50 tsf; and (iii) a bottom layer, about 5 ft thick, in which Q_t is about 20–30 tsf. The top 10-ft layer at times includes a hardened upper part that is up to 2-ft thick with Q_t values in the 30–50 tsf range. This division into four layers (instead of two) is significant for foundation design because of its effect on the applicable soil-parameter values. For example, foundations for which the zone of influence lies within the top layer would be designed using a Q_t value in the 12–17 tsf range, whereas wider (or deeper) foundations for which the zone of influence extends into the second layer would be designed with larger values of Q_t. Therefore, the staff will require additional information on soil classification. This is considered as an open item in Section 2.2 of this SER.

Stability and Settlement of the Cask Storage Pads Under Static Loading

The cask storage pads, each 30 ft wide and 64 ft long with 5-ft length-wise spacing between pads, will be loaded to a bearing pressure of 1.94 ksf, considering the dead load plus long-term live load for a 30 ft × 64 ft × 3 ft concrete pad loaded with eight casks. The allowable bearing pressure of 4.0 ksf given in the SAR (Section 2.6.1.12.1) for the cask storage pads was calculated using an average undrained shear strength (c_u) of 2.2 ksf. Using this approach to calculate the allowable bearing pressure is not valid, because the parameter c_u is not a valid measure of the shear strength of cohesion-less soils. The applicant argued that the estimated allowable bearing pressure should be considered conservative because the shear strength of 2.2 ksf used in the calculation is close to the lowest applicable value for the site, having been determined from laboratory testing of specimens from the weakest soil horizon encountered at the site. The cone penetrometer test profiles (ConeTec, Inc., 1999, Appendix A) indicate the smallest values of cone resistance occur in the depth range of about 5–12 ft below the ground surface and indicates the soil within this depth range have the smallest shear strength. The test

specimens from which c_u of 2.2 ksf was determined were indeed obtained from the 5–12 ft depth range.

The settlement of the cask storage pad under the bearing pressure of 1.94 ksf is given in the SAR as 3.3 in, having been calculated using laboratory compressibility data. The settlement calculation was revised using the cone penetrometer test data, which gave a cask-pad settlement smaller than 1.0 in. (Donnell, 1999a). Independent calculations were performed by the staff using Meyerhof's procedure (Meyerhof, 1956) to determine the standard penetration resistance (N) and cone resistance (Q_t) values that are needed to ensure that the total cask-pad settlement does not exceed 1.0 in under the bearing pressure of 1.94 ksf. The calculations give the required values as $N = 4.8$ and $Q_t = 19.4$ tsf, which represent the minimum required values for averages of N and Q_t taken over a depth range of 3–33 ft (from the base of the foundation to a depth equal to the foundation width below the base). The profiles of N and Q_t indicate that their averages would exceed the required values within the 3–33 ft depth range. The width of the foundation (30 ft), which extends the zone of influence of the foundation beyond the top, relatively weak, soil layer, is essential for the conclusion that the averages of the measured N and Q_t would exceed the required values. For example, using cone penetrometer test–18 data from Appendix F of ConeTec, Inc. (1999), the available average Q_t would be 14.7 tsf for a 7-ft wide foundation at a 3-ft depth and 27.1 tsf for a 21-ft wide foundation at the same depth.

The staff reviewed the applicant's evaluation regarding the estimated allowable bearing pressure and found it acceptable. Independent calculations were performed by the staff using a procedure suggested by Meyerhof (1956) to determine the values of N or 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. The calculations give 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 and Appendix A of ConeTec, Inc. (1999)]. Therefore, the proposed cask-pad design provides an adequate safety factor against bearing failure under static loading.

The staff determined that the current foundation design for the cask storage pad which is 30-ft in width, 64-ft in length, and 3-ft in depth, and is loaded to a bearing pressure of 1,940 psf, is acceptable considering the geotechnical stability of the foundation under static loading. The staff notes this conclusion is only valid for this particular pad design and bearing pressure. Therefore, a new foundation design would require further staff evaluation.

Stability of Cask Storage Pads Under Dynamic Loading

The staff did not evaluate the SAR analysis regarding stability of the cask storage pads under dynamic loading because this analysis requires cask-specific information. Specification of a value for the sliding resistance between the casks and the foundation pad is needed to evaluate the overturning analysis. In addition, the applicant needs to demonstrate a sufficient factor of safety against sliding failure under dynamic loading. Therefore, the staff will require additional information on cask-specific sliding resistance values and other analyses in order to evaluate the stability of cask storage pads under dynamic loading. This is considered as an open item in Section 2.2 of this SER.

Stability of the Canister Transfer Building Foundation Under Static and Dynamic Loading

The staff did not evaluate the SAR analysis regarding stability of the canister transfer building foundation under static and dynamic loading because additional overturning and sliding analyses are needed to perform the evaluation. Therefore, the staff will require additional information and overturning and sliding analyses in order to evaluate the stability of canister transfer building. This is considered as an open item in Section 2.2 of this SER.

Liquefaction Potential

The subsurface materials are not likely to undergo liquefaction. The relatively compressible soil layers within the top 25–30 ft depth would not undergo liquefaction because of the depth of the water table (125 ft below the ground surface). Also, the material below a depth of 25–30 ft 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 of the SAR, Stability of Subsurface Materials. The risk of liquefaction or other soil instability due to vibratory ground motion has been sufficiently assessed. However, the staff does not find this section acceptable because:

- The soil classification is not adequate. The applicant needs to present detailed soil profiles and maps showing how soil properties vary laterally and with depth in the storage pad and Canister Transfer Building areas.

- The stability of the cask storage pads under dynamic loading is not adequate. The applicant needs to present overturning (i.e., bearing) and sliding analyses of the storage pads using cask-specific sliding values.

- The stability of the Canister Transfer Building under static and dynamic loadings is not adequate. The applicant needs to present additional overturning (i.e., bearing) and sliding analyses of the Canister Transfer Building.

As discussed above, these are considered as open items in Section 2.2 of this SER.

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. Consequently, the geotechnical design of the embankments is not presented.

The staff reviewed the 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 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 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 (when 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 phenomena 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 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. At this time, the staff cannot make a determination that the requirements in 10 CFR Part 72, Subpart E, Siting Evaluation Factors, have been fully satisfied. As discussed above, additional information and analyses are required for the staff to complete its review.

Open Items

2-1 Military Aircraft Hazards

As discussed in Section 2.1.2 of the SER, the staff has determined that additional information is needed to assess the potential hazards from military aircraft flying in the vicinity of the Facility.

2-2 Meteorological Characteristics

As discussed in Section 2.1.3.2 of the SER, the staff has determined that additional information regarding the site meteorological data is needed to assure appropriate use of the information in future cask-specific analyses.

2-3 Seismic Design and Exemption Request

As discussed in Section 2.1.6.2 of the SER, the staff has determined that additional information is needed to assess the affects of ground vibrations on the Facility. The applicant has requested an exemption to 10 CFR 72.102(f)(1) and proposes to use a PSHA approach with a 1,000-year return period, instead of the DSHA approach. The staff agrees with the PSHA approach, but it should use a 2,000-year return period instead of the applicant-proposed 1,000-year return period.

2-4 Soil Classification

As discussed in Section 2.1.6.4 of the SER, the staff has determined that additional information regarding soil classification (e.g., detailed soil profiles) is needed to assess stability of subsurface materials.

2-5 Stability of Cask Storage Pads and Canister Transfer Building

As discussed in Section 2.1.6.4 of the SER, the staff has determined additional information regarding stability of the cask storage pad and Canister Transfer Building is needed to assess stability of subsurface materials. Additional information that is required includes analyses that use cask-specific sliding resistance values and address overturning and sliding of the storage pad and Canister Transfer Building under a design basis earthquake.

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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 cask systems 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 that the proposed operating procedures are adequate for those Facility activities that are not dependent on the design of the cask system. These 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). These findings pertain only to facility operations that are not dependent on the design of the dry cask storage system. The staff has not reviewed operations that are dependent on the design of the cask system.

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

At the site, handling of the canisters containing spent fuel relies, largely, on cask specific procedures. Therefore, the staff has made no findings regarding the adequacy of the spent fuel handling system.

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 must 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. 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, 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 cask designs 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

If shipping cask repair or maintenance is necessary, the Facility plans to conduct the maintenance in the Operation and Maintenance building or at a vendor-designated location. Repair and maintenance of the cask systems rely on the use of cask specific procedures. The staff considers shipping cask repair and maintenance to be cask specific and has not developed evaluation findings for adequacy.

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 for Facility activities that are not dependent on the design of the cask system. These 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. This finding only pertains to facility operations that are not specific to or dependent on the design of the dry cask storage system. The staff did not review operations that are specific to or dependent on the design of the cask system.

Proposed License Condition

LC3-1 The staff has determined that a license condition should be imposed to assure that prior to opening the shipping cask, the gas inside the is sampled to verify that canister confinement boundary is intact.

3.3 References

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 3.48. Washington, DC: Nuclear Regulatory Commission, Office of Nuclear Regulatory Research.

Parkyn, J.D. 1998. *Response to Request for Additional Information.* Letter (May 19) to Director, Office of Nuclear Material Safety and Safeguards, Nuclear Regulatory Commission. La Crosse, WI: Private Fuel Storage Limited Liability Company.

Parkyn, J.D. 1999. *Response to Request for Additional Information.* Letter (February 10) to Director, Office of Nuclear Material Safety and Safeguards, Nuclear Regulatory Commission. La Crosse, WI: Private Fuel Storage Limited Liability Company.

Private Fuel Storage Limited Liability Company. 1999. *Safety Analysis Report for Private Fuel Storage Facility.* Revision 3. Docket No. 72-22. La Crosse, WI: Private Fuel Storage Limited Liability Company.

4 STRUCTURES, SYSTEMS, AND COMPONENTS AND DESIGN CRITERIA EVALUATION

The structures, systems, and components and design criteria evaluation relies, in part, on cask-specific or cask-dependent information. In addition, the applicant has not yet provided sufficient information for the staff to complete the site-specific aspects of the review. Therefore, the staff cannot make a determination regarding the adequacy of the structures, systems, and components and design criteria for the proposed Facility.

5 INSTALLATION AND STRUCTURAL EVALUATION

The installation and structural evaluation relies, in part, on cask-specific or cask-dependent information. In addition, the applicant has not yet provided sufficient information for the staff to complete the site-specific aspects of the review. Therefore, the staff cannot make a determination regarding the adequacy of the installation design and structural integrity of the proposed Facility.

6 THERMAL EVALUATION

The thermal evaluation relies, in part, on cask-specific or cask-dependent information. In addition, the applicant has not yet provided sufficient information for the staff to complete the site-specific aspects of the review. Therefore, the staff cannot make a determination regarding the adequacy of the thermal design of the proposed Facility.

7 SHIELDING EVALUATION

The shielding evaluation relies upon cask-specific or cask-dependent information and is, therefore, not included in this SER at this time.

8 CRITICALITY EVALUATION

The criticality evaluation relies upon cask-specific or cask-dependent information and is, therefore, not included in this SER at this time.

9 CONFINEMENT EVALUATION

The confinement evaluation relies upon cask-specific or cask-dependent information and is, therefore, not included in this SER at this time.

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 is 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 commissioning; 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 Preoperational Testing and Startup Operations

10.1.2.1 Preoperational 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, the staff has proposed as a license condition that PFS be required 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 complies 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 the 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.

Generalized 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, and the structural characteristics of the Facility.

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.

Proposed 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.

Nuclear Regulatory Commission. 1987. *Qualification and Training of Personnel for Nuclear Power Plants.* Regulatory Guide 1.8. Revision 2. Washington, DC: Nuclear Regulatory Commission.

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.

Parkyn, J.D. 1999. *Response to Request for Additional Information.* Letter (February 10) to Director, Office of Nuclear Material Safety and Safeguards, Nuclear Regulatory Commission. La Crosse, WI: Private Fuel Storage Limited Liability Company.

Private Fuel Storage Limited Liability Company. 1999. *Safety Analysis Report for Private Fuel Storage Facility.* Revision 3. Docket No. 72-22. La Crosse, WI: Private Fuel Storage Limited Liability Company.

11 RADIATION PROTECTION EVALUATION

11.1 Conduct of Review

The review of the health physics program of the proposed ISFSI included Chapter 7, Radiation Protection, of the SAR. Information included in the references cited in Section 11.4 was also considered in the review. Chapter 7 of the SAR, Radiation Protection, describes the radiation protection features of the proposed ISFSI that ensure that radiation exposures to workers and members of the public meet the regulatory requirements. The review of Chapter 7 considered how the information in the SAR addresses the following regulatory requirements:

- 10 CFR 20.1101(a) requires that a licensee develop, document, and implement a radiation protection program.

- 10 CFR 20.1101(b) requires that a licensee use sound radiation protection principles to achieve ALARA.

- 10 CFR 20.1101(c) requires that a licensee periodically (at least annually) review the radiation protection program.

- 10 CFR 20.1101(d) requires that a licensee, as part of the radiation protection program, establish a constraint for air emissions of radioactive materials to the environment such that a member of the public is not expected to receive a total effective dose equivalent in excess of 10 mrem (0.1 mSv) per year.

- 10 CFR 20.1201(a) requires that a licensee control occupational dose to the following annual dose limits: A total effective dose equivalent of 5 rem (0.05 Sv) or the sum of the deep-dose equivalent and committed dose equivalent to any individual organ or tissue other than the lens of the eye of 50 rem (0.5 Sv), whichever is most limiting, a dose equivalent of 15 rem (0.15 Sv) to the lens of the eye, and a shallow-dose equivalent of 50 rem (0.50 Sv) to the skin or an extremity.

- 10 CFR 20.1301(a) establishes dose limits for a member of the public, including a total effective dose equivalent of 0.1 rem (1 mSv) in a year, and a maximum dose in any unrestricted areas of 0.002 rem (0.02 mSv) in an hour.

- 10 CFR 20.1301(b) requires that if a licensee permits members of the public to have access to controlled areas, the limits for members of the public continue to apply to those individuals.

- 10 CFR 20.1301(d) requires that the licensee comply with the environmental radiation standards in 40 CFR Part 190.

- 10 CFR 20.1302(a) requires a licensee to perform radiation surveys and monitor radioactive materials in effluents in unrestricted and controlled areas to

demonstrate compliance with the dose limits for members of the public in 10 CFR 20.1301.

- 10 CFR 20.1302(b) requires that the licensee show compliance with the limits in 10 CFR 20.1301, by either demonstrating compliance with the dose limit to an individual by calculation or measurement, or by demonstrating that radioactivity in gaseous and liquid effluents to do not exceed the values in table 2 of Appendix B to Part 20, and the dose from external sources would not exceed 0.002 rem (0.02 mSv) in an hour and 0.05 rem (0.5 mSv) in a year.

- 10 CFR 20.1406 requires that an applicant describe how facility design and procedures for operation will minimize contamination and generation of radioactive waste, and facilitate decommissioning.

- 10 CFR 20.1501(a)(1) requires that a licensee make surveys necessary to comply with 10 CFR Part 20.

- 10 CFR 20.1501(c) requires that dosimeters that are used by licensee are processed and evaluated by a processor holding accreditation from the National Voluntary Laboratory Accreditation Program.

- 10 CFR 20.1701 requires that a licensee use process or other engineering controls to control the concentrations of radioactive material in the air.

- 10 CFR 20.1702 requires that when it is not practicable to apply process or other engineering controls, that the licensee shall increase monitoring and limit intakes by use of other controls, including access control, limitation of exposure times, use of respiratory protection, etc.

- 10 CFR 72.104(a) requires that, during normal operations and anticipated occurrences, the annual dose equivalent to any real individual beyond the controlled area must not exceed 25 mrem to the whole body, 75 mrem to the thyroid and 25 mrem to any other organ, from various sources, including planned discharges of radioactive materials to the environment.

- 10 CFR 72.104(b) requires that operational restrictions are established to meet ALARA objectives for radioactive materials in effluents.

- 10 CFR 72.104(c) requires that operational limits for radioactive materials in effluents are established to ensure that the dose limits in 72.104(a) are met.

- 10 CFR 72.106(b) requires that any individual located on or beyond the nearest controlled area boundary shall not receive a dose greater than 5 rem to the whole body or any organ from any design basis accident, and that the minimum distance from the spent fuel waste handling and storage facilities to the nearest boundary shall be at least 100 meters.

- 10 CFR 72.126(a) requires that radiation protection systems must be provided for areas and operations where onsite personnel may be exposed to radiation or airborne radioactive materials. Structures, systems, and components for which operation, maintenance, and inspections may involve occupational exposure, must be designed, fabricated, located, shielded, controlled and tested to control external and internal radiation exposures. The design must include means to, among other things, control access to areas of potential contamination or high radiation, measure and control contamination, minimize worker time, shield personnel.

- 10 CFR 72.126(c)(1) requires that, as appropriate for the handling and storage system, effluent systems must be provided, as well as methods for measuring the amount of radionuclides in the effluents.

- 10 CFR 72.126(c)(2) requires that 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 limit effluents to ALARA levels, and analyses must show that releases to the environment during normal operations and anticipated occurrences will be within the exposure limit given in 10 CFR 72.104.

11.1.1 As Low As Is Reasonably Achievable Considerations

This section evaluates whether the applicant has appropriately considered the goal of maintaining doses ALARA during the operation of the Facility. Section 7.1 of the SAR addressed ALARA considerations.

11.1.1.1 As Low As Is Reasonably Achievable Policy and Program

The ALARA policy and program for the proposed ISFSI are described in Section 7.1.1 of the SAR, Policy Considerations. The primary goal of the Radiation Protection Program is to minimize exposure to radiation such that the individual and collective exposure to personnel in all phases of operation and maintenance are kept ALARA. The ALARA program will maintain radiation exposures ALARA through the following methods:

- controlling and surveillance over internal and external radiation exposures to maintain worker and public exposures within permissible limits;

- ongoing reviews to determine how exposures may be reduced;

- sufficient training for personnel in radiation protection principles and procedures, protective measures, and emergency responses;

- giving radiation protection personnel sufficient authority to enforce safe Facility operation;

- making revisions to operating and maintenance procedures and modifications to Facility equipment and facilities when the proposed revisions will substantially reduce exposures at a reasonable cost; and

- ensuring that adequate equipment and supplies are provided for radiation protection work.

The ALARA program will follow the guidance of Regulatory Guides 8.10 and 8.8 (Nuclear Regulatory Commission, 1977, 1978) to ensure compliance with the requirements of 10 CFR 72.126 and 20.1101, which require radiation protection programs and systems.

The SAR states that the Facility management is committed to compliance with regulatory requirements regarding control of personnel exposures and will establish and maintain a comprehensive program at the Facility to keep individual and collective doses ALARA. The management will ensure that each staff member integrates appropriate radiation protection controls into work activities and each individual understands and follows procedures to maintain their radiation dose ALARA.

The ALARA program, as described in the SAR, includes using pertinent information concerning radiation exposure of personnel in design and operation activities. Applicable experience gained during the operation of nuclear power stations relative to radiation control is factored into procedures to ensure that the procedures continually meet the objectives of the ALARA program. Trends in the Facility personnel and job exposures will be reviewed to permit corrective actions to be taken with respect to adverse trends.

The staff considers that the implementation of the proposed ALARA program will provide reasonable assurance that doses to workers and members of the public will be maintained ALARA in accordance with the requirements of 10 CFR 20.1101(a–b) and 72.104(b) and (c), and 20.1101(c). The proposed program contains the applicable elements in Regulatory Guides 8.8 and 8.10, such as management commitment to the ALARA program and principles, written administrative procedures and instructions for operations involving potential radiation exposures, defining responsibility and authority for implementing the program, and using an effective measurement system to determine the success of the program and any trends in exposures.

11.1.1.2 Design Considerations

The description of the ALARA design considerations at the proposed ISFSI is provided in Section 7.1.2 of the SAR, Design Considerations. Specific features of the facility that consider ALARA include:

- use of thick shielding during all canister handling, transfer, and storage operations to minimize direct radiation levels;

- placement of the storage pads at a sufficient distance from the restricted area fence and controlled area boundary to assure doses are ALARA;

- adequate spacing between storage casks to permit workers to function efficiently during placement and removal of storage casks at the pads and during performance of maintenance and surveillance;

- use of metal canisters that are welded shut to confine radionuclides and prevent release of radioactive effluents from inside the canister;

- use of a passive system to require minimum maintenance and surveillance requirements by personnel;

- use of a temperature monitoring system that allows for remote readout of cask temperatures;

- use of power operated wrenches, where practical, to reduce the time associated with tasks involving bolt insertion and removal; and

- use of temporary shielding where it is determined to be effective in reducing total dose for a task.

The staff finds that the design of the proposed ISFSI will provide reasonable assurance that the doses to workers and members of the public will be maintained ALARA and meet the requirements of 10 CFR 72.126(a) because of the design and operating features listed above, including adequate shielding and features that minimize exposure of operating staff. The staff also finds that the design of the Facility adequately considers the minimization of contamination and generation of radioactive waste as required by 10 CFR 20.1406. The staff also finds that 10 CFR 72.126(d) is satisfied because the Facility uses welded canisters that are not opened at the Facility and, therefore, no effluents are expected.

11.1.1.3 Operational Considerations

The description of the ALARA operational considerations at the proposed ISFSI is located in Section 7.1.3 of the SAR, Operation Considerations. Plans and procedures at the ISFSI will be developed in accordance with Regulatory Guides 8.8 and 8.10 (Nuclear Regulatory Commission, 1978, 1977). Specific Facility operational considerations to achieve ALARA conditions include:

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

- Dry runs will be performed prior to canister transfer operations to train personnel on canister transfer procedures, and to refine procedures to achieve minimum probable exposures.

- Procedures and work practices will be used that reflect ALARA lessons learned from other ISFSIs that use dry cask storage.

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

- Surveys will be conducted as necessary to ensure that doses are maintained ALARA.

The NRC staff finds that the use of Regulatory Guides 8.8 and 8.10 (Nuclear Regulatory Commission, 1978, 1977) to plan operations to maintain doses ALARA is appropriate and will provide reasonable assurance that doses to workers and members of the public will be maintained ALARA. The use of casks that are welded closed and are surveyed for surface contamination prior to transport to the Facility meets the requirements of 10 CFR 20.1701 in that engineering controls are used to prevent airborne radioactivity. Surveys as required by 20.1501(a)(1) will be used to assure that personnel exposures are within 10 CFR Part 20 limits and maintained ALARA, and the surveys are identified as an element of the ALARA program. The operational elements listed above, including dry runs, and using lessons learned from similar operations, are accepted tools in implementing an effective ALARA program.

11.1.2 Radiation Protection Design Features

This section evaluates the radiation protection design features at the proposed ISFSI. Relevant information is contained in the SAR in Section 7.3, Radiation Protection Design Features (Private Fuel Storage Limited Liability Company, 1999).

11.1.2.1 Installation Design Features

The description of the installation radiation protection design features is provided in Section 7.3.1 of the SAR, Installation Design Features. Applicable portions of Regulatory Position 2 of Regulatory Guide 8.8 were followed in the design of the Facility, and are addressed in the following sections (e.g., access control, radiation shielding, etc.). The installation will be located far from populated areas, with the nearest town from the proposed ISFSI located over 10 miles away. The storage area will be located far from the controlled area boundary. The closest distance from a storage pad to the controlled area boundary will be 646 meters. The storage area will be located within a radiation area. The Canister Transfer Building is located within the same radiation area to minimize the route between the handling facility and storage pads, minimize additional traffic on the route, and maintain substantial distance from the controlled area boundary. Airborne radioactive material will be prevented by the use of the high-integrity welded canisters. The spent fuel will be maintained dry so no radioactive liquid will be available for release. All sources of radiation located on the site will be contained in heavily shielded shipping, storage, or transfer casks, except for low-level waste. The low-level waste consists of low-activity material and the dose rates on the outer surface of the low-level waste containers are expected to be negligible. Onsite work stations are located at large distances from the storage pads or are located in buildings with radiation shielding.

The staff finds that the use of Regulatory Position 2 of Regulatory Guide 8.8 (Nuclear Regulatory Commission, 1978) in designing the radiation protection features of the proposed ISFSI is appropriate. The installation design features include controls to provide reasonable assurance

that occupational and public exposures will be limited to levels that are within the limits of 10 CFR 72.104(a) and meet the ALARA requirements of 20.1101(b), and satisfy 10 CFR 72.1701. For example, the use of sealed canisters at the site provides reasonable assurance that contamination of the Facility and the generation of radioactive waste will be minimized in accordance with 10 CFR 20.1406, and will meet the allowable dose for members of the public and the ALARA requirements for effluents in 10 CFR 72.104(a), (b) and (c), and 72.126(d). The staff finds that the distance between the spent fuel handling and storage areas and the nearest boundary of the controlled area of the proposed ISFSI (646 meters) meets the minimum distance specified in 10 CFR 72.106(b), which is 100 meters. The radiation protection design requirements proposed by the applicant are, therefore, acceptable.

11.1.2.2 Access Control

The description of the access control to the proposed ISFSI is contained in Section 7.3.1 of the SAR, Installation Design Features. Access control to the restricted area is provided for both personnel radiological protection and Facility physical protection. The access control boundaries for the controlled areas and restricted areas are established along the site fence lines. The restricted area is the space that is controlled for purposes of protecting workers from exposure to radiation and for providing Facility physical security. The restricted area will contain all areas at the Facility at which the dose rate may exceed 2 mrem/hr. The controlled area is the area inside the site boundary surrounded by the controlled area fence. Dose rates outside the controlled area will not exceed 25 mrem/yr.

Access to the restricted area is controlled through a single access point in the Security and Health Physics Building. Provisions will be located in this building for donning and removing personal protective equipment, such as anti-contamination clothing or respirators, in the event of an accident or off-normal event leading to an area of the site becoming contaminated. This building will also contain provisions for personnel decontamination.

Under normal operations, no high radiation, very high radiation, contamination, or airborne radioactivity areas are expected to exist at the proposed ISFSI. However, radiation protection personnel will monitor radiation levels within the restricted area and may establish additional access requirements and area designations as needed.

The staff finds that the access control at the proposed ISFSI is acceptable, since it provides for security fencing, and limits access to a single point. The access point is controlled within the Security and Health Physics Building. This prevents the entry into radiologically controlled areas of unauthorized personnel. The description of the access control at the proposed ISFSI is acceptable and meets the requirements of 72.126(a)(3), by limiting access to radiologically controlled areas.

11.1.2.3 Radiation Shielding

The evaluation of the radiation shielding relies, in part, on cask-specific or cask-dependent information and is, therefore, not included in this SER at this time.

11.1.2.4 Confinement and Ventilation

The evaluation of confinement and ventilation systems relies, in part, on cask-specific or cask-dependent information and is, therefore, not addressed in this SER at this time.

11.1.2.5 Area Radiation and Airborne Radioactivity Monitoring Instrumentation

The description of the area radiation and airborne radioactivity monitoring instrumentation at the proposed ISFSI is provided in Sections 7.3.5, Area Radiation and Airborne Radioactivity Monitoring Instrumentation, and 7.6.1, Effluent and Environmental Monitoring Program, of the SAR. All spent fuel that will be stored on the site will be contained within canisters that are welded shut and there are no credible events that could result in the release of radioactive material from within the canisters or unacceptable increases in direct radiation levels. Therefore, area radiation and airborne radioactivity monitors are not needed at the storage pads. However, thermoluminescent dosimeters (TLDs) will be used to record dose rates in the restricted area and along the controlled area boundary fence which would be able to identify increases in the dose rate at the proposed ISFSI.

The Canister Transfer Building will be equipped with local radiation monitors with audible alarms to provide personnel with warning of abnormal radiation levels. Portable monitors will be used to perform airborne monitoring during canister handling operations to detect minor releases of loose contamination on the exterior of the canisters. Continuous air monitors will be located in the exhaust of each canister transfer cell. There are no anticipated liquid or gaseous effluent releases from the proposed ISFSI during storage.

The staff finds that the radiation monitoring instrumentation described in the SAR meets the requirements of 72.126(c)(2), which requires that areas containing radioactive materials must be provided with systems for measuring the direct radiation levels in these areas. The local area monitors include alarm systems to warn workers of unusual levels of radiation in the area. Continuous air monitoring will be performed during canister handling activities to assure that airborne radioactivity is within allowable levels as required by 10 CFR 20.1501(a)(1).

11.1.3 Dose Assessment

Dose assessment relies, in part, on cask-specific information and is, therefore, not included in this SER at this time.

11.1.4 Health Physics Program

Information about the health physics program is contained in Section 7.5 of the SAR, Radiation Protection Program (Private Fuel Storage Limited Liability Company, 1999).

11.1.4.1 Organization

The health physics program organization is described in Section 7.5.1 of the SAR, Organization. The Radiation Protection Manager, who reports to the General Manager, is responsible for administering the radiation protection program and for the radiation safety of the Facility. The responsibilities of the Radiation Protection Manager and the radiation protection technicians are consistent with the guidance contained in Regulatory Guides 8.10 and 8.8 (Nuclear Regulatory Commission, 1977, 1978).

The staff finds that the proposed radiation protection program satisfies 10 CFR 20.1101(a) with regard to the program organization described above, since it provides for a Radiation Protection Manager, who reports directly to the General Manager, and radiation protection technicians.

11.1.4.2 Equipment, Instrumentation, and Facilities

The equipment, instrumentation, and facilities that will be utilized in the health physics program at the proposed ISFSI are described in the SAR in Section 7.5.2, Equipment, Instrumentation, and Facilities. A sufficient inventory and variety of operable and calibrated portable and fixed radiological instrumentation will be maintained to allow for effective measurement and control of radiation exposure and radioactive material and to provide backup capability for inoperable equipment. Equipment will be appropriate to enable the assessment of sources of gamma, neutron, beta, and alpha radiation, including the capability to measure the range of dose rates and radioactivity concentrations expected. The radiological instrumentation proposed at the Facility in the radiological control program is properly selected, operated, maintained, and calibrated and includes the following:

- low-level waste contamination meters,
- beta/gamma portable survey meters,
- alarming beta/gamma personnel friskers,
- portable air samplers,
- external dosimetry devices used for monitoring whole body exposure, including TLDs and self-reading dosimeters or digital alarming dosimeters,
- respiratory protection equipment used to protect against airborne radioactivity,
- anti-contamination clothing to protect against removable contamination, and
- equipment necessary to conduct a bioassay program in accordance with Regulatory Guide 8.26, Application of Bioassay for Fission and Activation Products (Nuclear Regulatory Commission, 1980).

The staff finds that the requirements of 10 CFR 20.1101(a) are met in that the health physics equipment, instrumentation, and facilities described in the SAR are adequate to perform surveys of direct radiation and airborne radioactivity, as one element of a health physics program.

11.1.4.3 Policies and Procedures

The health physics program policies and procedures at the proposed ISFSI are described in Section 7.5.3 of the SAR, Procedures. Radiological practices used to control exposure include the following procedures:

- performing badging functions for access authorization to the restricted area;
- issuing personnel dosimetry and monitoring, recording, and tracking individual exposures;
- performing radiological safety training and refresher training;
- performing ALARA reviews of plant procedures and monitoring of operations;
- determining radiation doses on a periodic basis at restricted area and controlled area boundaries using TLDs;
- issuing, revising, and terminating radiation work permits and standing radiation work permits;
- roping off, barricading, and posting radiation control zones;
- decontaminating personnel, equipment, and areas;
- performing radiation surveys and smear swab sampling, counting, and calculation;
- calibrating detection, monitoring, and dosimetry instruments;
- quantifying airborne radioactivity; and
- maintaining records of the radiation protection program, including audits and other reviews of program content and implementation; radiation surveys; instrument calibrations; individual monitoring results; and records required for decommissioning.

The staff finds that the description of the health physics program policies and procedures, including the elements listed above, is sufficient to provide reasonable assurance that the health physics program will be implemented in accordance with 10 CFR 20.1101(a) and (b). The use of TLDs to determine dose rates at the edge of the controlled area satisfies 10 CFR 20.1302(a), which requires that surveys of radiation levels are made to assure compliance with the dose limits for individual members of the public. The radiation protection program procedures provide

reasonable assurance that the Facility will minimize contamination of the Facility and the environment in accordance with 10 CFR 20.1406 by the use of smear surveys to identify areas of contamination and limiting access to contamination areas. Performing radiation surveys and smear swab sampling, counting, and calculation are used as required by 10 CFR 20.1501(a)(1). Procedural controls to limit intakes of radioactive materials are in accordance with 10 CFR 20.1702, in that access may be limited, and respiratory protection may be used if engineering controls are not effective in limiting airborne radioactivity. The staff notes that the SAR does not indicate the frequency of review of the health physics program. The provisions of 10 CFR 20.1101(c) require that the health physics program be reviewed at least annually.

11.2 Evaluation Findings

Based on a review of the information in the SAR, the following evaluation findings can be made regarding the proposed ISFSI:

- The staff has reviewed the description of the ALARA program of the ISFSI and found reasonable assurance that occupational radiation exposures will be limited to levels that are ALARA, in compliance with 10 CFR 20.1101(b) and 72.104(b). The staff found that this will be achieved by acceptable means including minimizing contamination in accordance with 10 CFR 20.1406, using proper surveys in accordance with 10 CFR 20.1501, and using controls in compliance with 10 CFR 20.1701, 20.1702 and 72.126(a).

- The staff has reviewed the Health Physics program at the ISFSI and found that it has been adequately described. The staff found that the Health Physics program provides reasonable assurance that radiation exposures will be ALARA in accordance with 10 CFR 20.1101(b). The staff found that contamination will be minimized in accordance with 10 CFR 20.1406. The staff found that the description of the Health Physics program provides reasonable assurance that controls will be used as necessary to limit intakes of radionuclides in compliance with 10 CFR 20.1702. The staff found that the description of the Health Physics program provides reasonable assurance that compliance with dose limits will be demonstrated through surveys of radiation levels for workers and members of the public in accordance with 10 CFR 20.1302(a), 20.1501(a), and 72.126(c).

11.3 References

Holtec International. 1997. *Topical 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–951312. Revision 1. Docket 72-1014. Marlton, NJ: Holtec International.

Nuclear Regulatory Commission. 1977. *Operating Philosophy for Maintaining Occupational Radiation Exposures As Low As Is Reasonably Achievable.* Regulatory Guide 8.10. Revision 1. Washington, DC: Nuclear Regulatory Commission.

Nuclear Regulatory Commission. 1978. *Information Relevant to Ensuring That Occupational Radiation Exposures at Nuclear Power Stations Will Be As Low As Is Reasonably Achievable.* Regulatory Guide 8.8. Revision 3. Washington, DC: Nuclear Regulatory Commission.

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Private Fuel Storage Limited Liability Company. 1999. *Safety Analysis Report for Private Fuel Storage Facility.* Revision 3. Docket No. 72-22. La Crosse, WI: Private Fuel Storage Limited Liability Company.

Sierra Nuclear Corporation. 1997. *Safety Analysis Report for the TranStor™ Storage Cask System.* SNC–96–72SAR. Revision B. Docket 72-1023. Scotts Valley, CA: Sierra Nuclear Corporation.

12 QUALITY ASSURANCE

12.1 Conduct of Review

This chapter of the SER evaluates the applicant's Quality Assurance (QA) Program. The QA Program submitted by the applicant is described in the "Private Fuel Storage Quality Assurance Program Description" dated August 30, 1996, as supplemented by Chapter 11, Quality Assurance, of the Safety Analysis Report (SAR).

The requirements of 10 CFR Part 72, Subpart G, and the guidance in the draft final report of NUREG 1567, "Standard Review Plan for Spent Fuel Storage Facilities," were used to perform a review of the QA Program. The purpose of the review was to determine whether the QA Program complied with the requirements of 10 CFR Part 72, Subpart G. The staff's evaluation of the QA Program for the PFS facility is given below.

12.1.1 Organization

The description of the PFS organization was reviewed for conformance with the following requirements:

- 10 CFR 72.142, "Quality assurance organization," requires that: "The licensee shall be responsible for the establishment and execution of the quality assurance program....The licensee shall clearly establish and delineate in writing the authority and duties of persons and organizations....The quality assurance functions are: (a) Assuring that an appropriate quality assurance program is established and effectively executed and (b) Verifying...that activities affecting the functions that are important to safety have been correctly performed. The persons and organizations performing quality assurance functions must have sufficient authority and organizational freedom....The persons and organizations performing quality assurance functions shall report to a management level that assures that the required authority and organizational freedom...are provided."

In its application, PFS stated that: (1) the QA program applies to all activities affecting quality; (2) the QA program assures safe operation of facilities for independent spent nuclear fuel storage, high level radioactive waste storage, and use of radioactive shipping packagings; (3) QA personnel are responsible for establishing and executing the QA program which meets the requirements of 10 CFR Part 72, Subpart G; (4) QA personnel have the authority and organizational freedom to assure that applicable requirements are met; and (5) QA personnel have direct access to the PFS Board of Directors. In its QA Program, the applicant describes the interrelationships, responsibilities, and authority of persons and organizations for the day-to-day execution of the QA Program.

The staff reviewed the QA Program and determined that PFS established and delineated the authority and duties of persons and organizations to assure that its QA program is effectively executed. The staff determined that PFS organizations performing QA functions have sufficient authority and organizational freedom to perform their duties, and that activities affecting the functions that are important to safety are correctly performed. The staff, therefore, concludes

that the PFS organizational structure, responsibilities, and authority, as described in the QA Program, satisfy the requirements specified in 10 CFR 72.142.

12.1.2 Quality Assurance Program

The description of the PFS QA program was reviewed for conformance with the following requirements:

- CFR 72.144, "Quality assurance program," requires that: "The licensee shall establish...a quality assurance program....The licensee...shall provide control over activities affecting the quality of the identified systems, structures, and components to an extent commensurate with the importance to safety....The licensee shall provide for indoctrination and training of personnel performing activities affecting quality....The licensee shall review the status and adequacy of the quality assurance program at established intervals."

In its application, PFS stated that: (1) the QA program is in full compliance with the requirements of 10 CFR Part 72, Subpart G; (2) the QA program is comprised of the QA Program description and QA procedures which contain detailed implementing instructions; (3) the QA program sets forth the requirements for the control of quality in the design, fabrication, operation, and maintenance of ISFSIs and the use of shipping containers; (4) the QA program provides control over activities affecting quality in systems, structures, and components that are important to safety; (5) training and evaluation of personnel qualifications are required for all QA functions; and (6) the QA program will be reviewed at established intervals to assure its adequacy and status, and that the program is being effectively implemented.

The staff reviewed the QA Program and determined that PFS has established a QA program that provides control over activities affecting the quality of the identified structures, systems, and components to an extent commensurate with the importance to safety. The staff determined that the QA program provides for indoctrination and training of PFS personnel performing activities affecting quality and requires that qualified PFS personnel review the QA program for adequacy at established intervals. The staff, therefore, concludes that the QA Program, as described in the SAR, satisfies the requirements of 10 CFR 72.144.

12.1.3 Design Control

The description of the PFS design control process was reviewed for conformance with the following requirements:

- 10 CFR 72.146(a), "Design control," requires that: "The licensee shall establish measures to assure that applicable regulatory requirements and the design basis...are correctly translated into specifications, drawings, procedures, and instructions...."

- 10 CFR 72.146(b) requires that: "The licensee shall establish measures for the identification and control of design interfaces and for coordination among participating design organizations...The licensee shall apply design control

measures to items such as: criticality physics, radiation, shielding, stress, thermal, hydraulic, and accident analyses; compatibility of materials; accessibility for in-service inspection, maintenance, and repair; features to facilitate decontamination; and delineation of acceptance criteria for inspections and tests....For the verifying or checking process, the licensee shall designate individuals or groups other than those who were responsible for the original design...."

- 10 CFR 72.146(c) requires that: "The licensee shall subject design changes...to design control measures commensurate with those applied to the original design."

In its application, PFS stated that it will establish the requirements to assure that structures, systems, and components are designed, added, deleted, or modified in accordance with applicable regulatory requirements, codes, and standards. Specifically, the applicant states that: (1) The design control process shall be implemented in accordance with written procedures; (2) Design input and criteria are translated into specifications, drawings, procedures, calculations, instructions, and procurement documents prepared and reviewed by qualified personnel; (3) The procedures shall provide identification and control of design interfaces and for coordination among participating design organizations; (4) The procedures shall provide for the review of items such as stress, hydraulic, thermal, criticality physics, radiation, shielding, and accident analyses; compatibility of materials; accessibility for inspection, maintenance, and repair; features to facilitate decontamination; and delineation of acceptance criteria for inspections and tests; (5) The procedure shall provide for a design review by qualified personnel other than those performing the design; and (6) Any design change or field change shall be subjected to the same design control measures as specified for the original design.

The staff reviewed the QA Program and determined that PFS has established measures to assure that applicable regulatory requirements and the design basis are correctly translated into specifications, drawings, procedures, and instructions. The staff determined that the QA Program has established measures to assure that design interfaces are identified and controlled; there is coordination among participating design organizations; design control measures apply to items important in the development of the design; design changes are subjected to design control measures commensurate with those applied to the original design; and individuals or groups, other than those who were responsible for the original design, will review the design. The staff, therefore, concludes that PFS's program for design control, as described in the QA Program, satisfies the requirements of 10 CFR 72.146.

12.1.4 Procurement Document Control

The description of the PFS procurement document control process was reviewed for conformance with the following requirements:

- 10 CFR 72.148, "Procurement document control," requires that: "The licensee shall establish measures to assure that applicable regulatory requirements, design bases, and other requirements...are included or referenced in the

documents for procurement....the licensee shall require contractors or subcontractors to provide a quality assurance program."

In its application, PFS stated that it will establish measures to assure that procurement documents covering materials, equipment, and services specify appropriate quality requirements. Specifically, the applicant states that: (1) The procurement documents shall specify or reference the applicable requirements, design bases, codes, and standards to assure quality, and that purchase orders shall include specifications which contain all the information necessary to assure that material, equipment, and services are of adequate quality; (2) All procurement activity shall be performed in accordance with written procedures delineate the requirements for preparation, review, approval, and control of procurement documentation; (3) To the extent necessary, procurement documents shall require suppliers of material, equipment, and services to have QA programs complying with the pertinent provisions of 10 CFR Part 72, Subpart G.

The staff reviewed the QA Program and determined that PFS has establish measures to assure that applicable regulatory requirements, design bases, and other requirements are included or referenced in the documents for procurement, and that the contractors or subcontractors are required, as appropriate, to provide quality assurance programs. The staff, therefore, concludes that PFS's procurement document control program, as described in the QA Program, satisfies the requirements of 10 CFR 72.148.

12.1.5 Instructions, Procedures, and Drawings

The description of the PFS instruction, procedures, and drawings development and control process was reviewed for conformance with the following requirements:

- 10 CFR 72.150, "Instructions, procedures, and drawings," requires that: "The licensee shall prescribe activities affecting quality by documented instructions, procedures, or drawings....The instructions, procedures, and drawings must include appropriate quantitative or qualitative acceptance criteria for determining that important activities have been satisfactorily accomplished."

In its application, PFS stated that it will establish the measures to assure that activities affecting quality are performed in accordance with approved instructions, procedures, and drawings. Specifically, the applicant states that: (1) Procedures shall be developed and implemented for those operations affecting quality; (2) All instruction, procedures, and drawings are to be developed, reviewed, approved, utilized, and controlled in accordance with the requirements of approved procedures; and (3) Procedures and instructions shall be established and maintained to assure that sufficient records are specified, reflect the quality of work performed, and comply with appropriate codes, standards, and regulatory requirements.

The staff reviewed the QA Program and determined that PFS has prescribed activities affecting quality by documented instructions, procedures, or drawings, and that the instructions, procedures, and drawings will include appropriate quantitative or qualitative acceptance criteria for determining that activities important to quality have been satisfactorily accomplished. The

staff, therefore, concludes that PFS's control of instructions, procedures, and drawings, as described in the QA Program, satisfies the requirements of 10 CFR 72.150.

12.1.6 Document Control

The description of the PFS document control process was reviewed for conformance with the following requirements:

- 10 CFR 72.152, "Document control," requires that: "The licensee shall establish measures to control the issuance of documents such as instructions, procedures, and drawings....These measures must assure that documents, including changes, are reviewed for adequacy, approved for release by authorized personnel, and distributed and used at the location where the prescribed activity is performed. These measures must assure that changes to documents are reviewed and approved."

In its application, PFS stated that it will control the issue, use, review, approval, distribution, and revision of quality related documents. Specifically, the applicant states that: (1) Procedures shall be developed to identify individuals/organizations responsible for control, review, approval, and issuance of documents; (2) Documents, including revisions, that are to be controlled, shall be prepared, reviewed, and approved by qualified personnel using documented control procedures; and (3) Documents shall be distributed to, and used at, the location where the activity prescribed by the document is performed.

The staff reviewed the QA Program and determined that PFS has established measures to control the issuance of documents such as instructions, procedures, and drawings; assure that documents, including changes, are reviewed for adequacy, approved for release by authorized personnel, and distributed and used at the location where the prescribed activity is performed; and assure that changes to documents are reviewed and approved by authorized personnel. The staff, therefore, concludes that PFS's document control program, as described in the QA Program, satisfies the requirements of 10 CFR 72.152.

12.1.7 Control of Purchased Material, Equipment, and Services

The description of the PFS purchased material, equipment, and services control process was reviewed for conformance with the following requirements:

- 10 CFR 72.154(a), "Control of purchased material, equipment, and services," requires that: "The licensee shall establish measures to assure that purchased material, equipment and services...conform to the procurement documents."

- 10 CFR 72.154(b) requires that: "The licensee shall have available documentary evidence that material and equipment conform to the procurement specifications prior to installation or use."

- 10 CFR 72.154(c) requires that: "The licensee or designee shall assess the effectiveness of the control of quality by contractors and subcontractors."

In its application, PFS stated that it will assure that purchased material, equipment, and services, whether purchased directly or through contractors and subcontractors, conform to the procurement documents. Specifically, the applicant states that: (1) Procedures shall be implemented and used for determining supplier selection and evaluation and that this would include the suppliers's capability to comply with codes and standards, supplier performance, and review of the supplier's QA program and facility operation; (2) Supplier performance evaluations shall be performed based on the importance, complexity, and quantity of the product or services; and (3) Receipt inspection consistent with the importance and complexity shall be performed using approved procedures to assure: (a) The material, components, or equipment is properly identified and corresponds with the receiving documentation; (b) The material, component, or equipment, and acceptance records are inspected and are acceptable; (c) Inspection records and certificates of conformance attesting to the acceptance of material and components are available; and (d) Accepted items are identified as to the inspection status prior to release.

The staff reviewed the QA Program and determined that PFS has established measures to assure that purchased material, equipment and services conform to the procurement documents; documentation will be available showing that material and equipment conform to the procurement specifications; and the effectiveness of the control of quality by contractors and subcontractors will be assessed. The staff, therefore, concludes that PFS's control of purchased materials, equipment, and services, as described in the QA Program, satisfies the requirements of 10 CFR 72.154.

12.1.8 Identification and Control of Materials, Parts, and Components

The description of the PFS material, parts, anc components identification and control process was reviewed for conformance with the following regulation:

- 10 CFR 72.156, "Identification and control of materials, parts, and components," requires that: "The licensee shall establish measures for the identification and control of materials, parts, and components. These measures must assure that identification of the items is maintained...either on the item or on records traceable to the item as required, throughout fabrication, installation, and use of the item."

In its application, PFS stated that it will identify and control material, parts, and components from the time of receipt through installation and use. Specifically, the applicant states that: (1) Approved instructions and procedures shall be implemented for the identification and control of materials, parts, and components; (2) An identification system shall be established using purchase order numbers, heat numbers, serial numbers, or other means to identify and control materials, parts, and components; and (3) Specifications shall require that materials, parts, and components are identified by some means and shall require that documentation have identification providing traceability to the item.

The staff reviewed the QA Program and determined that PFS has established measures for the identification and control of materials, parts, and components. The staff determined that PFS's measures assure that identification of the items are maintained either on the items or on records traceable to the items, throughout fabrication, installation, and use of the items. The staff,

therefore, concludes that PFS's identification and control of materials, parts, and components, as described in the QA Program, satisfy the requirements of 10 CFR 72.156.

12.1.9 Control of Special Processes

The description of the PFS special processes controls were reviewed for conformance with the following requirements:

- 10 CFR 72.158, Control of special processes," requires that: "The licensee shall establish measures to assure that special processes, including welding, heat treating, radioactive waste processing, and non-destructive testing, are controlled and accomplished by qualified personnel using qualified procedures, in accordance with applicable codes, standards, specifications, criteria, and other special requirements."

In its application, PFS stated that it will assure that special processes, including welding, heat treating, radioactive waste processing, and non-destructive testing, are controlled and accomplished by qualified personnel, using qualified procedures, in accordance with applicable codes, standards, specifications, criteria, and other special requirements. Specifically, the applicant states that: (1) Special processes shall be planned through items such as documented work instructions defining the sequence of operations, special environments, suitable equipment, and criteria for workmanship standards; (2) Each special process shall be performed in accordance with instructions, procedures, drawings, checklists, or other appropriate means; (3) Equipment used for accomplishing special processes shall be calibrated, maintained, stored, handled, and issued in accordance with applicable procedures; and (4) Personnel shall be qualified to assure proficiency in the special skills required for the process.

The staff reviewed the QA Program and determined that PFS has establish measures to assure that special processes are controlled and accomplished by qualified personnel using qualified procedures. The staff determined that these measures include planning and performing special processes using instructions, procedures, drawings, checklists, or other appropriate means; qualifying the personnel performing special processes to assure their proficiency in the skills required for the process; and specifying the proper use and control of equipment, materials, and supplies to be used in the performance of special processes. The staff, therefore, concludes that PFS's control of special processes, as described in the QA Program, satisfies the requirements of 10 CFR 72.158.

12.1.10 Inspection

The description of the PFS inspection process was reviewed for conformance with the following requirements:

- 10 CFR 72.160, "Licensee inspection," requires that: The licensee shall establish and execute a program for inspection of activities affecting quality...to verify conformance with the documented instructions, procedures, and drawings for accomplishing the activity. The inspection must be performed by individuals other than those who performed the activity being inspected. Examinations,

measurements, or tests of material or products processed must be performed for each work operation where necessary to assure quality....If mandatory inspection hold points...are required, the specific hold points must be indicated in appropriate documents."

In its application, PFS stated that it will establish a program for inspection of all activities affecting quality, whether performed by company or contractor personnel, to verify conformance with approved procedures, drawings, and specifications. Specifically, the applicant states that: (1) Approved procedures shall be implemented delineating inspection methods, characteristics, and documentation; (2) Inspections shall be performed by qualified personnel other than those who performed or supervised the work being inspected; and (3) Mandatory inspection hold points, which require witnessing or inspecting of an activity before processing, shall be indicated in the appropriate procedures or specifications.

The staff reviewed the QA Program and determined that PFS has established a program for inspection of activities affecting quality to verify conformance with the documented instructions, procedures, and drawings for accomplishing the activity. The staff determined that the program will assure that inspections are performed by individuals other than those who performed the activities being inspected; examinations, measurements, or tests of material or products processed will be performed for each work operation where necessary to assure quality; and if mandatory inspection hold points are required, the hold points will be indicated in appropriate documents. The staff, therefore, concludes that PFS's inspection control program, as described in the QA Program, satisfies the requirements of 10 CFR 72.160.

12.1.11 Test Control

The description of the PFS test control process was reviewed for conformance with the following requirements:

- 10 CFR 72.162, "Test control," requires that: "The licensee shall establish a test program to assure that all testing required...is identified and performed in accordance with written test procedures....The licensee shall document and evaluate the test results to assure that test requirements have been satisfied."

In its application, PFS stated that it will establish a test program to demonstrate that structures, systems, and components will perform satisfactorily in service. Specifically, the applicant states that: (1) Testing shall be performed in accordance with approved test procedures which incorporate or reference the requirements and acceptance criteria contained in applicable design documents and specifications; (2) Test results shall be documented and evaluated to assure that test requirements have been satisfied; and (3) Test results which fail to meet the requirements and acceptance criteria shall be properly noted and appropriate corrective action taken.

The staff reviewed the QA Program and determined that PFS established a test program to assure that all testing required is identified and performed in accordance with written test procedures. The staff determined that the test program assures that all required testing will be identified and documented; testing will be performed in accordance with approved test

procedures that incorporate or reference the requirements and acceptance criteria; test results will be documented and evaluated to assure that test requirements have been satisfied; and appropriate corrective action will be taken when test results fail to meet the test requirements and acceptance criteria. The staff, therefore, concludes that PFS's test control program, as described in the QA Program, satisfies the requirements of 10 CFR 72.162.

12.1.12 Control of Measuring and Test Equipment

The description of the PFS measuring and test equipment control process was reviewed for conformance with the following requirements:

- 10 CFR 72.164, "Control of measuring and test equipment," requires that: "The licensee shall establish measures to assure that tools, gauges, instruments, and other measuring and testing devices used in activities affecting quality are properly controlled, calibrated, and adjusted...."

In its application, PFS stated that it has established the requirements for the control, calibration, and periodic adjustment of tools, gauges, instruments, and other measuring and test equipment (IM&TE) used to verify conformance to requirements. Specifically, the applicant states that: (1) Inspection, test, and work procedures shall include provisions to assure that IM&TE used in activities affecting quality are of the proper range, type, and accuracy to verify conformance to established requirements and test parameters; and (2) To assure equipment accuracy, IM&TE shall be controlled, calibrated, adjusted, and maintained periodically, or prior to use.

The staff reviewed the QA Program and determined that PFS has established measures to assure that IM&TE used in activities affecting quality are properly controlled, calibrated, adjusted, and maintained to assure equipment accuracy. The staff determined that PFS has established measures to implement procedures that will assure that IM&TE used in activities affecting quality are of the proper range, type, and accuracy to verify conformance to requirements. The staff, therefore, concludes that PFS's control of measuring and test equipment, as described in the QA Program, satisfies the requirements of 10 CFR 72.164.

12.1.13 Handling, Storage, and Shipping

The description of the PFS handling, storage, and shipping control process was reviewed for conformance with the following requirements:

- 10 CFR 72.166, "Handling, storage, and shipping control," requires that: "The licensee shall establish measures to control...the handling, storage, shipping, cleaning, and preservation of materials and equipment to prevent damage or deterioration. When necessary for particular products, special protective environments, such as inert gas atmosphere, and specific moisture content and temperature levels must be specified and provided."

In its application, PFS stated that it will control the handling, storage, shipping, cleaning, packaging, and preservation of material and equipment to prevent damage, deterioration, or loss through shipment, installation, or use. Specifically, the applicant states that: (1) Approved

procedures and instructions shall be implemented delineating the requirements for handling, storage, shipping, cleaning, and preservation of materials and equipment; and (2) When required, procedures shall describe special equipment to be used, protective environments and coatings, or other protective measures.

The staff reviewed the QA Program and determined that PFS has established measures to control the handling, storage, shipping, cleaning, and preservation of materials and equipment to prevent damage or deterioration. The staff determined that PFS has established measures to specify in approved procedures and instructions the requirements for handling, storage, shipping, cleaning, and preservation of materials and equipment, and to specify special equipment to be used, protective environments, coatings, or other protective measures, and required documentation. The staff, therefore, concludes that PFS's control of handling, storage, and shipping, as described in the QA Program, satisfies the requirements of 10 CFR 72.166.

12.1.14 Inspection, Test, and Operating Status

The description of the PFS inspection, test, and operating status control process was reviewed for conformance with the following requirements:

- 10 CFR 72.168(a), "Inspection, test, and operating status," requires that: "The licensee shall establish measures to indicate, by the use of markings such as stamps, tags, labels, routing cards, or other suitable means, the status of inspections and tests performed upon individual items of the ISFSI....These measures must provide for the identification of items which have satisfactorily passed required inspections and tests where necessary to preclude inadvertent bypassing of the inspections and tests."

- 10 CFR 72.168(b) requires that: "The licensee shall establish measures to identify the operating status of structures, systems, and components of the ISFSI...such as tagging valves and switches to prevent inadvertent operation."

In its application, PFS stated that it will indicate the inspection, test, and operating status of components and systems. Specifically, the applicant states that: (1) The status of inspections and tests shall be indicated on the item to the extent possible, or in documents traceable to the item; (2) The status is identified by the use of tags, markings, stamps, or other means to assure required inspections or tests are not bypassed; (3) The operating status of systems and components shall be controlled through the use of tags secured to appropriate valves, switches, or control mechanisms.

The staff reviewed the QA Program and determined that PFS has established measures to indicate the status of inspections and tests performed upon individual structures, systems, and components to preclude inadvertent bypassing of the inspections and tests and to identify the operating status of structures, systems, and components. The staff determined that PFS has established measures to indicate the status of inspections and tests on items or in documents traceable to the items; use tags, markings, stamps, or other means to assure that required inspections and tests are not bypassed; and prevent the operation of equipment or systems using status indicators. The staff, therefore, concludes that PFS's inspection, test, and

operating status control program, as described in the QA Program, satisfies the requirements of 10 CFR 72.168.

12.1.15 Nonconforming Material, Parts, or Components

The description of the PFS nonconforming material, parts, and components control process was reviewed for conformance with the following requirements:

- 10 CFR 72.170, "Nonconforming material, parts, or components," requires that: "The licensee shall establish measures to control materials, parts, or components that do not conform to the licensee's requirements in order to prevent their inadvertent use or installation. These measures must include, as appropriate, procedures for identification, documentation, segregation, disposition, and notification to affected organizations. Nonconforming items must be reviewed and accepted, rejected, repaired, or reworked in accordance with documented procedures."

In its application, PFS stated that it will control materials, parts, or components which do not conform to requirements in order to prevent their inadvertent use or installation. Specifically, the applicant states that: (1) Approved procedures shall be implemented to provide requirements for identifying, segregating, and reporting discrepancies and dispositioning of non-conforming items as well as notification to affected organizations; (2) Materials, parts, or components which do not conform to requirements shall be identified and placed in a hold status; (3) Nonconforming items shall remain in a segregated area as appropriate until approved disposition has been received; (4) The disposition of nonconformances shall be evaluated and approved by appropriate personnel in accordance with approved procedures; and (5) Nonconformances shall be closed by qualified personnel in accordance with written procedures to include verification that the corrective action was adequate, complete, and documented appropriately.

The staff reviewed the QA Program and determined that PFS has establish measures to control materials, parts, or components that do not conform to the licensee's requirements in order to prevent their inadvertent use or installation. The staff determined that PFS has established measures to identify, segregate, and disposition non-conforming items and report discrepancies to affected organizations; segregate non-conforming items until properly dispositioned by appropriate personnel; and close nonconformances using qualified personnel to verify that corrective actions were adequate, complete, and documented. The staff, therefore, concludes that PFS's nonconformance control program, as described in the QA Program, satisfies the requirements of 10 CFR 72.170.

12.1.16 Corrective Action

The description of the PFS corrective action control process was reviewed for conformance with the following requirements:

- 10 CFR 72.172, "Corrective action," requires that: "The licensee shall establish measures to assure that conditions adverse to quality...are promptly identified

and corrected. In the case of a significant condition adverse to quality, the measures must assure that the cause of the condition is determined and corrective action is taken to preclude repetition. The identification of the significant condition adverse to quality, the cause of the condition, and the corrective action token must be documented ane reported to appropriate levels of management."

In its application, PFS stated that it will promptly identify and correct conditions adverse to quality, such as failures, malfunctions, deficiencies, deviations, defective material and equipment, and nonconformances. Specifically, the applicant states that: (1) Conditions adverse to quality, such as failures, malfunctions, deficiencies, deviations, and defective material and equipment, shall be identified and reported to appropriate personnel using approved procedures; (2) For significant conditions adverse to quality, the cause of the condition and corrective action necessary to prevent recurrence shall be identified, implemented and then followed-up to verify corrective action effectiveness using approved procedures; and (3) Appropriate levels of management will be notified of significant conditions adverse to quality and the disposition of these conditions.

The staff reviewed the QA Program and determined that PFS has established measures to assure that conditions adverse to quality are promptly identified and corrected. The staff determined that PFS has established measures to identify and report to appropriate personnel conditions adverse to quality; for significant conditions adverse to quality, identify the causes of the conditions, determine the corrective actions necessary to prevent recurrence of the conditions, and notify appropriate levels of management of the conditions and the dispositions of the conditions; and verify corrective action effectiveness. The staff, therefore, concludes that PFS's corrective action program, as described in the QA Program, satisfies the requirements of 10 CFR 72.172.

12.1.17 Quality Assurance Records

The description of the PFS QA records control process was reviewed for conformance with the following requirements:

- 10 CFR 72.174, "Quality assurance records," requires that: "The licensee shall maintain sufficient records to furnish evidence of activities affecting quality....Records must be identifiable and retrievable. Records pertaining to the design, fabrication, erection, testing, maintenance, and use of structures, systems, and components important to safety shall be maintained by or under the control of the licensee until the Commission terminates the license."

In its application, PFS stated that it will maintain records of activities affecting quality. Specifically, the applicant states that: (1) Approved procedures shall be developed and implemented to establish controls for the identification, receipt, storage, preservation, safekeeping, traceability, retrieval, and disposition of records; and (2) At a minimum, records pertaining to the design, fabrication, erection, testing, maintenance, and use of structures, systems, and components important to safety shall be maintained until termination of the license.

The staff reviewed the QA Program and determined that PFS has established measures to maintain sufficient records to furnish evidence of activities affecting quality. The staff determined that PFS has established measures to control the identification, receipt, storage, preservation, safekeeping, traceability, retrieval and disposition of records, and to maintain records pertaining to the design, fabrication, erection, testing, maintenance, and use of structures, systems, and components important to safety until termination of the license. The staff, therefore, concludes that PFS's QA records control program, as described in the QA Program, satisfies the requirements of 10 CFR 72.174.

12.1.18 Audits

The description of the PFS audit control process was reviewed for conformance with the following requirements:

- 10 CFR 72.176, "Audits," requires that: "The licensee shall carry out a comprehensive system of planned and periodic audits to verify compliance with all aspects of the quality assurance program and to determine the effectiveness of the program. The audits must be performed in accordance with written procedures or checklists by appropriately trained personnel not having direct responsibilities in the areas being audited. Audited results must be documented and reviewed by management having responsibility in the area audited. Follow-up action, including re-audit of deficient areas must be taken where indicated."

In its application, PFS stated that it will establish a system of planned and documented audits to verify compliance with all aspects of the QA Program and to assess the effectiveness of the program. Specifically, the applicant states that: (1) Audits shall be performed in accordance with written procedures or checklists by appropriately trained personnel having no direct responsibilities in the area audited; (2) Audit results shall be documented and reported to the management having responsibility in the area audited; and (3) Follow-up actions, including a re-audit, shall be performed to verify that corrective actions have been taken to correct the deficiencies or nonconformances.

The staff reviewed the QA Program and determined that PFS has established measures to carry out a comprehensive system of planned and periodic audits to verify compliance with all aspects of the QA program and to determine the effectiveness of the program. The staff determined that PFS has established measures to perform audits in accordance with written procedures or checklists using trained personnel having no direct responsibilities in the area audited; document and review audit results with management responsible for the area audited; and perform audits to verify that corrective actions have been taken to correct deficiencies and nonconformances. The staff, therefore, concludes that PFS's audit program, as described in the QA Program, satisfies the requirements of 10 CFR 72.174.

12.2 Evaluation Findings

Based upon its review and evaluation of the PFS QA Program described in the SAR, the staff concludes that the PFS QA Program:

- establishes and delineates the authority and duties of persons and organizations performing activities affecting quality;

- provides for indoctrination and training of personnel performing activities affecting quality;

- assures conformance to the approved design of structures, systems, and components;

- provides for approved procedures or instructions to document the QA program and requires that these procedures and instructions be followed;

- provides control over activities affecting quality to an extent commensurate with the importance to safety;

- defines and establishes requirements, processes, and controls that will comply with the requirements of 10 CFR 72, Subpart G;

- assures management review of the QA program to determine program status and adequacy; and

- satisfies the requirements of 10 CFR 72.40(a)(7).

Therefore, the staff has determined that, when properly implemented, the PFS QA Program is acceptable.

12.3 References

Private Fuel Storage Limited Liability Company. 1999. *Safety Analysis Report for Private Fuel Storage Facility*. Revision 3. Docket No. 72-22. La Crosse, WI: Private Fuel Storage Limited Liability Company.

Private Fuel Storage Limited Liability Company. 1996. *Private Fuel Storage Quality Assurance Program Description*. Docket No. 72-22. La Crosse, WI: Private Fuel Storage Limited Liability Company. August 30, 1996.

13 DECOMMISSIONING EVALUATION

13.1 Conduct of Review

The objective of the review is to determine whether the applicant's provisions for decommissioning the Facility provide reasonable assurance that decontamination and decommissioning of the Facility at the end of its useful life provide adequate protection to the health and safety of the public. The review considers information presented in the Preliminary Decommissioning Plan in Appendix B of the License Application as well as the Chapters 3, 6, 7, and 9 of the SAR.

13.1.1 Design Features

At this time, the staff cannot make a determination that the Facility structures, systems, and components have been adequately designed for decommissioning. This evaluation relies, in part, on cask-specific information.

13.1.2 Operational Features

At this time, the staff cannot make a determination that the Facility operational features have been adequately designed for decommissioning. This evaluation relies, in part, on cask-specific information.

13.1.3 Decommissioning Plan

Review of the Preliminary Decommissioning Plan included consideration of: (a) the overall adequacy and completeness of the Preliminary Decommissioning Plan, including proposed decontamination and decommissioning activities, (b) the decommissioning cost estimate, and (c) the financial assurance mechanism. The review considered how the Preliminary Decommissioning Plan and SAR addressed the following requirements of 10 CFR 72.30, Financial Assurance and Recordkeeping for Decommissioning:

- 10 CFR 72.30(a) requires that each application under this part include a proposed decommissioning plan that contains sufficient information on the proposed practices and procedures for the decontamination of the site and for disposal of residual radioactive materials after all spent fuel has been removed. This plan must also identify and discuss those design features of the ISFSI that facilitate decontamination and decommissioning.

- 10 CFR 72.30(b) requires that the proposed decommissioning plan also include a decommissioning funding plan.

- 10 CFR 72.30(c) requires that the financial assurance for decommissioning be provided.

The staff also notes that, in accordance with 10 CFR 72.30(d), PFS must keep records of information important to the decommissioning of the Facility.

Preliminary Decommissioning Plan, 10 CFR 72.30(a)

The applicant submitted a Preliminary Decommissioning Plan which describes the conceptual program for decontaminating and decommissioning the Facility. The plan states that "the objective of decommissioning activities for the Facility is to remove all radioactive materials having activities above the applicable NRC release limits in order that the site may be released for unrestricted use, and the NRC license terminated."

As part of meeting this objective, the applicant plans to implement measures that reduce the potential for contamination, thus, facilitating decontamination and decommissioning. The dual-purpose canisters arriving at the Facility are expected to have only minimal, if any, external surface contamination. A canister would not be permitted to be transported to the Facility if surveys performed at the originating power plant during loading indicate that the surface contamination levels exceed the acceptable limits for the specific canister design. A survey will be performed upon receipt of the canisters at the Facility, and canisters with surface contamination levels above acceptable limits will be returned to the originator. Further, the canisters which are sealed by welding will not be opened at the Facility. Thus, the possibility of the canister contaminating the Canister Transfer Building, the canister transfer equipment, the transfer cask, the storage casks, and storage pad is minimized. Nevertheless, the interior concrete surfaces of the Canister Transfer Building will be coated with paint or epoxy, which is non-porous and can be easily decontaminated by wiping.

The Preliminary Decommissioning Plan also provides a general discussion of decommissioning tasks. Before decommissioning activities begin, the spent fuel canisters will be shipped off-site using a cask approved under 10 CFR Part 71. Facility structures and components, including storage casks, will be decontaminated to the extent practicable by conventional methods such as wiping or stripping of paint. Structures or components with residual contamination or activation levels above regulatory limits will be packaged and disposed of as low-level waste. Radioactive waste generated during decontamination will also be package and disposed of as low-level waste. A radiological survey of the Facility will be performed, with particular attention to areas of known or historic contamination. A final radiation survey will be conducted to verify that any radioactivity at the site is below the applicable NRC limits such that the site can be released for unrestricted use.

The staff has reviewed the Preliminary Decommissioning Plan. The staff finds that the Preliminary Decommissioning Plan includes sufficient discussion of the applicant's proposed practices and procedures for minimizing contamination at the Facility. The plan also includes sufficient discussion of the applicant's conceptual program for decommissioning the facility. However, the staff has made no findings as to whether the Preliminary Decommissioning Plan sufficiently identifies and discusses the design and operational features of the ISFSI that facilitate its decontamination and decommissioning. That determination relies, in part, on cask-specific information which is not being evaluated at this time. Thus, at this time, the staff has not determined that the Preliminary Decommissioning Plan satisfies 10 CFR 72.30(a).

Financial Assurance, 10 CFR 72.30(b) and 72.30(c)

The applicant's financial qualifications are evaluated in Chapter 17 of this SER. The objective of that evaluation is to determine compliance with the decommissioning funding and financial assurance requirements of 10 CFR 72.30(b) and 72.30(c).

Records of Information Important to Decommissioning, 10 CFR 72.30(d)

The Preliminary Decommissioning Plan includes a commitment by the applicant to maintain the following records that are identified by 10 CFR 72.30(d) as important to decommissioning:

- records of spills or other unusual occurrences involving the spread of contamination [required by 10 CFR 72.30(d)(1)];
- as-built drawings and modifications of structures and equipment in restricted areas [required by 10 CFR 72.30(d)(2)];
- a document, which is updated a minimum of every 2 years, listing all areas designated at any time as restricted areas and all areas outside of restricted areas involved in a spread of contamination, [required by 10 CFR 72.30(d)(3)]; and
- records of the cost estimate performed for the decommissioning funding plan [as required by 10 CFR 72.30(d)(4)].

13.2 Evaluation Findings

The staff has not made the determination that the Preliminary Decommissioning Plan satisfies the requirements of 10 CFR 72.30(a). Specifically, the staff cannot conclude that the Preliminary Decommissioning Plan sufficiently identifies and discusses those design and operational features of the proposed cask system that facilitate decommissioning. That determination relies upon cask-specific information which is not being evaluated at this time. A final determination regarding the adequacy of the plan will be made at the time that the staff issues its final SER which will include consideration of at least one dry cask storage system.

13.3 References

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, and May 19, August 10, August 27, and September 8, and September 21, 1999.

Private Fuel Storage Limited Liability Company. 1999. *Safety Analysis Report for Private Fuel Storage Facility.* Revision 3. Docket No. 72-22. La Crosse, WI: Private Fuel Storage Limited Liability Company.

14 WASTE CONFINEMENT AND MANAGEMENT EVALUATION

14.1 Conduct of Review

This chapter of the SER evaluates the waste management systems of the Facility. Chapter 6 of the SAR provides information about the waste confinement and disposal systems that are part of the Facility. The review objectives for this chapter are to establish that the Facility provides safe confinement and management of radioactive waste generated at the Facility, and the generation of radioactive waste and release of the radioactive material to the environment meet the regulatory standards.

14.1.1 Waste Sources

Review of the sources of radioactive waste described in Section 6.1 of the SAR included consideration of dry solid radioactive waste produced during operation of the Facility. The review considered how the SAR addresses 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 beyond the controlled area must not exceed 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 organ, from various sources, including planned discharges of radioactive materials to the environment.

- 10 CFR 72.122(b)(4) requires that, for facilities located over an aquifer, measures be taken to preclude transport of radioactive materials to the environment through this pathway.

- 10 CFR 72.128(a)(5) requires that systems for handling radioactive materials in the ISFSI must be designed to minimize the quantity of radioactive wastes generated.

The SAR describes the radioactive waste sources, which are limited to those produced during the operation of the Facility. The solid waste will consist of anti-contamination clothing, rags, and associated health physics materials. This solid waste will be packaged and temporarily stored in the low-level waste holding cell at the Canister Transfer Building and then will be transferred to an offsite low-level waste disposal facility. The dual-purpose canisters arriving at the Facility are expected to have only minimal, if any, external surface contamination. A canister would not be permitted to be transported to the Facility if surveys performed at the originating power plant during loading indicated that the surface contamination levels exceed the acceptable limits for the specific canister design. A canister would not be accepted at the Facility if surveys conducted upon receipt indicated surface contaminations exceed the acceptable limits. The canisters, which are sealed by welding, would not be opened at the Facility. The welded canisters must be designed such that they are not breached under normal and off-normal conditions of transfer, handling, and storage. Therefore, no release of radioactive materials from inside the canister is expected under these conditions.

The staff finds that the SAR adequately describes waste sources and that there are no routine effluents discharged to the environment due to the operation of the Facility including normal and off-normal conditions. Since there are no liquid or gaseous effluents, the staff finds that the requirement of 10 CFR 72.122(b)(4) regarding precluding transport of radioactivity to an aquifer is met, and that the dose limits of 10 CFR 72.104(a) are met with respect to release of effluents. The generation of radioactive waste is limited because the surface contamination on canisters is limited, the canisters are welded closed, and the canisters are not opened at the Facility. Therefore, the staff has determined that the requirements of 10 CFR 72.128(a)(5) are met in that the design of the Facility systems will minimize the generation of radioactive waste.

14.1.2 Off-Gas Treatment and Ventilation

The review of Section 6.2 of the SAR regarding off-gas treatment and ventilation considered how the SAR addresses the following regulatory requirements:

- 10 CFR 20.2001 authorizes a licensee to dispose of radioactive materials only by certain methods, including transfer to an authorized recipient, and by limited release in effluents.

- 10 CFR 72.104(a) requires that, during normal operations and anticipated occurrences, the annual dose equivalent to any real individual beyond the controlled area must not exceed 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 organ, from various sources, including planned discharges of radioactive materials to the environment.

- 10 CFR 72.104(b) requires that operational restrictions be established to meet ALARA objectives for radioactive materials in effluents.

- 10 CFR 72.104(c) requires that operational limits for radioactive materials in effluents be established to ensure that the dose limits in 10 CFR 72.104(a) are met.

- 10 CFR 72.122(h)(3) requires that ventilation systems and off-gas systems must be provided where necessary to ensure confinement of airborne radioactive particles during normal or off-normal conditions.

- 10 CFR 72.126(c)(1) requires that, as appropriate for the handling and storage system, effluent systems must be provided, as well as methods for measuring the amount of radionuclides in the effluents.

- 10 CFR 72.126(d) requires that the ISFSI must be designed to limit effluents to ALARA levels.

- 10 CFR 72.128(a)(5) requires that systems for handling radioactive materials in the ISFSI must be designed to minimize the quantity of radioactive wastes generated.

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

According to the SAR, there will be no gaseous release from the storage systems at the Facility, as the Facility will only handle sealed canisters. The staff finds that under normal operations, no radioactive materials will be released to the environment as gaseous effluents. Because there are no gaseous effluents, no special off-gas or ventilation systems are needed, and the requirements of 10 CFR 72.122(h)(3), 72.126(c)(1), 72.126(d), 72.128(a)(5), and 72.128(b) are met. Since there are no gaseous effluents, the staff finds that the design and operation of the ISFSI meet 10 CFR 72.104(a), (b), and (c) with regard to off-site doses from effluents.

14.1.3 Liquid Waste Treatment and Retention

The review of Section 6.3 of the SAR regarding liquid waste treatment and retention considered how the SAR addresses the following regulatory requirements:

- 10 CFR 20.2001 authorizes a licensee to dispose of radioactive materials only by certain methods, including transfer to an authorized recipient, and by limited release in effluents.

- 10 CFR 20.2003 authorizes a licensee to dispose of radioactive materials by discharge into sanitary sewerage, with certain restrictions.

- 10 CFR 72.104(a) requires that, during normal operations and anticipated occurrences, the annual dose equivalent to any real individual beyond the controlled area must not exceed 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 organ, from various sources, including planned discharges of radioactive materials to the environment

- 10 CFR 72.104(b) requires that operational restrictions be established to meet ALARA objectives for radioactive materials in effluents.

- 10 CFR 72.104(c) requires that operational limits for radioactive materials in effluents be established to ensure that the dose limits in 72.104(a) are met.

- 10 CFR 72.122(b)(4) requires that, for facilities located over an aquifer, measures be taken to preclude transport of radioactive materials to the environment through this pathway.

- 10 CFR 72.126(c)(1) requires that, as appropriate for the handling and storage system, effluent systems must be provided, as well as methods for measuring the amount of radionuclides in the effluents.

- 10 CFR 72.126(d) requires that the ISFSI must be designed to limit effluents to ALARA levels.

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

According to the SAR, liquid wastes will not be routinely generated at the Facility under normal operations. Drain sumps are provided in the Canister Transfer Building to catch and collect water that may drip from the shipping cask. The collected water will be sampled and analyzed for radioactive contamination prior to release. If the water is found to be contaminated, it will be collected in a suitable container and then solidified by a suitable agent so that it qualifies as solid waste. The solidified waste will be stored temporarily in the low-level waste holding cell and then transported to the offsite low-level waste disposal facility.

The staff finds that there are no special liquid radioactive waste treatment and retention systems needed at the Facility. The applicant has identified and described an appropriate method for treating contaminated liquids, should it be needed, and therefore the staff finds that the requirements of 10 CFR 72.128(b) are met for contaminated liquids. There are no liquid radioactive effluents that will be discharged to the environment under normal operations. Because there are no liquid effluents, the requirements of 10 CFR 20.2001, 20.2003, 72.122(b)(4), and 72.126(c)(1) and (d) are met with respect to possible release of radioactive material in liquid effluents. Since there are no liquid effluents, the staff finds that the design and operation of the ISFSI meet 10 CFR 72.104(a), (b), and (c) with regard to doses from liquid effluents. The proposed method of handling contaminated liquid meets the requirements of 20.2001 for off-site disposal.

14.1.4 Solid Wastes

Review of the handling of solid wastes described in Section 6.4 of the SAR included the description of collection, packaging, and storage of solid wastes. The review considered how the SAR addresses 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 beyond the controlled area must not exceed 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 organ, from various sources, including planned discharges of radioactive materials to the environment.

- 10 CFR 72.104(b) requires that operational restrictions be established to meet ALARA objectives for radioactive materials in effluents.

- 10 CFR 72.104(c) requires that operational limits for radioactive materials in effluents be established to ensure that the dose limits in 10 CFR 72.104(a) are met.

- 10 CFR 72.122(h)(3) requires that ventilation systems and off-gas systems must be provided where necessary to ensure confinement of airborne radioactive particles during normal or off-normal conditions.

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

The SAR included a description of collection, packaging, and storage of solid wastes. The solid waste at the Facility will be generated from decontamination activities. A small amount of solid waste will be generated in the form of smears, disposable clothing, tape, blotter paper, rags, and related health physics material. The solid waste will be identified and packaged in suitable low-level waste containers. The solid waste will be temporarily stored at the low-level waste holding cell at the Canister Transfer Building and will then be transferred to an offsite low-level waste disposal facility. Because the solid waste will typically be generated due to an off-normal or accident event, the volume of solid waste is expected to be minimal.

The staff agrees that the provisions for handling solid wastes are appropriate and meet the requirements of 10 CFR 72.128(b). The method described would not be expected to produce radioactive effluents, and therefore meets the requirements of 10 CFR 72.104(a), (b), and (c) with respect to doses from effluents, and meets 10 CFR 72.122(h)(3) with respect to release of effluents.

14.1.5 Radiological Impact of Normal Operations

Review of the summary of radiological impacts of normal operations in Section 6.5 of the SAR considered how the SAR addresses the following regulatory requirements:

- 10 CFR 20.1101 requires that a licensee, as part of the radiation protection program, establish a constraint for air emissions of radioactive materials to the environment such that a member of the public is not expected to receive a total effective dose equivalent in excess of 10 mrem (0.1 mSv) per year.

- 10 CFR 20.1301(a) establishes dose limits for a member of the public, including a total effective dose equivalent of 0.1 rem (1 mSv) in a year, and a maximum dose in any unrestricted areas of 0.002 rem (0.02 mSv) in an hour.

- 10 CFR 20.1301(b) requires that the licensee comply with the environmental radiation standards in 40 CFR Part 190.

- 10 CFR 20.1302(b) requires that the licensee show compliance with the limits in 10 CFR 20.1301, by either demonstrating compliance with the dose limit to an individual by calculation or measurement, or by demonstrating that radioactivity in gaseous and liquid effluents to do not exceed the values in table 2 of Appendix B to Part 20, and the dose from external sources would not exceed 0.002 rem (0.02 mSv) in an hour and 0.05 rem (0.5 mSv) in a year.

- 10 CFR 72.40(a)(13)(i) states that the Commission will issue a license under 10 CFR Part 72 upon a determination that the application for a license meets the standards and requirements of the Act and the regulations of the Commission, and upon finding that, among other things, the activities authorized by the license can be conducted without endangering the health and safety of the public.

The SAR included a summary of radiological impact of normal operations. Under all normal and off-normal conditions of transfer, handling, and storage, the welded canisters will remain sealed and no radioactive material will be released from inside the canister. Additionally, the practices and procedures that have been proposed to limit and control contamination at the Facility will assure that radiological impacts are minimized and that ALARA principles are maintained. No release of radioactive material to the environment is expected during normal Facility operations and no liquid or gaseous effluents are anticipated from the Facility. The radiological impacts to the environment from the normal operations at the Facility will be minimal. The staff has determined that the radiological impact of the Facility under normal operations has been adequately and appropriately described, and that the radiological impacts from releases will be minimal and will not endanger the health and safety of the public. Based on these considerations, the staff has determined that the requirements of 10 CFR 20.1101 have been met with respect to potential releases of radioactive materials under normal operations, and that 10 CFR 72.40(a)(13)(i), 20.1301 and 20.1302 have been met with respect to doses to members of the public from potential releases of radioactive materials under normal operations. The dose to members of the public due to direct radiation, and other sources, is cask-specific and will be evaluated in the final SER.

14.2 Evaluation Findings

The staff has reviewed Sections 6.1 through 6.5 of the SAR and has determined that the waste confinement and management of the proposed facility:

- meet the requirements of 10 CFR 20.2001, 20.2003, 72.104(b), 72.104(c), 72.122(b)(4), 72.122(h)(3), 72.126(c)(1), 72.126(d) with respect to the potential release of effluents to the environment,

- meet the radioactive waste management and minimization requirements of 10 CFR 72.128(a)(5) and 72.128(b), and

- meet the dose limits for members of the public of 10 CFR 20.1101, 20.1301, 20.1302, and 72.104(a) with respect to radioactive materials released as effluents. The dose to members of the public due to direct radiation, and other sources, is cask-specific and will be evaluated in the final SER.

- will not endanger health and safety of the public as required by 10 CFR 72.40, with respect to radioactive materials released as effluents.

14.3 References

Private Fuel Storage Limited Liability Company. 1999. *Safety Analysis Report for Private Fuel Storage Facility.* Revision 3. Docket No. 72-22. La Crosse, WI: Private Fuel Storage Limited Liability Company.

15 ACCIDENT ANALYSIS

The accident analysis relies, in part, on cask-specific or cask-dependent information. In addition, the applicant has not yet provided sufficient information for the staff to complete the site-specific aspects of the review. Therefore, the staff cannot make a determination regarding the adequacy of the accident analysis for the proposed Facility.

16 EMERGENCY PLAN

16.1 Conduct of Review

This chapter of the SER evaluates the Emergency Plan submitted by PFS in support of the application to license the Facility at the Skull Valley Reservation in Tooele County, Utah. This safety evaluation is based on Revision 6 to the PFS Facility Emergency Plan.

Section 72.32(a), *Emergency Plan*, of 10 CFR Part 72 provides the regulatory requirements for ISFSI emergency plans. Section 72.40(a)(11) requires that for the issuance of a license, the applicant's emergency plan must comply with 10 CFR 72.32. The draft NUREG-1567 *Standard Review Plan for Spent Fuel Dry Storage Facilities,* October 1996, provides guidance for staff reviewers.

16.1.1 Facility Description

10 CFR 72.32(a)(1) requires a brief description of the licensee's facility and area near the site. The Emergency Plan describes the Facility and site and provides detailed maps of the site, including the cask storage area and important supporting structures. Detailed maps and descriptions of the area adjacent to the site and the area near the site are provided. The Emergency Plan, therefore, meets the requirements of 10 CFR 72.32(a)(1).

16.1.2 Types of Accidents

10 CFR 72.32(a)(2) requires the identification of each type of radioactive materials accident. The Emergency Plan describes types of accidents that could result in the release of radioactive material, the processes and physical locations where they could occur and how they could occur. The possible on site and off site consequences of potential accidents are discussed. The Emergency Plan describes the potential accidents that could occur at the facility. The Emergency Plan, therefore, meets requirements of 10 CFR 72.32(a)(2).

16.1.3 Classification of Accidents

10 CFR 72.32(a)(3) requires a classification system for classifying accidents as "alerts." As stated above, the applicant identified a list of potential radiological accidents. Additionally, NUREG-1567 provides guidance on the types of events that the Emergency Plan may consider. The Emergency Plan provides a classification system, based on emergency action levels, that classifies the accidents identified by the applicant in accordance with the guidance of NUREG-1567. The Emergency Plan provides emergency action levels that specifically characterize the occurrence of accidents that warrant the declaration of an alert. The Emergency Plan, therefore, meets the requirements of 10 CFR 72.32(a)(3).

16.1.4 Detection of Accidents

10 CFR 72.32(a)(4) requires that the means for detecting accident conditions be described. The Emergency Plan provides a complete description of the means for detecting accident conditions that is applicable to each of the potential accidents that were identified through compliance with

10 CFR 72.32(a)(2). The Emergency Plan, therefore, meets the requirements of 10 CFR 72.32.(a)(4). The staff notes that the installation or deployment of instrumentation and equipment, the training of staff in its use and the training of staff in the reporting of emergency conditions will be inspected prior to operation.

16.1.5 Mitigation of Consequences

10 CFR 72.32(a)(5) requires that the Emergency Plan briefly describe the means and equipment provided to mitigate the consequences of the accidents identified in the Emergency Plan. The mitigation of consequences must also be described in terms of protection of workers and a description of the program to maintain equipment must be provided. NUREG-1567 provides guidance on limiting actions performed by installed systems and trained site personnel, appropriate protective measures for site personnel and necessary types of protective facilities and equipment.

The Emergency Plan describes the equipment to be installed and design features that mitigate emergency events. The Emergency Plan describes actions to be taken by trained site personnel to mitigate emergency events. The Emergency Plan describes actions, including radiological protective actions, to be taken to protect onsite personnel. The Emergency Plan describes arrangements made for first aid, medical and hospital services. The Emergency Plan describes response equipment, facilities and communications equipment that will be available to support mitigation efforts. The types of equipment and the locations of the equipment are described. Provisions to inventory and test equipment are described.

The Emergency Plan, therefore, meets the requirements of 10 CFR 72.32(a)(5). The staff notes that the installation of equipment, procedures for the use of equipment, procedures for personnel mitigative actions and the training of personnel will be inspected prior to operation.

16.1.6 Assessment of Releases

10 CFR 72.32(a)(6) requires that Emergency Plan contain a brief description of the methods and equipment that will be used to assess releases of radioactive material. The Emergency Plan describes radiological sampling and monitoring methods that will be used to assess the extent of radioactive releases. The Emergency Plan describes the instrumentation and equipment that will be used by trained personnel to assess the extent of radioactive releases. The Emergency Plan identifies the personnel who will be trained and qualified in the methods and the use of instrumentation and equipment for the assessment of radioactive releases. The Emergency Plan, therefore, meets the requirements of 10 CFR 72.32(a)(6).

16.1.7 Responsibilities

10 CFR 72.32(a)(7) requires that the Emergency Plan provide a brief description of the responsibilities of personnel should an accident occur, including identification of personnel responsible to promptly notify offsite response personnel and the NRC. Personnel responsible for developing, maintaining and updating the Emergency Plan are also to be identified.

The Emergency Plan describes the normal site organization and identifies personnel responsible for maintaining and updating the Emergency Plan, implementing procedures and Emergency Plan related records. The Emergency Plan identifies the personnel who are responsible for ensuring that offsite notifications are performed promptly. The Emergency Plan describes the emergency response organization and the responsibilities and authority of key positions within it. Personnel with the responsibility to declare emergencies during normal and off normal hours are identified. The Emergency Plan identifies the communications chain for notifying and mobilizing emergency response personnel during normal and non-working hours. The Emergency Plan describes methods for activation of the staff necessary for Emergency Plan implementation. The personnel responsible for overall direction of emergency response and notification of State and local agencies, as well as NRC are identified for normal and off normal hours.

The Emergency Plan, therefore, meets the requirements of 10 CFR 72.32(a)(7).

16.1.8 Notification of Coordination

10 CFR 72.32(a)(8) requires that the Emergency Plan contain a commitment to and a brief description of the means to promptly notify offsite response organizations; that a control point be established; that notification and coordination be planned; that unavailability of some personnel, parts of the facility and some equipment will not prevent notification and coordination. The licensee must also commit to notify the NRC immediately after notification of the appropriate offsite response organizations and not later than one hour after the licensee declares an emergency.

The Emergency Plan contains a commitment to promptly notify offsite response organizations. The Emergency Plan describes the means to notify offsite response organizations, the means to request offsite assistance, including medical assistance and the identification of the personnel responsible to perform the notifications. The organization described in the Emergency Plan is responsible to activate the emergency response organization and perform notifications in a timely manner under accident conditions, during normal and off normal hours. Diverse methods of notification are described. Facilities with notifications equipment are described as spatially separated. These features allow notification and activation to be performed even if some personnel, equipment and/or parts of the facility are unavailable. The Emergency Plan contains a commitment to notify NRC after completion of local notifications, but not later than one hour after the alert has been declared. The Emergency Plan, therefore, meets the requirements of 10 CFR 72.32(a)(8).

16.1.9 Information to be Communicated

10 CFR 72.32(a)(9) requires that the Emergency Plan provide a brief description of the types of information on facility status, radioactive releases and recommended protective actions, if necessary, to be given to offsite response organizations and the NRC. NUREG-1567 provides further guidance on the types of information that should be communicated.

The Emergency Plan provides a description of the minimum information which will be communicated to offsite response organizations and the NRC in the event of an emergency,

which is in compliance with the information required by 10 CFR 72.32(a)(9). This information is consistent with NUREG-1567 guidance. The responsible offsite agencies are listed in the Emergency Plan. The Emergency Plan, therefore, meets the requirements of 10 CFR 72.32(a)(9).

16.1.10 Training

10 CFR 72.32(a)(10) requires that the Emergency Plan describe, briefly, the training the licensee will provide to workers on how to respond to an emergency and any special instructions and orientation tours that will be offered to fire, police, medical and other emergency personnel.

The Emergency Plan describes the program to train facility personnel on how to respond to an emergency. The Emergency Plan describes the training program that will be offered to offsite support agency personnel who may be called upon to provide support to the facility. The Emergency Plan, therefore, meets the requirements of 10 CFR 72.32(a)(10). The staff notes that prior to operation, site inspection will verify that appropriate personnel have attended training and that the content of training is in accordance with the commitments of the Emergency Plan.

16.1.11 Safe Condition

10 CFR 72.32(a)(11) requires that the Emergency Plan provide a brief description of means for restoring the facility to safe operation after an accident. NUREG-1567 provides guidance suggesting that this description include a commitment to ensure all equipment important to safety has been restored to a state of readiness.

The Emergency Plan contains a description of the means for restoring the facility to safe operation after an accident, a commitment to develop written recovery procedures should such evolutions be necessary. The Emergency Plan contains criteria for the return to operations and a commitment to ensure that equipment important to safety has been checked and restored to normal operation before the facility is returned to operation. The Emergency Plan, therefore, meets the requirements of 10 CFR 72.32(a)(11).

16.1.12 Exercises

10 CFR 72.32(a)(12) requires that the Emergency Plan provides a description of the program for conduct of exercises, drills and communications tests.

The Emergency Plan describes the drill and exercise program. The program includes: biennial exercises, annual radiological drills, annual medical drills, and annual fire drills. The Emergency Plan states that offsite response organizations will be invited to participle in the biennial exercise. The Emergency Plan commits to check communications equipment semiannually, including the check and update of all necessary phone numbers. The Emergency Plan describes the evaluation of drills and correction of identified deficiencies and the confidentiality of exercise scenarios. The Emergency Plan identifies the Emergency Preparedness Coordinator as responsible to determine and implement corrective actions in response to identified deficiencies. The Emergency Plan, therefore, meets the requirements of 10 CFR

72.32(a)(12). The staff notes that prior to operation, site inspection will verify that drills have been conducted in accordance with the Emergency Plan. The staff expects that prior to operation, the licensee will demonstrate, in a biennial exercise, that PFS personnel can implement the Emergency Plan.

16.1.13 Hazardous Chemicals

10 CFR 72.32(a)(13) requires that the Emergency Plan certify that the licensee has met its responsibilities under the Emergency Planning and Community Right-to-know Act of 1986, Title III, Pub. L. 99-499.

The Emergency Plan states that there will be no hazardous substances on site in excess of the threshold planning quantities stipulated by the Emergency Planning and Community Right-to-Know Act of 1986, Title III, Pub. L. 99-499. Therefore, planning for hazardous chemicals is not required to be addressed in the Emergency Plan because this Act does not apply. The Emergency Plan states that the requirements of the Act have been met. The Emergency Plan, therefore, meets the requirements of 10 CFR 72.32(a)(13).

16.1.14 Comments on the Emergency Plan

10 CFR 72.32(a)(14) requires that the licensee allow offsite response organizations expected to respond in case of an accident 60 days to comment on the initial submission of the Emergency Plan before submitting it to the NRC and provide any comments received to the NRC.

The Emergency Plan contains copies of a letter documenting that the Emergency Plan has been reviewed by the applicable offsite response organization. The Emergency Plan contains the applicant's response to said comments. The Emergency Plan commits to providing Emergency Plan revisions to the offsite response organization. The Emergency Plan, therefore, meets the requirements of 10 CFR 72.32(a)(14).

16.1.15 Offsite Assistance

10 CFR 72.32(a)(15) requires that the Emergency Plan briefly describe arrangements for requesting and effectively using offsite assistance.

The Emergency Plan describes the means for requesting assistance from offsite response organizations when necessary. The Emergency Plan identifies offsite response organizations expected to provide support. The Emergency Plan states that assistance agreements will be documented in letters of agreement that will be reviewed annually and renewed every five years. The Emergency Plan describes the training that will be offered to offsite response organizations. The Emergency Plan, therefore, meets the requirements of 10 CFR 72.32(a)(15).

16.2 Evaluation Findings

10 CFR 72.40(a)(11) requires that the applicant's Emergency Plan comply with 10 CFR 72.32. The Emergency Plan submitted in support of the application to license the facility meets the requirements of 10 CFR 72.32(a). By virtue of this Emergency Plan the content of the

application meets the requirements of 10 CFR 72.24(k). The staff has concluded that the Private Fuel Storage Facility Emergency Plan, Revision 6, meets applicable regulations and guidance and is adequate.

16.3 References

Private Fuel Storage Limited Liability Company. *Emergency Plan for the Private Fuel Storage Facility, Revision 6.* Docket Number 72-22. September 8, 1999.

17 FINANCIAL QUALIFICATIONS AND DECOMMISSIONING FUNDING ASSURANCE

17.1 Conduct of Review

17.1.1 Background

Private Fuel Storage L.L.C. is a United States limited liability company owned by eight member companies (members or owners), which is organized under the laws of the State of Delaware and headquartered in La Crosse, Wisconsin. PFS is registered and authorized to transact business in the State of Utah, where it plans to construct, operate, and decommission an ISFSI to store spent fuel from U.S. nuclear power plants, including fuel from its members. These eight members are: Consolidated Edison Company; Genoa Fuel Tech, Inc., an affiliate of Dairyland Power Cooperative; GPU Nuclear Corporation; Illinois Power Company; Indiana Michigan Power Company; Northern States Power Company; Southern California Edison Company; and Southern Nuclear Operating Company, Inc.

In various proprietary documents sent to the NRC supplementing the PFS License Application, PFS has provided details pertaining to the legal, financial, and organizational relationships among its members, as well as financial estimates of various components of expected costs by year. These documents include the PFS Amended and Restated Limited Liability Company Agreement (PFS Agreement) and the PFS Business Plan.

The Facility is designed for a capacity of 40,000 MTU, which will require about 4,000 storage casks and about 500 pads, each pad being capable of supporting eight casks. Each cask will house one sealed metal canister containing multiple spent fuel assemblies. The Facility is designed to store spent fuel for up to 40 years, by which time it is anticipated that the spent fuel will have been transferred offsite so that the Facility can be decommissioned. The initial license request is for a term of 20 years, with plans to renew the license for another 20 years.

With respect to the NRC's financial qualifications requirements, under 10 CFR 72.22(e), an applicant for an ISFSI license must submit sufficient information to demonstrate its financial qualifications to carry out the activities for which the license is sought, in accordance with 10 CFR Part 72 regulations. The information must show "that the applicant either possesses the necessary funds, or that the applicant has reasonable assurance of obtaining the necessary funds, or that by a combination of the two, the applicant will have the necessary funds available to cover the following:

(1) estimated construction costs;

(2) estimated operating costs over the planned life of the ISFSI; and

(3) estimated decommissioning costs, and the necessary financial arrangements to provide reasonable assurance prior to licensing that decommissioning will be carried out after the removal of spent fuel and/or high level radioactive waste from storage."

Regarding decommissioning and decommissioning funding assurance, under 10 CFR 72.30(a), an applicant must provide a proposed decommissioning plan that describes its proposed practices and procedures for decontamination and decommissioning of the site. Further, under 10 CFR 72.30(b), an applicant must submit a "decommissioning funding plan containing information on how reasonable assurance will be provided that funds will be available to decommission the ISFSI." Furthermore, this information "must include a cost estimate for decommissioning and a description of the method of assuring funds for decommissioning from [10 CFR 72.30(c)] including means of adjusting cost estimates and associated funding levels periodically over the life of the ISFSI."

The staff also took into consideration the Commission's ruling in Louisiana Energy Services, L.P. (Claiborne Enrichment Center), CLI-97-15, 46 NRC 294 (1997), which pertains to an application by Louisiana Energy Services (LES) to construct and operate a uranium enrichment facility pursuant to 10 CFR Part 70. Among other things, the ruling held that "the NRC is not required as a matter of law to apply the strict financial qualification provisions of Part 50 to all Part 70 license applications." *Id.*, 46 NRC at 298. Rather, "Part 70 calls for a case-by-case inquiry into whether the applicant 'appears to be financially qualified' to take safety measures necessary to assure that activities under the license will not create undue risk to public health and safety." *Id.* at 299. The Commission further observed that Part 50, which applies to nuclear reactors, requires a demonstration at the construction permit stage that the applicant "possesses or has reasonable assurance of obtaining the funds necessary to cover estimated construction costs and related fuel cycle costs," and that the Part 50 financial assurance requirements are "far more detailed and comprehensive" than the "general language" found in Part 70 -- which indicates that a Part 70 license may be issued if the applicant "appears to be financially qualified." *Id.* The Commission further observed that the shorter, more flexible language in Part 70 allows "a less rigid, more individualized approach" to determine whether an applicant has demonstrated its financial qualifications, and stated that if the Commission "had intended the Part 50 standards and criteria to apply to all Part 70 applicants . . . the regulations would have either restated the Part 50 criteria or incorporated them by reference." *Id.* at 300. In sum, the Commission concluded that "the general language of Part 70 leaves the Commission free to review the reasonableness of an applicant's financial plan in light of all relevant circumstances," which might or might not lead to application of any or all of the criteria stated in Part 50. *Id.* at 302.

In considering the "relevant circumstances" present in the LES application, the Commission observed that LES lacked contractual commitments by its partners to fund any portion of the project, and also lacked agreements by lending institutions to fund any portion of the project. Nonetheless, the Commission took notice of commitments made by LES in the proceeding not to proceed with the project until certain funding commitments were in hand. Specifically, the Commission found that LES made a financial commitment of not constructing the proposed project in the absence of sufficient advance funding commitments (30% equity and 70% debt) to cover the project's cost, and sufficient advance purchase contracts for the plant's output to cover the construction and operating costs incurred during the term of the contract, including a return on investment. *Id.* at 304-05. The Commission relied on these commitments in developing and imposing two financial assurance conditions in its Order approving the LES application.

The PFS application for an ISFSI under Part 72 has some significant similarities to the LES Part 70 application, such as the fact that it is for a new, joint venture-type entity, made up of significant, financially secure corporations; it requests approval of a non-Part 50 facility application that has less health and safety risks than is associated with the operation of nuclear reactors; the application is not strictly subject to the Part 50 financial assurance requirements; and the applicant has made financial commitments that it will not proceed with construction of the Facility in the absence of sufficient advance funding commitments. The staff has considered such similarities in this review of the PFS application and in recommending herein certain financial assurance license conditions that the staff believes should be part of any determination to approve it. While Part 72 contains language that differs from Part 70, it is also less prescriptive than Part 50. *Compare* 10 CFR 72.22(e) *with* 10 CFR 50.33(f). Accordingly, as in the LES decision, the staff did not find it necessary or appropriate to rely on Part 50 standards and criteria for its review of the PFS application.

17.1.2 Financial Assurance for Construction Funding

PFS estimates costs of about $10 million for design and licensing and about $92 million for Facility construction. Key construction phase components include: site preparation; access road construction; building and storage pad construction; procurement of canister transfer and transport equipment; and transportation corridor (rail line) construction from the main rail line to the Facility site. PFS provided cost estimates of key components of each of the major phases of construction in a response to an NRC RAI, which the staff has reviewed and found to be adequate. These estimates are not shown in this SER, however, since they are proprietary.

Construction is to be funded through several mechanisms, with a total of $6 million expected from equity contributions from PFS members pursuant to Subscription Agreements and the remainder from revenue commitments from Service Agreements with member and nonmember Customers. If the combination of equity and revenue are insufficient to complete construction, PFS plans to finance the remainder through committed sources of debt financing. The License Application states that no construction will proceed unless and until Service Agreements for a significant commitment of fuel storage have been signed.

PFS plans to execute the Service Agreements referred to above with member and nonmember Customers after the granting of a license by the NRC, and will not have these agreements in place before a license is issued. In addition, PFS has not presented assurance that each member will provide its share of the planned $6 million aggregate equity contribution or that, if a member fails to provide its share, other members will make up the deficiency.

On the other hand, PFS has supplied information in proprietary documents to the NRC that demonstrate to the staff's satisfaction that PFS has reasonable assurance that it will have adequate funding as required in 10 CFR 72.22(e) before commencing the construction or operation of the Facility. This information, coupled with the financial information that has been provided in non-proprietary documents, the nature of the Facility, and the nature and size of the project's members, provide reasonable assurance of PFS' financial qualifications to construct and operate the Facility without undue risk to public health and safety. The specified initial capacity figure is a proprietary number, which is specified in the PFS' September 15, 1998, and December 3, 1999, submittals (Parkyn, 1998; Gaukler, 1999), and, therefore, is not stated

herein. The staff considers this initial capacity figure to be acceptable. Accordingly, the staff recommends that PFS be required to meet the following financial assurance conditions before constructing or operating the Facility and that these conditions should be part of any order approving the PFS application, in order to demonstrate compliance with 10 CFR 72.22(e):

- Construction of the Facility shall not commence before funding (equity, revenue, and debt) is fully committed that is adequate to construct a Facility with the initial capacity as specified by PFS to the NRC. Construction of any additional capacity beyond this initial capacity amount shall commence only after funding is fully committed that is adequate to construct such additional capacity.

- PFS shall not proceed with the Facility's operation unless it has in place long-term Service Agreements with prices sufficient to cover the operating, maintenance, and decommissioning costs of the Facility, for the entire term of the Service Agreements.

17.1.3 Financial Assurance for Operating Costs

PFS plans to fund Facility operations through agreements with Customers obligated under the Service Agreements to pay PFS an annual fee sufficient to fund operational expenses that are not funded by the capital contributions of PFS members. The PFS Business Plan states this annual fee and shows the forecast of annual and total operating costs and revenues based on a "reference case" scenario extending over a 40-year period from 2002-2042. The Business Plan forecasts positive cumulative cash flows and a positive return on equity over the 40-year period. Specific financial forecasts and other data from the Business Plan cannot be cited herein because of their proprietary nature.

The PFS forecast that its own members will store fuel at a significant level over the life of the Facility, approximating the reference case level of usage, provides a considerable degree of assurance that a base level of revenue to meet operating and maintenance (O&M) costs is likely to be available from the members themselves. Collectively, these members have substantial assets and financial resources so that, in the aggregate, they could provide adequate funding for a project of the size and scope proposed by PFS. The License Application states that the Service Agreements will provide assurance for the continued payment of O&M costs by requiring Customers to meet creditworthiness requirements and, if necessary, provide additional financial assurances (such as irrevocable letters of credit or a third-party guarantee).

In sum, the staff finds that the foregoing factors cited in 17.1.3, in combination with the recommended license conditions cited above, provide reasonable assurance that PFS will have adequate funding to operate the Facility.

17.1.4 Financial Assurance for Decommissioning Funding

As noted earlier, decommissioning funding assurance requires a decommissioning cost estimate and a funding plan providing reasonable assurance that adequate funding will be available for decommissioning costs, pursuant to 10 CFR 72.30(b). Furthermore, the Commission's

regulations require that financial assurance for decommissioning must be provided by one or more of the following methods, pursuant to 10 CFR 72.30(c):

- Prepayment prior to the start of operations in the form of a trust, escrow account, government fund, certificate of deposit, or deposit of government securities.

- A surety method, insurance, or other guarantee method. These methods guarantee that decommissioning costs will be paid. For example, a surety method may be in the form of a surety bond, letter of credit, or line of credit.

- An external sinking fund in which deposits are made at least annually, coupled with a surety method or insurance, the value of which may decrease by the amount being accumulated in the sinking fund.

PFS states on page 1-1 of Appendix B of the License Application that before the end of Facility life, the sealed canisters containing spent fuel will be transferred from their storage casks into shipping casks and then transported off site. Since these canisters will be designed to meet DOE guidance for multipurpose canisters for spent fuel storage, transport, and disposal, the fuel assemblies will remain sealed in the canisters such that decontamination of the canisters will not be required. After shipment of the canisters off site, the Facility will be decommissioned by identification and removal of any residual materials above NRC limits. The site will be released for unrestricted use followed by termination of the NRC license.

PFS states on page 1-7 of the License Application that, while its intention is to maintain the Facility free of radiological contamination at all times, the decommissioning cost estimate conservatively assumes that certain areas and components will require decontamination. The method of funding the Facility decommissioning activities will consist of two components: storage cask decommissioning and decommissioning of the remainder of the Facility.

The estimated decommissioning cost for each storage cask is $17,000, which will be prepaid into an externalized escrow account under the Service Agreement with each Customer prior to shipment of each spent fuel canister to the Facility. PFS plans to place the full amount estimated for decommissioning the casks in a segregated escrow account for this purpose. The staff notes that PFS' proposal to secure payment prior to shipment of the cask to the Facility constitutes a departure from the language in 10 CFR 72.30(c)(1), which indicates that if an applicant selects prepayment as the method of decommissioning funding, payment should be made "prior to the start of operation." Notwithstanding this difference, however, the PFS proposal assures that (a) reasonable assurance of adequate funding to decommission the Facility will be provided prior to the commencement of operations (see the following paragraph), as required in 10 CFR 72.30(c); and (b) funding to decommission the casks will be provided prior to construction of each cask (i.e., prior to commencement of any operations involving that cask), thus assuring that each cask that is constructed will be decommissioned. Accordingly, PFS' decommissioning funding plan provides reasonable assurance that decontamination and decommissioning at the end of Facility operations will provide adequate protection of the public health and safety and satisfies 10 CFR 72.30(c). Although funding for decommissioning the casks will be provided prior to cask construction rather than prior to the commencement of Facility operations, since the decommissioning funding plan provides reasonable assurance of

adequate funding, an exemption from strict compliance with the language in 72.30(c)(1) would be issued as part of the license, if necessary, to authorize implementation of the PFS plan.

PFS estimates the cost of decommissioning the remainder of the Facility and site to be $1.631 million, which is to be funded through a letter of credit coupled with an external sinking fund, pursuant to 10 CFR 72.30(c)(3). Customers will be required under the Service Agreements to pay the cost of decontaminating any portion of the Facility for which they may be responsible for contaminating. As the actual cost of decontamination and decommissioning is paid into the external sinking fund, PFS plans for the letter of credit to be reduced by an equivalent amount, pursuant to 10 CFR 72.30(c)(3). The per-canister fee and amounts of the escrow account, external sinking fund, and letter of credit are to be reviewed and adjusted annually to account for inflation and any changes in the scope of decommissioning.

PFS estimates the specific cost of components of decommissioning the remainder of the Facility and site as follows (these are non-proprietary figures cited in Appendix B of the LA):

Site Characterization Survey	$250,000
Decommissioning Four Transfer Casks	$200,000
Decommissioning Eight Shipping Casks	$400,000
Decontaminating Canister Transfer Building	$230,000
Storage Pad Decontamination	$241,000
Final Release Survey	$260,000
Independent Verification Survey	$ 50,000
Total	$1,631,000

The staff finds these estimates of decommissioning costs to be reasonable. Further, the staff finds this surety method of a letter of credit coupled with an external sinking fund, and per-cask prepayment, as proposed by PFS to be acceptable for meeting the requirements of 10 CFR 72.30(c).

17.1.5 PFS Liability Insurance

PFS has committed to pursue and to maintain nuclear liability insurance in the maximum commercially available amount of $200 million. The NRC does not have specific insurance and indemnity requirements for Part 72 facilities. PFS' commitment to provide nuclear liability insurance, in addition to the funding required by NRC regulations, is acceptable to the staff.

17.2 Evaluation Findings

PFS has identified anticipated sources of equity capital and revenue to fund construction of the Facility, with much of the total revenue being required from Customers as prepayments before they actually ship spent fuel to the Facility. To fund ongoing operations, Customers will pay some additional prepaid fees, plus a relatively small annual storage fee in comparison to their prepaid fees. Also, the estimated $17,000 cost for decommissioning each Customer storage cask is to be prepaid by Customers in accordance with terms of the Service Agreement. The estimated $1.631 million cost of decommissioning the remainder of the Facility and the site is a small fraction of the construction cost and is guaranteed by a surety method acceptable to the NRC.

Accordingly, the staff believes that PFS has provided reasonable assurance of its financial qualifications to construct, operate, and decommission the Facility as proposed, subject to the conditions stated herein, in accordance with the requirements of 10 CFR Part 72.

Proposed License Conditions

LC17-1 Construction of the Facility shall not commence before funding (equity, revenue, and debt) is fully committed that is adequate to construct a facility with the initial capacity as specified by PFS to the NRC. Construction of any additional capacity beyond this initial capacity amount shall commence only after funding is fully committed that is adequate to construct such additional capacity.

LC17-2 PFS shall not proceed with the Facility's operation unless it has in place long-term Service Agreements with prices sufficient to cover the operating, maintenance, and decommissioning costs of the Facility, for the entire term of the Service Agreements.

17.3 References

Donnell, J.L. Private Fuel Storage Limited Liability Company. *Supplemental Response to RAIs.* Letter to Director, Office of Nuclear Material Safety and Safeguards, Nuclear Regulatory Commission. Docket No. 72-22. June 15, 1999.

Donnell, J.L. Private Fuel Storage Limited Liability Company. *Response to RAI LA 1-6.* Letter to Director, Office of Nuclear Material Safety and Safeguards, Nuclear Regulatory Commission. Docket No. 72-22. June 18, 1999.

Gaukler, P.A. Shaw Pittman. *Private Fuel Storage – Docket No. 72-22 – ASLB No. 97-732-02.* Letter to the E. M. Julian, U. S. Nuclear Regulatory Commission. December 3, 1999.

Parkyn, J.D. Private Fuel Storage Limited Liability Company. *Response to Request for Additional Information.* Letter to Director, Office of Nuclear Material Safety and Safeguards, Nuclear Regulatory Commission. Docket No. 72-22. May 19, 1998.

Parkyn, J.D. Private Fuel Storage Limited Liability Company. *Supplemental Response to RAIs.* Letter to Director, Office of Nuclear Material Safety and Safeguards, Nuclear Regulatory Commission. Docket No. 72-22. September 15, 1998.

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, and May 19, August 10, August 27, and September 8, and September 21, 1999.

18 PHYSICAL PROTECTION PLAN

18.1 Conduct of Review

This chapter of the SER provides a Safeguards Evaluation Report that addresses the Physical Protection Plan (Plan) submitted by PFS in support of the application to license the Private Fuel Storage Facility. The Plan was submitted to NRC for review. Formal requests for additional information were sent to the applicant, and the applicant's responses to these requests were adequate. This safeguards evaluation is based on Revision 2 to the PFS Physical Protection Plan.

PFS's Physical Protection Plan was reviewed for conformance with the following regulations:

- 10 CFR 72.180 requires that the licensee establish, maintain, and follow a detailed plan for physical protection as described in 10 CFR 73.51.

- 10 CFR 73.51 specifies the requirements for the physical protection of spent nuclear fuel and high-level radioactive waste.

The Standard Review Plan for Physical Protection Plans for the Independent Storage of Spent Fuel and High-Level Radioactive Waste, NUREG-1619, July 1998, provided guidance for the staff reviewers.

18.1.1 Facility Description

The PFS Plan provides an adequate description of the facility and site. It includes site maps showing the cask storage area, important supporting structures, and the boundaries of the protected area as well as descriptions of the area adjacent to the site.

18.1.2 General Performance Objectives

The general objective of the physical protection system is to provide high assurance that activities involving spent nuclear fuels do not constitute an unreasonable risk to public health and safety.

To achieve this objective, the physical protection system must provide for the following performance capabilities in accordance with 10 CFR 73.51(b):

- store spent nuclear fuel and high level radioactive waste only within a protected area;

- grant access to the protected area only to individuals who are authorized to enter the protected area;

- detect and assess unauthorized penetration of, or activities within the protected area;

- provide timely communication to a designated response force whenever necessary; and

- manage the physical protection organization in a manner that maintains its effectiveness.

In addition, 10 CFR 73.51(b)(3) requires that the physical protection system be designed to protect against loss of control of the facility that could be sufficient to cause a radiation exposure exceeding the dose specified in 10 CFR 72.106(b) from any design basis accident.

The applicant has reaffirmed the general design objective of the implemented physical protection system to protect the storage of spent fuel and to protect the facility from loss of control by providing a physical protection plan with commitments that meet the requirements of 10 CFR 72.180 and 73.51.

18.1.3 Physical Barrier Systems

As required by 10 CFR 73.51(d)(1), the applicant must store spent fuel only within a protected area so that access to this material requires passage through or penetration of two physical barriers, one barrier at the perimeter of the protected area and one barrier offering substantial penetration resistance.

The applicant has provided for spent fuel to be stored within a protected area such that access to stored spent fuel requires passage through or penetration of at least two security barriers. The first barrier is the protected area barrier which is comprised of double fences, meeting the definition of physical barrier in 10 CFR 73.2. The protected area barrier includes twenty-foot isolation zones between the outer and inner fences as well as on either side of the protected area barrier system. The inner isolation zone is free from clutter and is provided with an intrusion detection system prior to penetration of the inner fence of the protected area barrier system. The second barrier (the storage casks), which are constructed of high density concrete, along with metal liners, offers substantial penetration resistance by requiring specialized equipment or explosives to penetrate the cask and disperse nuclear materials. Once installed, these barriers will be the included in the pre-operational inspection.

The commitments in the Plan for physical barriers meet the requirements of 10 CFR 73.51(d)(1).

18.1.4 Illumination

As required by 10 CFR 73.51(d)(2), illumination must be sufficient to permit adequate assessment of unauthorized penetrations of or activities within the protected area.

The applicant has provided for sufficient illumination to allow surveillance and adequate assessment within the protected area. Illumination will be included in pre-operational performance inspections to assure the illumination levels are sufficient.

The commitments in the Plan for illumination therefore meet the requirements of 10 CFR 73.51(d)(2).

18.1.5 Surveillance

As required by 10 CFR 73.51(d)(3), the perimeter of the protected area must be subject to continual surveillance and be protected by an active intrusion alarm system which is capable of detecting penetrations through the isolation zone and that is monitored in a continually-staffed primary alarm station and in one additional continually-staffed location. The primary alarm station must be located within the protected area and have bullet-resisting walls, doors, ceiling and floor; and the interior of the station must not be visible from outside the protected area. A timely means for assessment of alarms must also be provided. Regarding alarm monitoring, the redundant location need only provide a summary indication that an alarm has been generated.

The applicant has committed to have the capability to detect unauthorized penetrations through the isolation zones at the perimeter of the protected area. The intrusion detection system covers all of the inner areas of the protected area. The intrusion detection system is comparable to those systems described in Regulatory Guide 5.44, "Perimeter Intrusion Detection Systems." The applicant commits to meeting Regulatory Guide 5.44. The intrusion detection system is tamper-indicating and has line supervision.

The applicant has provided a primary alarm station (PAS) located in the Security and Health Physical Building Access Control Facility. This PAS is a hardened facility that is within the protected area. It is protected by the protected area intrusion detection system, access control, and barriers which meets acceptable standards (UL 752, Standard for Bullet-Resisting Equipment). All access control and intrusion alarms are monitored from this facility. A summary indication of alarms also annunciate in the alternate alarm station (AAS) which is located in the Administration Building.

Once installed, these surveillance systems including the alarm stations will be the included in the pre-operational inspection.

The commitments in the Plan for alarm surveillance and annunciation therefore meets the requirements of 10 CFR 73.51(d)(3).

18.1.6 Security Patrols

As required by 10 CFR 73.51(d)(4), the protected area must be monitored by daily random patrols.

The applicant has provided for security force personnel who are on duty at all times. Normal duties include the operation of the PAS, the AAS, and control of personnel entry, including searches of persons who enter the protected area. Security force personnel conduct daily random patrols to monitor the protected area boundaries for the presence of unauthorized persons or activities, and for physical protection system or barrier degradation.

The commitments in the Plan for patrols therefore meets the requirements of 10 CFR 73.51(d)(4).

18.1.7 Security Organization

As required by 10 CFR 73.51(d)(5), a security organization with written procedures must be established. The security organization must include sufficient personnel per shift to provide for monitoring of detection systems and the conduct of surveillance, assessment, access control, and communications to assure adequate response. Members of the security organization must be trained, equipped, qualified, and re-qualified to perform assigned job duties in accordance with Appendix B to Part 73, Sections I.A.1.a. and b., I.B.1.a., and the applicable portions of Section II.

The applicant has established a security organization that includes trained individuals, oversight, and written procedures in which to carry out security duties. This organization provides for a security force captain, sergeants, and officers. Each shift has sufficient armed individuals to meet regulatory requirements. The applicant has chosen to provide trained armed individuals (guards) instead of watchmen in order to augment the ability of the guard to control the site pending the arrival of the offsite response force. Shift manning levels may be increased dependent upon planned daily activities. In addition the applicant has provide guard training and qualification sufficient to meet the requirements of Section III of Appendix B to 10 CFR Part 73.

18.1.7.1 Qualifications for Employment in Security

The applicant has committed to perform screening for individuals, including security personnel, granted unescorted access to the protected area where spent fuel is stored prior to the granting of such access. Security force personnel shall meet the requirements of 10 CFR Part 73, Appendix B, General Criteria for Security Personnel, Sections I. A. 1. a., Educational Development; I. A. 1. b., Felony Convictions; I. B. 1. a., Physical Weaknesses or Abnormalities; and the applicable portions of Section II, Training and Qualifications. The screening includes a five-year local criminal history check of counties the individual has resided in within the five-year period prior to assignment as a security force member. Psychological evaluation, Federal Bureau of Investigations criminal history records, drug and alcohol testing and a continual behavioral observation program is included in the PFS established access authorization program.

18.1.7.2 Security Force Training

The applicant submitted an ISFSI Security Training and Qualification Plan as an attachment to its Physical Protection Plan. The Plan documents that the applicable criteria of Appendix B to Part 73 will be met.

The applicant has committed to training and qualifying all non-supervisory security personnel to all non-supervisory duty functions including PAS and AAS operator, physical searches, personnel identification, and logging functions as well as response functions. The shift sergeant will also be trained and qualified to perform all of the non-supervisory functions. All shift security

personnel are to be trained in searching for firearms, explosive materials, and incendiary devices.

The ISFSI Security Training and Qualification Plan is included as part of the "Private Fuel Storage Independent Spent Fuel Storage Installation Security Plan" and includes firearms training which meets **the requirement** of Appendix B to 10 CFR Part 73 Section III.

18.1.7.3 Security Organization - Staff Evaluation Finding

Once implemented the security organization and training will be included in the pre-operational inspection.

Based on the discussions above, the staff finds that the commitments in the Plan for the security organization meets the requirements of 10 CFR 73.51(d)(5).

18.1.8 Response Liaison

As required by 10 CFR 73.51(d)(6), documented liaison with a designated offsite response force or local law enforcement agency (LLEA) must be established to permit timely response to unauthorized penetration or activities.

The applicant has included a Site Safeguards Contingency Plan as an attachment to its Physical Protection Plan. The Contingency Plan includes documented liaison with the Tooele County Sheriff as the LLEA. Timely response is provided through the use of an augmented armed onsite response force combined with the offsite LLEA response.

The commitments in the Plan for offsite response therefore meets the requirements of 10 CFR 73.51(d)(6).

18.1.9 Identification and Controlled Lock Systems

As required by 10 CFR 73.51(d)(7), a personnel identification system and a controlled lock system must be established and maintained to limit access to authorized individuals.

The applicant has included in its Physical Protection Plan an identification system which will be used at the facility. The system provides unique identification of individuals granted unescorted access to the protected area. In addition, the identification system identifies individuals requiring escort while within the protected area.

The licensee has implemented a key and lock control system that will limit access to, and within, the protected area to authorized individuals.

Once implemented the identification and controlled lock system will be included in the pre-operational inspection.

The commitments in the Plan for identification and controlled lock systems therefore meet the requirements of 10 CFR 73.51(d)(7).

18.1.10 Communications Capability

As required by 10 CFR 73.51(d)(8), redundant communications capability must be provided between onsite security force members and designated response force or LLEA.

The applicant in its Physical Protection Plan commits to each security individual being equipped with two-way radios capable of maintaining continuous communications with the security posts. The Primary Alarm Station has both a base radio system and a commercial telephone to maintain contact with the LLEA. Onsite communication is backed up by an uninterruptible power supply (UPS). Therefore, redundant communications is available between the onsite security force and the offsite response force.

Once implemented the communication capability will be included in the pre-operational inspection.

The commitments in the Plan for communications capability therefore meets the requirements of 10 CFR 73.51(d)(8).

18.1.11 Access Controls at the Protected Area

As required by 10 CFR 73.51(d)(9), all individuals, vehicles, and hand-carried packages entering the protected area must be checked for proper authorization and visually searched for explosives before entry.

18.1.11.1 Access to Protected Areas

The Physical Protection Plan describes procedures for determining an individual's need for access to the protected area. Access to protected areas is limited to individuals authorized escorted or unescorted access in order to perform job duties. Procedures are also described for dealing with required access of emergency response personnel vehicles.

18.1.11.2 Access Controls at the Protected Area

The applicant's has provided procedures for granting access of individuals and packages into the protected area. Only those vehicles listed on the Designated Vehicles List are allowed into the protected area. Authorization is checked and individuals, packages, and vehicles are searched for firearms, incendiary devices, and explosives. The search is conducted visually and by physical search (pat down) or with the use of a portable explosive detector.

18.1.11.3 Escorts and Escorted Individuals

The applicant's Plan identifies the individuals designated to be granted unescorted access into the protected area as well as describes the requirements and procedures for escorting individuals who need escorted access.

18.1.11.4 Access Controls at the Protected Area - Staff Evaluation Finding

Once implemented the access control measures will be included in the pre-operational inspection.

Based on the discussions above, the staff finds that the commitments in the Plan for access control commitments meet the requirements of 10 CFR 73.51(d)(9).

18.1.12 Procedures

As required by 10 CFR 73.51(d)(10), written response procedures must be established and maintained for addressing unauthorized activities within the protected area including Category 5, "Procedures," of Appendix C to Part 73. The applicant shall retain a copy of response procedures as a record for 3 years or until termination of the license for which the procedures were developed. Copies of superseded material must be retained for 3 years after each change or until termination of the license.

The applicant response procedures for dealing with detection of unauthorized presence or activities within the protected area are described in its Physical Protection Plan. These procedures detail the actions to be taken and decisions to be made by each member or unit of the response organization

Once implemented the security procedures will be included in the pre-operational inspection.

The commitments in the Plan to provide procedures therefore meets the requirements of 10 CFR 73.51(d)(10).

18.1.13 Equipment Operability

As required by 10 CFR 73.51(d)(11), all detection systems and supporting subsystems must be tamper-indicating with line supervision. These systems, as well as surveillance/assessment and illumination systems, must be maintained in operable condition. Timely compensatory measures must be taken after discovery of an inoperable condition, to assure that the effectiveness of the security system is not reduced.

The applicant has committed to perform testing of all security related equipment to applicable manufacturer's specifications. The applicant has committed to check the security systems and support equipment for operability weekly and each time the equipment is used. The applicant has committed to a repair and preventive maintenance program as well as interim compensatory measures until the system is restored to normal capability. The applicant commits to following Regulatory Guide 5.44, "Perimeter Intrusion Alarm Systems" operability tests.

Once implemented the measures to assure equipment operability will be included in the pre-operational inspection.

The commitments in the Plan for equipment operability therefore meet the requirements of 10 CFR 73.51(d)(11).

18.1.14 Audits

As required by 10 CFR 73.51(d)(12), the Physical Protection Program must be reviewed once every 24 months by individuals independent of both Physical Protection Program management and personnel who have direct responsibility for implementation of the Physical Protection Program. The Physical Protection Program review must include an evaluation of the effectiveness of the physical protection system and a verification of the liaison established with the designated response force or LLEA.

The applicant has committed to conduct security audits at least every 24 months by individuals independent of both security program management and of personnel directly responsible for implementation of the security program. The audits include evaluation of the effectiveness of the physical protection system and verification of the liaison established with the LLEA. The reports are maintained in a form sufficient for auditing, available for inspection, for a period of three years.

The commitments in the Plan for the audit program therefore meets the requirements of 10 CFR 73.51(d)(12).

18.1.15 Documentation

As required by 10 CFR 73.51(d)(13), documentation must be retained as a record for 3 years after the record is made or until termination of the license. Duplicate records to those required under 72.180 of Part 72 and 73.71 of this part need not be retained under the requirements of this section.

The applicant's Contingency Plan describes response record data and commits to maintaining those records for a period of three years. These records include:

- screening records until the affected individual terminates employment;
- training and qualification records required by Appendix B, Section II. B;
- current written procedures that require access control personnel to identify authorized versus unauthorized entry for the period the applicant stores spent fuel;
- the record of escorted individuals for a period of three years from the date of the record;
- written procedures for key and lock control for the period the applicant stores spent fuel;
- audit reports and resolutions; and
- a record of assessment and response to alarms.

The commitments in the Plan for record keeping therefore meets the requirements of 10 CFR 73.51(d)(13).

18.2 Evaluation Findings

As required by 10 CFR 72.180, the Physical Protection Plan describes how the applicant will meet the requirements of 10 CFR 73.51. The staff has concluded that the Private Fuel Storage Facility Physical Protection Plan, Revision 2; Safeguards Contingency Plan, Revision 1; and the Security Training and Qualification Plan, Revision 1 are adequate and meet the requirements of 10 CFR 72.180 and 10 CFR 73.51. Therefore, the requirements of 10 CFR 72.180 have been satisfied. Further, when fully implemented the applicant's physical protection program satisfies the provisions of 10 CFR 72.40(a)(14) by providing for the common defense and security and the protection of the health and safety of the public.

Proposed License Conditions

LC18-1 The licensee shall follow the Physical Protection Plan entitled, "Private Fuel Storage, L.L.C. Independent Spent Fuel Storage Installation Security Plan," dated June 8, 1999, and as it may be further amended under the provisions of 10 CFR 72.44(e) and 72.84(d).

LC18-2 The licensee shall follow the Safeguards Contingency Plan entitled, "Private Fuel Storage, L.L.C. Independent Spent Fuel Storage Installation Safeguards Contingency Plan," dated June 8, 1999, and as it may be further amended under the provisions of 10 CFR 72.44(e) and 72.84(d).

LC18-3 The licensee shall follow the Guard Training and Qualification Plan entitled "Private Fuel Storage, L.L.C. Independent Spent Fuel Storage Installation Security Training and Qualification Plan," dated June 8, 1999, and as it may be further amended under the provisions of 10 CFR 72.44(e) and 72.84(d).

18.3 References

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.

19 TECHNICAL SPECIFICATIONS

The evaluation of the proposed technical specifications relies upon cask-specific information and is, therefore, not included in this SER at this time.

20 CONCLUSIONS

The staff has reviewed the site-related design, testing, operations, and maintenance activities for the Private Fuel Storage Facility, as described in the following documents submitted by the applicant:

- the License Application, which contains general and financial information, the applicant's technical qualifications, technical specifications, and a preliminary decommissioning plan;

- the Safety Analysis Report for the Private Fuel Storage Facility;

- the Emergency Plan for the Private Fuel Storage Facility; and

- the Security Plan for the Private Fuel Storage Facility, which includes the safeguards contingency plan.

Based on the information provided, the staff has reasonable assurance that the applicant's financial qualifications, Quality Assurance Program, Emergency Plan for the Facility, and Security Plan for the Facility are acceptable and meet the applicable requirements of 10 CFR Part 72. The staff cannot make a final determination on the viability of the proposed site or on whether the Facility is capable of meeting the requirements of 10 CFR Part 72 until PFS has provided the information on the open items identified in this SER.

Section 72.40(a) of 10 CFR Part 72, specifies the conditions that must be satisfied for the Commission to issue a license under this Part. The status of the Private Fuel Storage Facility application with respect to these conditions is as follows:

- 10 CFR 72.40(a)(1) - The staff is unable to make the finding that the applicant's proposed ISFSI design complies with Subpart F of 10 CFR Part 72. To make this finding, the specific design of the dry cask storage system must be considered and the open items identified in this SER must be resolved.

- 10 CFR 72.40(a)(2) - The staff is unable to make the finding that the proposed site complies with the criteria in Subpart E of 10 CFR Part 72. To make this finding, the open items identified in this SER must be resolved.

- 10 CFR 72.40(a)(3) - This condition does not apply because the proposed ISFSI is not located on the site of a nuclear power plant or other licensed activity or facility.

- 10 CFR 72.40(a)(4) - Based on the evaluation presented in Chapter 10 of this SER, the staff has made the finding that applicant is qualified by reason of training and experience to conduct the operation covered by the regulations in this part.

- 10 CFR 72.40(a)(5) - In Chapter 3 of this SER, the staff found that the applicant's proposed operating procedures to protect health and to minimize danger to life or property are adequate for those activities that are not dependent on the design of the dry cask storage system. The staff has made no findings regarding the adequacy of the cask-specific or cask-dependent operating procedures which requires consideration of the specific design of the dry cask storage system.

- 10 CFR 72.40(a)(6) - In Chapter 17 of this SER, the staff found that the applicant for the ISFSI is financially qualified to engage in the proposed activities in accordance with the regulations in this part.

- 10 CFR 72.40(a)(7) - In Chapter 12 of this SER, the staff found that the applicant's quality assurance plan complies with Subpart G of 10 CFR Part 72.

- 10 CFR 72.40(a)(8) - In Chapter 18 of this SER, the staff found that the applicant's physical protection provisions comply with Subpart H of 10 CFR Part 72.

- 10 CFR 72.40(a)(9) - In Chapter 10 of this SER, the staff found that the applicant's personnel training program complies with Subpart I of 10 CFR Part 72.

- 10 CFR 72.40(a)(10) - As discussed in Chapter 13 of this SER, staff is unable to make the overall finding that the applicant's decommissioning plan, pursuant to §72.30, provide reasonable assurance that the decontamination and decommissioning of the ISFSI at the end of its useful life will provide adequate protection to the health and safety of the public. To make this finding, the specific design of the dry cask storage system must also be considered.

- 10 CFR 72.40(a)(11) - In Chapter 16 of this SER, the staff found that the applicant's emergency plan complies with 10 CFR 72.32.

- 10 CFR 72.40(a)(12) - The applicable review fees, which have been assessed in accordance with 10 CFR Part 170, have not been paid. A license will not be issued until the fees have been remitted.

- 10 CFR 72.40(a)(13) - The staff is unable to make the overall finding that there is reasonable assurance that: (i) The activities authorized by the license can be conducted without endangering the health and safety of the public and (ii) these activities will be conducted in compliance with the applicable regulations of this chapter. To make this finding, the specific design of the dry cask storage system must be considered and the open items identified in this SER must be resolved.

- 10 CFR 72.40(a)(14) - In Chapter 18 of this SER, the staff found that the issuance of a license for the Private Fuel Storage Facility will not be inimical to the common defense and security.