72-22

Private Fuel Storage, LLC

SEE REPORT

P.O. Box C4010, La Crosse, WI 54602-4010
John D. Parkyn, Chairman of the Board

May 19, 1998

Director
Office of Nuclear Material Safety and Safeguards
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

RESPONSES TO REQUEST FOR ADDITIONAL INFORMATION
PRIVATE FUEL STORAGE FACILITY
DOCKET NO. 72-22 / TAC NO. L22462
PRIVATE FUEL STORAGE L.L.C.

References: 1) NRC Letter, Delligatti to Parkyn, Request for Additional Information, dated April 1, 1998
2) PFSLLC Letter, Parkyn to Director, Office of Material Safety and Safeguards, Response to Request for Additional Information, dated April 29, 1998

Please find enclosed Private Fuel Storage responses (original plus 15 copies) to the NRC Request for Additional Information (Ref. 1). A response is provided for each RAI, but as noted in my April 29, 1998 letter (Ref. 2), certain RAIs require supplemental information in accordance with the included schedule.

Some RAI responses contain proprietary information as noted and will be submitted under separate cover with the required affidavit in accordance with 10 CFR 2.790.

If you have any questions regarding this response, please contact me at 608-787-1236 or our Project Director, John Donnell, at 303-741-7009.

Sincerely yours,

John D. Parkyn, Chairman
Private Fuel Storage, LLC

JDP:cls
Enclosures

cc: Mr. Leon Bear
Ms. Denise Chancellor
Mr. Mark Delligatti
Mr. Jay Silberg

NF06

RESPONSES TO REQUEST FOR ADDITIONAL INFORMATION

The responses are organized in the same format as that presented in the RAI. The License Application RAIs are addressed first, followed by those for the Safety Analysis Report. Emergency Plan RAIs are addressed under SAR Chapter 9.

Before each RAI response is presented, the NRCs RAI comment is repeated. A response is provided for each RAI, although the response may only include a schedule of when supplemental information will be provided or that the response contains proprietary information which is submitted under separate cover.

Enclosures are placed following the responses and include:

- A 3.5" diskette containing meteorological data in two files. Referenced in the response to SAR RAI 2-1.

- J. D. Stevenson report, "Tipping Evaluation of Spent Fuel Storage Casks Subjected to Site Specific Earthquake Loading (ISFSI DE) for the Private Fuel Storage Facility", Revision 0, June 17, 1997. Referenced in the response to SAR RAI 3-2.

- Calculation No. 05996.02-UR-5, "Dose Rate Estimates from Storage Cask Inlet Duct Clearing Operations". Referenced in the response to SAR RAI 7-2.

- Calculation No. 05996.02-UR-6, "Calculational Basis for PFSF SAR Tables 7.4-1 and 7.4-2, Estimated Personnel Exposures for Canister Transfer Operations". Referenced in the response to SAR RAI 7-3.

- Calculation No. 05996.02-UR-3, rev. 1, "Postulated Release of Removable Contamination from Canister Outer Surfaces - Dose Consequences". Referenced in the response to SAR RAI 8-5.

- Calculation No. 05996.02-UR-7, "Ingestion Dose Rate Estimates to Nearest Resident from Off-normal Contamination Release of Surface Contamination from the Exterior of a Canister". Referenced in the response to SAR RAI 8-5.

LICENSE APPLICATION

LA Chapter 1, Section 1-6

1-1 Provide the text of the subscription agreement with PFS member utilities showing the terms and schedule for their provision of equity funds for the independent spent fuel storage installation (ISFSI) facility construction, including the contingency for providing additional funds if some of the eight members decide not to participate.

RESPONSE

The schedule for submitting the response to this RAI is September 15, 1998.

LICENSE APPLICATION

LA Chapter 1, Section 1-4

1-2 Provide a list of the eight member utilities which the PFS LA states are the owners of PFS and which are responsible for funding a portion of facility construction, operations, and decommissioning, plus a copy of the limited liability company agreement among them.

RESPONSE

The schedule for submitting the response to this RAI is September 15, 1998.

LICENSE APPLICATION

LA Chapter 1, Section 1-6

1-3 (a) Provide adequate information to explain the basis for the $100 million estimated cost for facility construction.

- Specify whether this amount is anticipated as being needed for the 15,000 MTU nominal target for the facility or for the 40,000 MTU facility capacity.

(b) Provide an itemized description for each of the major construction tasks in the overall estimate.

RESPONSE

The response to this RAI contains proprietary information and will be submitted under separate cover.

LICENSE APPLICATION

LA Chapter 1, Section 1-6

1-4 Provide the PFS financing plan and the text of the service agreement with customers, which together should show:

(a) The customer charge to fund the non-equity portion of facility construction and the terms and schedule for payment to PFS.

(b) The plan for debt financing which PFS would use to finance the non-equity portion of construction if PFS chooses this option in whole or in part (debt financing is referred to on page 1-6 of the LA).

RESPONSE

The schedule for submitting the response to this RAI is September 15, 1998.

LICENSE APPLICATION

LA Chapter 1, Section 1-6

1-5 (a) Provide the information used as the basis for determining the estimated average annual operation and maintenance (O&M) cost of the facility.

- It is unclear whether the estimated average annual O&M costs of $49 million per year (for a 20 year facility life) and of $31 million per year (for a 40 year life) are based on a full 4,000 cask capacity utilization rate or some other amount.
- It is also unclear whether these estimates are expressed in 1998 dollars or future dollars.

(b) Describe how customer fees are to be adjusted as O&M costs vary over time, especially if costs are much greater than now expected.

RESPONSE

(a) The elements that make up the estimated annual operation and maintenance costs include the following: labor, operations support, storage canisters, storage casks, transportation fees, transport and storage consumables, maintenance and parts, regulatory fees, quality assurance and other expenses, low-level radioactive waste disposal, contingencies, radiological decommissioning funds, non-radiological decommissioning fund, and associated costs of operating a facility. Note that the O&M costs of $49 million per year for a 20 year facility life and of $31 million per year for a 40 year life include such high-priced items as the storage system canisters / casks and shipping rates. When these canister fees are extracted, the routine annual O&M costs are approximately $10 million per year.

The O&M costs noted above are based on a nominal design capacity case of 15,000 Mtu (see the response to RAI LA 1-3). All dollars expressed are in current year dollars at the time of the license application submittal (1997).

(b) The customers of PFS will be signing Service Agreements which will include escalators that are tied to specific costs of doing business at the site. Services, such as labor and utilities, will be tied to nationally published indices for the regional area in Utah. Costs, such as Nuclear Regulatory Commission and insurance fees, will be escalated at actual escalation numbers. Therefore, customers will be responsible for the actual costs of ensuring operating and maintenance funding for the facility on a year-by-year basis as long as their fuel is stored. Member utilities also sign separate Customer Agreements to ensure that these same restrictions apply.

LICENSE APPLICATION

LA Appendix B, Chapters 4 and 5

1-6 (a) Provide the facility size associated with the PFS $1,631,000 decommissioning estimate for the facility and site--whether it is 15,000 MTU or 40,000 MTU.

(b) Provide the basis for estimating each key decommissioning cost component.

RESPONSE

The schedule for submitting the response to this RAI is June 15, 1998.

LICENSE APPLICATION

LA Appendix B, Chapter 5, Section 5-2

1-7 Provide a copy of the actual PFS letter of credit (or its proposed text) which PFS states will provide decommissioning funding assurance for the $1,631,000 which PFS estimates will be needed for facility and site decommissioning costs.

- It should state whether the amount in the letter of credit will escalate over time if the cost of decommissioning increases above the estimated amount.

RESPONSE

The response to this RAI contains proprietary information and will be submitted under separate cover.

LICENSE APPLICATION

LA Appendix B, Chapter 5, Section 5-2

1-8 (a) Provide a description of the specific methods which will be used to monitor the annual adjustments in anticipated decommissioning costs as proposed by PFS on page 5-2 of Appendix B of the PFS LA.

- The description should include the use of a specific indicator of inflation, revised cost estimates, or other means by which PFS will monitor expected changes in specific components of expected future decommissioning costs.

(b) Indicate what method will be used to assure additional funds if for some reason(s) the actual facility and site decommissioning costs were to be significantly greater than the estimated $1,631,000.

RESPONSE

(a) Changes in the cost of decommissioning will be accounted for through an annual review of the decommissioning cost estimate to ensure that both the individual elements and the overall estimate either remain valid or are revised to account for any changes in the tasks, scope, cost or schedule for decommissioning. Additionally, the decommissioning cost estimate will be adjusted annually to account for the effects of inflation, utilizing the conservatively high Consumer Price Index, published by the Bureau of Labor Statistics. The amount of the Letter of Credit will be adjusted to account for any changes in the overall decommissioning costs and for deposits into the external sinking fund.

(b) The most significant element in providing sufficient financial assurance for future decommissioning costs is a decommissioning cost estimate which is both comprehensive and conservative. A good cost estimate, which is reviewed and adjusted annually, will not involve uncertainties which could cause the total amount to be exceeded. A financial assurance amount based on such a conservative cost estimate will in fact be adequate to cover all costs associated with decommissioning.

LICENSE APPLICATION

LA Chapter 9 Physical Protection

9-1 Describe the physical security and safeguards plans which will be put in place for transportation activities at Rowley Junction.

- A separate RAI will be provided regarding the information previously submitted by PFS.

RESPONSE

As stated in SAR Section 4.5, operations at the intermodal transfer point will be performed under the requirements of 10 CFR 71. Therefore, security operations while spent fuel is present at the transfer point, will be in accordance with the requirements of 10 CFR 73.37, "Requirements for Physical Protection of Irradiated Reactor Fuel in Transit," including specific requirements of sections (c) "Shipments by Road" and (d) "Shipments by Rail." These security operations will include:

- Notification to the NRC of planned shipments,
- Use of security procedures for coping with safeguards emergencies,
- Use of full time escorts to maintain visual surveillance of the shipment,
- Establishment of a continuously staffed communications center that maintains contact with the shipment escorts,
- Maintenance of written logs by the escorts of the shipment and significant events,
- Arrangements with local law enforcement agency (LLEA) for their response as necessary,
- Maintain communications with a communications center at least every 2 hours, LLEA, and one another to provide shipment status and report safeguards emergencies,
- NRC approval of shipment routes and transfer point, and
- Training of escort personnel per 10 CFR 73, Appendix D.

The intermodal transfer point is a transition from shipment by rail to shipment by road, both of which will meet the requirements listed above. The PFS will maintain escort staff for both rail and road security. While in transit, the status of the spent fuel shipment will be continuously monitored, including the intermodal transfer point, by the communications center.

Additional security measures that will be implemented at the intermodal point include the following:

- An 8 ft. high chain link security fence encompassing the entire site, including the weather enclosure, railroad spur tracks, and yard area.

- A Metal building (weather enclosure) housing the crane and transfer equipment.
- Locks on the gates and building doors to help prevent unauthorized entry of the site when unoccupied by PSFS personnel.
- High pressure sodium yard lights that will fully illuminate the site throughout the night.
- Procedures that verify integrity of equipment prior to every transfer operation.
- Motion detectors located around the yard and in the building that will alarm at the PFSF if tripped.

In the event the motion detectors are tripped while the point is unoccupied, several actions will be taken to ensure the security and integrity of the intermodal point equipment, which include:

- Dispatching security personnel to investigate the site for any unauthorized entry.
- Procedures for inspecting and testing transfer equipment prior to reuse.

SAFETY ANALYSIS REPORT

CHAPTER 1—INTRODUCTION AND GENERAL DESCRIPTION OF INSTALLATION

Section 1.6 Material Incorporated by Reference

1-1 Provide a means of tracking changes to the storage system casks as represented in the SARs that may influence the conclusions used to complete this analysis.

- This review is based on the assumption that the design and analysis of the storage system casks, as included in the SARs which the staff is currently reviewing, are found to be adequate and are certified. Any changes to the design or analysis could directly impact this review.

RESPONSE

The PFSF project is currently on controlled distribution for vendor SARs, SAR questions and responses, vendor calculations, vendor drawings and associated design/licensing information. This information is provided by the vendors for both the storage system and transportation system. All transmittals or letters which accompany this information are filed in the project job book.

Design information and SAR changes received from the vendors are accompanied by a receipt acknowledgment form. After review of the information to determine that the complete package has been received, the design information or SAR change is entered into the project action tracking log and the receipt acknowledgment form is signed and returned to the vendor.

The information is then reviewed in detail to determine the impact, if any, on the PFSF license application, design drawings or calculations. The project action tracking log will be used to record what actions are necessary by the PFSF project and the scheduled completion date for each action, as well as tracking the actions to completion.

As needed, PFSF calculations and drawings are revised and any required revisions to the PFSF license application and calculation package will be prepared, reviewed and submitted to the NRC. Completed actions are recorded in the project action tracking log.

The revised information is filed in the appropriate project location (i.e., SAR, Drawing file, Calculation file, etc.).

SAR CHAPTER 2 - SITE CHARACTERISTICS

Section 2.3.3 Onsite Meteorological Measurement Program

2-1 Provide the following information relative to the meteorology discussed in Section 2.3.3 of the SAR:

(a) Indicate when the onsite meteorological monitoring program was initiated.

(b) A representative sample of the actual data acquired in the onsite meteorological monitoring conducted at the site.

(c) A summary of the data collected from inception of onsite monitoring to the present.

- NUREG–1567 (Section 2.5.3.3), Onsite Meteorological Measuring Program, indicates this information should be included.

RESPONSE

a) The onsite meteorological monitoring program was initiated on December 19, 1996.

b) A representative sample of the actual data acquired in the onsite meteorological monitoring conducted at the site is included in the enclosed disk in two files. The file labeled “METPFSF.XLS” is a MicroSoft Excel spreadsheet containing the calendar year 1997 hourly average meteorological data. The file labeled “METPFSF.Y97” is a file containing the calendar year 1997 hourly average meteorological data in a format similar to that of NRC format. This format is as follows:

Column	Format	Parameter	Units
1 - 6	I6	Identifier	ONSITE
7 - 8	I2	Year	Last two digits
9 - 11	I3	Julian Day	001 - 365
12 - 15	I4	Hour	0100 - 2400
16 - 20	F5.1	Measurement Level	meters
21 - 25	F5.1	Wind Direction	degrees
26 - 30	F5.1	Wind Speed	m/sec
31 - 35	F5.1	Sigma Theta	Degrees

36 - 40	F5.1	Air Temperature	^oC
41 - 45	F5.1	Relative Humidity	percent
46 - 50	F5.1	Barometric Pressure	millibars
51 - 55	F5.1	Precipitation	millimeters
56 - 60	F5.1	Solar Radiation	W/m^2
61 - 65	F5.1	Soil Temperature	^oC

c) A summary of the data collected from inception of onsite monitoring to the present is attached in Table 1. This table provides monthly and annual summaries of the meteorological data in terms of maximum, average, and minimum values. Monthly and calendar year cumulative precipitation amounts are also provided. Wind directions shown as averages are vector averages where the north-south components have been extracted, averaged individually, and an overall direction re-calculated.

TABLE 1

PRIVATE FUEL STORAGE FACILITY
SKULL VALLEY GOSHUTE RESERVATION
Hourly On-Site Meteorological Data
December 19, 1996 to March 27, 1998
Monthly/Annual Summaries

Month		Wind Speed (mph)	Wind Direct. (deg)	Sigma Theta (deg)	Temp (F)	Relative Humidity (%)	Solar Radiation (W/m^2)	Barometri Pressure (mb)	Soil Temp (F)	Monthly Precip (in)	Cum. Precip (in)
1996											
December	Avg	12.1	143.8	11.1	36.7	62.3	55.5	857.1	41.7	0.20	0.20
	Max	26.9		73.3	59.4	95.5	533.2	870.5	42.5		
	Min	2.1		3.4	10.9	25.8	0.0	844.9	40.9		
1997											
January	Avg	8.9	134.1	13.3	27.6	77.4	78.3	861.5	40.4	0.60	0.60
	Max	32.8		64.6	57.1	98.4	536.1	875.6	42.6		
	Min	0.3		0.1	-7.0	38.4	0.0	846.4	38.5		
February	Avg	8.4	6.7	16.4	29.8	72.7	134.8	862.2	38.4	0.44	1.04
	Max	28.9		69.2	55.8	97.5	717.0	872.5	38.9		
	Min	0.3		2.9	6.7	25.0	0.0	839.7	37.8		
March	Avg	9.4	132.3	17.7	40.1	55.7	196.6	861.9	41.4	0.06	1.10
	Max	32.0		75.9	74.9	96.9	823.0	874.9	46.4		
	Min	1.7		2.1	6.4	8.3	0.0	849.7	37.5		
April	Avg	9.7	349.4	18.2	42.9	59.3	228.6	858.0	47.5	1.06	2.16
	Max	32.9		72.0	76.2	96.8	897.0	866.3	50.9		
	Min	1.8		3.0	10.6	12.4	0.0	845.1	45.4		
May	Avg	7.9	4.4	23.3	58.4	50.1	279.7	860.0	55.7	0.60	2.76
	Max	23.5		73.1	94.5	94.0	977.0	866.2	59.6		
	Min	2.0		3.3	21.2	6.9	0.0	851.9	50.4		

TABLE 1 continued

Month		Wind Speed (mph)	Wind Direct. (deg)	Sigma Theta (deg)	Temp (F)	Relative Humidity (%)	Solar Radiation (W/m^2)	Barometric Pressure (mb)	Soil Temp (F)	Monthly Precip (in)	Cum. Precip (in)
June	Avg	10.1	155.5	20.2	67.2	45.9	278.1	856.4	61.7	2.80	5.56
	Max	32.8		72.9	93.4	97.6	988.0	865.3	65.0		
	Min	1.8		0.2	36.5	5.2	0.0	848.3	56.9		
July	Avg	8.5	166.8	23.9	71.7	39.5	287.9	860.9	67.8	0.53	6.09
	Max	30.0		79.6	99.3	96.5	958.0	866.4	70.2		
	Min	1.7		4.2	36.5	3.9	0.0	851.5	65.0		
August	Avg	9.9	156.1	18.6	75.3	39.6	256.5	861.1	69.8	0.78	6.87
	Max	25.5		73.7	96.6	98.1	914.0	868.5	71.1		
	Min	2.1		3.6	49.2	8.9	0.0	850.1	65.1		
September	Avg	8.9	160.4	18.5	63.5	56.6	193.4	861.5	68.0	1.12	7.99
	Max	28.5		70.4	91.4	98.4	817.0	870.1	71.2		
	Min	1.6		2.6	30.3	10.5	0.0	850.2	62.3		
October	Avg	9.8	157.5	16.0	47.9	55.3	153.4	860.8	59.1	0.44	8.43
	Max	32.5		72.7	84.6	96.5	756.0	874.4	63.7		
	Min	1.6		2.5	18.6	11.1	0.0	844.4	53.7		
November	Avg	6.7	142.5	17.9	36.7	72.6	97.8	861.2	49.9	0.34	8.77
	Max	28.4		70.6	66.6	97.8	556.2	873.2	53.7		
	Min	1.1		3.1	9.2	18.1	0.0	841.4	46.7		
December	Avg	7.6	125.9	14.6	21.0	80.8	83.4	864.7	42.7	0.72	9.49
	Max	29.4		60.0	49.0	98.0	481.7	881.8	46.7		
	Min	1.0		2.0	-4.7	33.3	0.0	845.3	38.9		

TABLE 1 continued

Month		Wind Speed (mph)	Wind Direct. (deg)	Sigma Theta (deg)	Temp (F)	Relative Humidity (%)	Solar Radiation (W/m^2)	Barometri Pressure (mb)	Soil Temp (F)	Monthly Precip (in)	Cum. Precip (in)
1997	Avg	8.8	142.4	18.2	48.6	58.7	189.4	860.9	53.6	9.49	
Annual	Max	32.9		79.6	99.3	98.4	988.0	881.8	71.2		
	Min	0.3		0.1	-7.0	3.9	-0.2	839.7	37.5		
1998											
January	Avg	8.7	141.4	14.3	33.8	71.0	86.4	859.0	37.3	0.23	0.23
	Max	28.6		64.4	55.8	97.5	539.3	867.9	38.9		
	Min	1.3		2.7	4.3	20.4	0.0	845.8	35.4		
February	Avg	9.7	139.7	12.4	33.7	75.8	104.4	856.0	39.7	0.52	0.75
	Max	32.0		60.7	54.1	97.9	707.0	867.7	40.2		
	Min	1.0		3.2	4.7	32.8	0.0	841.5	38.3		
March	Avg	8.2	66.7	18.3	38.1	65.0	179.9	859.1	39.6	0.15	0.90
	Max	29.7		68.9	75.3	96.1	778.0	870.4	42.9		
	Min	1.0		1.8	1.3	17.8	0.0	841.7	38.4		

SAR CHAPTER 2 - SITE CHARACTERISTICS

Section 2.4.1.1 Site and Structures

2-2 Justify the conclusions reached regarding stream flows based on water level observations that did not occur during the expected wettest months.

- For the stream channel that drains across Sections 5 and 6 of the site, the stream flow observation period cited was from June 1996 through February 1997. This period does not coincide with the time rainfall is expected to be greatest [i.e., during the months of March, April, and May (according to Table 2.3-3 in the SAR)].

- NUREG–1567 (Section 2.5.4.1), Hydrologic Description, indicates this information should be included.

RESPONSE

We have examined topographic maps of the site area at scales of 1:24,000, 1:100,000 (metric), 1:250,000, and 1:500,000 and have found no streams crossing the site. These maps include both intermittent and perennial streams. The closest stream identified on these maps is an intermittent stream about 1,500 feet northeast of the site, described in SAR Section 2.4.1.1.

While it is true that observations of flow in this stream were not made during a complete 12-month period, the monthly variation in precipitation is very small, generally 0.5 inches or less (see SAR Table 2.3-3 data, eg, 42 year input for Dugway). If these slight differences were significant enough to cause stream flow on a regular basis during March, April, and May, one would expect to find intermittent stream channels developed on the site. Since none exist, the lack of observations is immaterial.

SAR CHAPTER 2 - SITE CHARACTERISTICS

Section 2.5.1 Regional Characteristics

2-3 Provide the following information relative to the withdrawal and use of water on or near the proposed Private Fuel Storage Facility (PFSF):

(a) A map that shows where water withdrawal is occurring on or in the vicinity of the PFSF site with particular reference to the proposed storage pad. At the least, include all wells located within a minimum 8-km (5 mi) radius of the PFSF.

(b) For each identified well-
- Depth to water
- Formation from which water is withdrawn
- Quantity of water withdrawn annually and pumping rates
- Discussion of use of the water from each well with particular reference to any consumption by humans or animals

(c) If no water wells are located within the specified 8-km radius of the proposed PFSF site, include a specific statement such as "No groundwater is extracted within the 8-km (5 mi) radius of the proposed PFSF."

(d) Potentiometric contours of groundwater at and around the proposed PFSF site (if relevant).

(e) Classification of the aquifer beneath the PFSF site based on class of use and water quality (if relevant).
- NUREG–1567 (Section 2.4.5), Subsurface Hydrology, indicates this information should be provided.

RESPONSE

We will obtain available well records placed on-file at the State Engineer's Office since the publication of the report cited in the SAR concerning Skull Valley groundwater (Hood and Waddell, 1968), and provide a more up-to-date summary of groundwater characteristics for the PFSF vicinity. The information to be provided will include a map showing all known wells within 8 km (5 miles) listed in available sources. A table will be prepared that includes the specified details of each well, where available. If sufficient data are available, the potentiometric map of Hood and Waddell (1968) will be modified. Otherwise, their potentiometric map will be reproduced and provided. At present, the State of Utah has not applied any classification system to the aquifer in Skull Valley. The schedule for submitting this additional data is June 15, 1998.

The 1968 Hood and Waddell report contains some of the attributes that are being requested, including a potentiometric map of the entire valley. Unfortunately the data are outdated or incomplete, with no way to tell which wells may still be in use. There are few useful water quality data, no indication of formation or aquifer name, and no data on withdrawals or pumping rates. Location and elevation data indicate that most of the wells within 8 kilometers (5 miles) of the site are northeast of the site and are topographically and stratigraphically above the unconsolidated materials at the PFSF. These wells are, by-and-large, developed in the sand and gravel aquifer that occurs along the base of the Stansbury Mountains. Two exceptions are a stock well 4 miles southwest of the site and a stock well south of Hickman Knolls, about 5 miles from the PFSF. Both of these wells are developed in a deeper part of the valley aquifer. However, neither of these wells would be considered to be on the downstream flow-path from the PFSF and the modest withdrawals expected to occur at the site (5,000 gallons/day [3.5 gallons/minute] during construction and 1,500 gallons/day [1 gallon/minute] during operation) are very unlikely to affect either of these wells at those distances.

The closest well to the site is located at a house on the Reservation, about 2 miles east of the PFSF. No well record was on-file at the State Engineer's Office. It is assumed that the record was not filed because the Reservation is not subject to the State filing requirements. The location of the house would place it on the unconsolidated alluvial fan deposits that border the Stansbury Mountains, near the lower edge. A well, developed at this location, would not likely be negatively affected by a well at the PFSF at the anticipated withdrawal flow rates.

SAR CHAPTER 2 - SITE CHARACTERISTICS

Section 2.6.1 Basic Geologic and Seismic Information

2-4 Provide a column with geologic descriptions summarizing the eastern Great Basin stratigraphy.

- NUREG–1567 (Section 2.4.6.1), Basic Geology and Seismic Information, indicates this information should be included.

RESPONSE

We will comply with the request for a stratigraphic column, to be based on existing published literature sources. The schedule for submitting this additional data is June 15, 1998.

SAR CHAPTER 2 - SITE CHARACTERISTICS

2-5 Justify the declaration that surface features in the PFSF vicinity are not fault-related as reported by Currey (1996) in the SAR.

- Geology of nearby basins (such as the Tooele Basin) suggests that there may be active faults within the interior of similar basins.

Additional information should include the following:

(a) Aerial and field photographs supporting conclusion that the fault scarps identified by Sack (1993) are, in fact, not seismic features but surficial features related to lacustrine processes as reported by Currey (1996) in the SAR.

(b) Low sun angle air photographs showing present land surfaces supporting the conclusion that no fault scarps are found near the PFSF.

(c) Geophysical data (gravity or magnetic maps) supporting the conclusion that no active faults are located in the vicinity of the PFSF.

(d) Discussion providing interpretation of faults shown in Figures 4-1 through 4-5 of the SAR.

- NUREG–1567 (Section 2.4.6.1), Basic Geology and Seismic Information, indicates this information should be included.

RESPONSE

(a) The project will provide copies of available aerial photographs of the site and vicinity. One set, flown in 1996, was used as a basis to prepare topographic maps of the site and access road. The other set was purchased from the USDA and is at a scale of 1:1320, flown in 1993.

(b) At this time, we are not aware of any low-sun-angle photos available for the siting area. Low-sun-angle photos may be performed as part of the program discussed under (d) below; if so, these will be submitted in a supplemental response.

(c) At this time, we are not aware of any gravity or magnetic surveys performed in the siting area. Neither method would be able to differentiate between active and inactive faults, if faults were indicated. As noted in (d) below, the project is performing additional work to supplement its geophysical data to verify that there are no active faults beneath the facility.

(d) We believe that the reference to Figures 4-1 through 4-5 should be Figures 4-6 and 4-7.

The results of the project geophysical investigations suggest that ancient bedrock faults are present beneath the site. These postulated faults have been interpreted to be not active or capable, based on the criteria established by the NRC per 10 CFR Part 100 Appendix A (see Reg. Guide 1.165), of one movement within the past 50,000 years or multiple movements within the past 500,000 years.

The project is planning to perform an enhanced investigative program which will focus on further defining the presence and age of the postulated faults under the siting area. The specific objective of this program will be to augment the previously completed investigative program under the siting area. The program will:

(1) Assure that all faults beneath the siting area have been identified. This element of the program will verify the presence or document the absence of faults in bedrock beneath the site.
(2) Assure that any faults that exist below the siting area are evaluated for capability. This element will characterize the recency, recurrence history, and seismogenic capability (maximum magnitude) of faults beneath the site.
(3) Assure that the interpretation of geologic relationships in the local site area is consistent with the regional geologic setting. This final element of the program will evaluate the structural relationship of faults within the basin to the adjacent range-bounding Stansbury fault, a recognized capable tectonic source.

This program will be executed in the near term and as such, we will defer the answer to this question until the investigation is complete. The schedule for submitting this supplemental information is December 15, 1998.

SAR CHAPTER 2 - SITE CHARACTERISTICS

Section 2.6.1.12 Stability of Foundations for Structures and Embankments

2-6 Provide additional analyses to:

(a) Support the values of allowable bearing pressure quoted for cask storage pads (Section 2.6.1.12.1) and wall footings and spread footings (Section 2.6.1.12.2).

(b) Support the values of total settlement quoted for cask storage pads (Section 2.6.1.12.1) and wall footings and spread footings (Section 2.6.1.12.2).

- Adequacy of soil conditions at the site to support the proposed foundation loading needs to be established using results of site-specific investigations and laboratory analyses [10 CFR 72.102(d)].
- Values of allowable bearing pressure and total settlement were quoted in the SAR without presenting analyses to show how the quoted values were derived from site-specific data on soil properties and load distributions expected from the proposed foundation configurations.
- NUREG–1567 (Section 2.4.6.4), Stability of Subsurface Materials, indicates this information should be provided.

RESPONSE

(a) <u>Allowable Bearing Pressure</u>

Details of the bearing capacity analyses are included in the calculations submitted subsequent to the License Application (PFS letter, Parkyn to Delligatti, 'Submittal of Calculation Package', dated 7/14/98). The calculations addressing bearing capacity include Calculation 05996.01-G(B)-04, "Stability Analyses of Storage Pads", and Calculation 05996.01-G(B)-07, "Allowable Bearing Capacity and Static Settlement of Strip and Square Footings". The analyses within these calculations are based on the results of the site-specific investigations and laboratory analyses that are described in SAR Appendix 2A. These analyses were performed using both total-stress and effective-stress strengths of the foundation soils to check conditions associated with the end-of-construction, as well as for a long term after construction.

The total stress analyses, which are applicable for short-term conditions, immediately after construction, assumed the strength of the soils was equal to the minimum undrained strength obtained in the UU tests. The results of the UU tests are presented in Attachment 2 of Appendix 2A of the SAR. Based on these results, the undrained strength, c, was set equal to 2.2 ksf.

The effective stress analyses, which are applicable for long-term conditions, assumed the drained strength of the soils was based on the friction angle of the soil. The friction angle, ϕ, was estimated using the relationship presented in Figure 18.1 of Terzaghi and Peck (1967) showing ϕ as a function of the plasticity index, PI. The PI of the soil, measured in the laboratory for the silty clay and clayey silt, ranged from 6 to 23, as indicated in Attachment 2 of Appendix 2A of the SAR. The value of ϕ used in these analyses was conservatively estimated to be 30°, based on the high end of this range of PI values, since ϕ decreases with increasing PI.

Allowable Bearing Pressure - Cask Storage Pads

Page 9 of Calculation 05996.01-G(B)-04 presents the results of the bearing capacity analysis of the cask storage pads using the undrained strength (total stress analysis) of the upper layer of silt, silty clay, and clayey silt. It illustrates that the allowable bearing pressure to obtain a factor of safety of 3 for static loads is greater than 4 ksf, as stated in Section 2.6.1.12.1 of the SAR, and it indicates that the actual factor of safety is greater than 6. Page 10 of this calculation presents the results of this analysis using drained strengths (effective stress analysis) of the upper layer of silt, silty clay, and clayey silt. It illustrates that this case is less critical than the end-of-construction case presented above, as the factor of safety against a bearing capacity failure for static loads is greater than 13.

The values of allowable bearing pressure for the storage pads for dynamic loads due to the design earthquake were determined on Pages 15 through 51 of this calculation. In these analyses, the factor of safety against a bearing capacity failure of the storage pads is determined, applying the loading due to the maximum inertial forces of the design earthquake in both the vertical and horizontal directions, as well as in only the horizontal direction. The maximum inertial forces are calculated using a peak vertical acceleration of 0.69g and a peak horizontal ground acceleration of 0.67g.

The analyses on Pages 15 through 19 and in Tables 1 and 2 (Pages 50 and 51) of the calculation are for the case where the storage pad is fully loaded with eight storage casks. The actual bearing pressures under the storage pads are estimated using the effective width and length of the footing to account for eccentricity of the loading and reductions are included to account for the inclination of the loading.

Comparison of the values shown on these tables indicates that the case when the vertical earthquake loads are 0 (Table 2) is more critical than when those forces act downward (Table 1), because of the reduction in capacity associated with the inclination of the loading for this case. Therefore, the values of allowable bearing pressures that are determined in Table 2 of this calculation are presented in SAR Section 2.6.1.12.1.

Allowable Bearing Pressure - Wall Footings and Spread Footings

The allowable bearing pressures of the wall footings and spread footings were developed in Calculation 05996.01-G(B)-07. This calculation uses the same method of analysis and the same geotechnical parameters as are described above for the bearing capacity analyses of the cask storage pads. These analyses were prepared for various combinations of footing widths and depths, for both square footings and strip footings, and curves were developed for use in the design of these footings.

Table 1 of the calculation presents the allowable bearing capacities for strip footings subjected to static loads, and Table 2 presents the allowable bearing capacities for square footings. Tables 3 and 4 present summaries of the allowable bearing capacities for strip and square footings, respectively, to resist dynamic loads due to the design earthquake.

Review of these tables indicates that the static analyses yield the minimum allowable bearing pressures, primarily due to the higher factor of safety required for static conditions. These results are plotted in Figures 5 and 6 of the calculation as the horizontal lines originating from the vertical axis and are included in the design curves presented in SAR Figures 2.6-10 and 1.6-11.

(b) Total Settlement

Details of the settlement analyses are included in the calculations submitted subsequent to the License Application. The calculations addressing settlement include Calculation 05996.01-G(B)-03, "Estimate Static Settlement of Storage Pads", and Calculation 05996.01-G(B)-07, "Allowable Bearing Capacity and Static Settlement of Strip and Square Footings". The analyses within these calculations are based on the results of the site-specific investigations and laboratory analyses that are described in SAR Appendix 2A.

Total Settlement – Cask Storage Pads

Table 3 in Calculation 05996.01-G(B)-03 presents the calculation of settlements of the cask storage pad for the upper layer of silt, silty clay, and clayey silt shown in SAR Figure 2.6-5, including elastic settlements, primary consolidation settlement, and settlements due to secondary compression. These results are combined on Page 21 of the calculation with the estimated elastic settlements for the underlying layers, calculated as discussed on Pages 8 through 12, to obtain the values of total settlement quoted for the cask storage pads in SAR Section 2.6.1.12.1.

Total Settlement - Wall Footings and Spread Footings

The total settlements of the wall footings and spread footings were developed in Calculation 05996.01-G(B)-07. This calculation uses the same methods of analysis and the same geotechnical parameters as were used for the settlement analyses of the cask storage pads. These analyses were prepared for various combinations of footing widths and depths, for both square footings and strip footings, and curves were developed for use in the design of these footings.

Analyses, performed to estimate the expected settlement of various strip and square footings due to various loadings, are included in Appendices C & D of this calculation. Appendix C presents the calculation of allowable bearing pressures to limit the settlement of strip footings to 2 inches and Appendix D presents those for limiting the settlement of square footings to 1.5 inches. These results are summarized in Tables 5 and 6 of the calculation, respectively. They are also presented in Figures 5 and 6 of the calculation, superimposed on plots of the results of the allowable bearing pressure to obtain a factor of safety against a shear failure of 3.0 for static loads. These are the bases of the design curves that are presented SAR Figures 2.6-10 and 2.6-11.

SAR CHAPTER 2 - SITE CHARACTERISTICS

Section 2.6.2 Vibratory Ground Motion

2-7 Provide detailed east-west structural cross-section(s) showing the relationship between the valley bounding structures, including the East Cedar Mountains and Stansbury faults, and stratigraphy primarily to show that the Stansbury fault is the master fault of this basin.

- The cross-section(s) should be drawn to include the entire width of the seismogenic crust.
- The basins in the Basin and Range are typically half-grabens comprised of a master fault and one or more antithetic subordinate faults.
- NUREG–1567 (Section 2.4.6.2), Vibratory Ground Motion, indicates this information should be provided.

RESPONSE

Data is not immediately available to generate the requested cross-section. We plan to perform the following work to respond to the question:

A regional east-west structural cross-section that crosses the site will be constructed to show the relationship between the major range-bounding Stansbury fault on the eastern margin of Skull Valley, antithetic or intrabasin faults within the valley, and fault(s) bounding the western margin of Skull Valley (i.e., the East Cedar Mountains fault). Possible sources of information that will be used to constrain the regional structure of Skull Valley include regional gravity data, such as Johnson and Cook (1957) and Baer and Benson (1987) (the latter appears to include results of a gravity survey across Skull Valley-Ripple Valley, Tooele County); regional COCORP data depending on proximity of line to site; geophysical (seismic) data developed for petroleum explorations, and well data.

Industry experts having knowledge of the region and these data sets (e.g., R. Smith and R. Bruhn at the University of Utah, M. L. Zoback, USGS Menlo Park, CA) will be contacted and interviewed to obtain the most recent data and interpretations. Local research will also be performed to compile and review available data at the University of Utah Department of Geology and Geophysics and State of Utah Geological Survey in Salt Lake City. In addition, the fault characterization data that is developed during the geophysical program, discussed in response to SAR RAI 2-5, will be used, as applicable, in developing the cross-section.

A supplemental response will be prepared that presents the structural cross section(s) and discusses the data used to construct the section. Limitations and uncertainties in the data will be noted where appropriate. Since the cross-section information may be affected by the results of the efforts performed in response to RAI 2-5, the schedule for submitting this supplemental response is December 15, 1998.

SAR CHAPTER 2 - SITE CHARACTERISTICS

Section 2.6.4.7 Response of Soil and Rock to Dynamic Loading and

Section 3.2.10.1 Input Criteria

2-8 Thoroughly analyze the potential for settlement owing to dynamic compaction of the foundation soil considering the high *in situ* void ratio of about 2.0 (porosity of about 67 percent).

- The assessment of dynamic settlement provided in the SAR relies on results of standard penetration tests and unconsolidated-undrained triaxial tests of cohesive soil layers. On the other hand, data presented in the SAR shows a high *in situ* void ratio for the cohesionless soils. Such a high void ratio indicates a material that is "loose" to "very loose", {i.e., relative density smaller than 30 percent [e.g., Figure 22.1 and Table 3.3 of Lambe and Whitman (1969) and Table 6 of Department of the Navy (1982)]}. Because of the high compressibility of such materials, the potential for dynamically induced settlement should be considered more carefully to satisfy the requirement of 10 CFR 72.102(c).

- NUREG–1567 (Section 2.4.6.4), Stability of Subsurface Materials, indicates this information should be provided.

RESPONSE

The in situ void ratio of 1.9 reported in SAR Section 2.6.1.11 for the upper layer of soils in the subsurface profile was determined based on data obtained in performing the consolidation tests. As indicated in the consolidation test results, which are presented in Attachment 2 of SAR Appendix 2A, these tests were performed on samples of the clayey silt. The void ratio of the nonplastic silts was not determined, but based on the standard penetration test (SPT) N-values of the soils, these nonplastic silts would not be characterized as loose.

A review of test results indicated that nonplastic silts were observed in the split-spoon samples obtained above and below Sample U2 in Boring A-2. Therefore, this Shelby tube was opened to see if it contained nonplastic silts that could be tested to determine the void ratio. We found, however, as indicated by the Atterberg limits test results shown on the table below, that this tube contained highly plastic clayey silt. Torvane tests performed on these soils demonstrated that the undrained shear strength ranged from 0.65 to 1.8 tons/ft^2, with an average value of 1.25 tons/ft^2, and the void ratio averaged 2.1. These results are consistent with the test results reported in the SAR for the clayey silt.

Additional Atterberg limits tests were performed on split-spoon samples obtained in Borings A-2, B-3, C-4, and D-4. These results, shown below, confirmed that Samples S3 in Borings A-2 and C-4, and Sample S3A in Boring D-4 were essentially nonplastic. However, these Atterberg limits indicate that Samples S1 in Borings A-2 and B-3 and Sample S2 in Boring D-4, which were described as nonplastic in the Boring Logs, are actually slightly or moderately plastic.

Private Fuels Storage Facility – Skull Valley, Utah
Atterberg Limits Testing Performed in April-May 1998

Boring	Sample	Depth Feet	Water Content %	LL %	PL %	PI %	Plastic
A-2	S1	1.0	15.6	28.9	23.3	5.6	Slightly
A-2	U2C	5.9	52.8	70.2	42.9	27.3	Highly
A-2	U2E	7.0	45.4	61.8	36.7	25.1	Highly
A-2	S3	11.0	18.4	27.0	24.5	2.5	NP
B-3	S1	1.0	8.9	26.6	19.7	6.9	Slightly
C-4	S3	11.0	18.2	26.5	26.0	0.5	NP
D-4	S2	6.0	38.0	49.3	27.7	21.6	Moderately
D-4	S3A	10.2	16.8	24.7	23.3	1.4	NP

A review of the sample descriptions included in the boring logs indicates that only two samples of nonplastic silt are characterized as "loose". These two samples, Samples S-1 in Borings AR-2 and AR-3, were both obtained at the ground surface along the access road. Soils at the ground surface are not of interest since they will be removed during construction. All other nonplastic silt samples for which density is included in the description are characterized as being dense, very dense, or compact.

The following discussion applies to the SPT samples obtained in the upper layer of silt, silty clay, and clayey silt in the areas of the site proposed for the cask storage pads, the Canister Transfer Building, and the Security and Health Physics Building. It excludes the samples obtained at the ground surface, which represent soils that will be excavated for construction of the facilities.

The borings in the vicinity of the proposed locations of the cask storage pads, the Canister Transfer Building, and the Security and Health Physics Building (Borings A-1 through A-4, B-1 through B-4, C-1 through C-4, D-1 through D-4, E-3, and E-4) indicate that the upper layer (~30 ft) consists mostly of soils with some plasticity, especially in the cask storage pad area. The average thickness of nonplastic soils in these borings is ~10 ft. Borings A-2 through A-4, B-1 through B-3, C-1 through C-3, and D-3 have less than or equal to 10 ft of nonplastic soils. Borings A-1 in the northwest, D-1 and D-2 in the northeast, and

B-4, C-4, D-4, and E-4 along the south have ~20 ft of nonplastic soils. Note that these nonplastic soils often include occasional thin layers of clay or slightly plastic silt, which will minimize the potential for dynamically induced settlement. A total of 64 SPT samples of silt (ML) were obtained. Of these, 31 were nonplastic and 33 exhibited some plasticity, ranging from slightly plastic to highly plastic. The N-values for the nonplastic silts in this layer ranged from 11 blows/ft to 40 blows/ft. The median N-value was 18 blows/ft, and the average was 20 blows/ft. This median N-value corresponds to a corrected blow count, N_1, of ~23 blows/ft, based on the relationship between penetration resistance and relative density developed by Gibbs and Holtz (1957) for granular soils.

If the nonplastic silts were cohesionless, they would behave more like fine sands rather than cohesive soils, and based on their N-values, would be classified as very dense rather than loose. Figure 7.5 of Lambe and Whitman (1969) presents the relationship between penetration resistance and relative density developed by Gibbs and Holtz (1957) for granular soils. Using this relationship to estimate the relative density of the non-plastic silts is very conservative, since a decrease in mean grain size tends to cause a decrease in SPT N-value for the same relative density, and the nonplastic silts at the site have a much smaller mean grain-size than the sand and fine sand used by Gibbs and Holtz. Using the 10 psi curve in this figure, or slightly below it, which is the approximate overburden stress for the mid-depth of this layer, fine sands having the median blow count of the nonplastic silts in this layer would be characterized as "very dense", not "loose".

The dynamic settlements of the nonplastic silts in this layer were estimated based on the method presented in Tokimatsu and Seed (1987). As they indicate, for soils above the groundwater table, dynamic settlements are calculated based on procedures originally developed by Silver and Seed (1971), and the effects of multidirectional shaking are estimated based on studies reported by Pyke, Seed, and Chan (1975). The dynamic settlement mechanism is compaction due to grain slip, and it is a function of the magnitude of the cyclic shear strain developed due to the earthquake, the applied number of cycles of this shear strain, and the relative density of the soils.

Figure 13 of Tokimatsu and Seed (1987) presents the relationship between volumetric strain due to compaction, cyclic shear strain, and corrected penetration resistance (N_1) of dry sands for 15 equivalent uniform strain cycles. The cyclic shear strain is estimated based on the average cyclic shear stress due to shaking caused by the Design Earthquake and the shear modulus of the soil. Figure 13 is used to estimate the volumetric strain due to compaction for 15 equivalent uniform strain cycles. Table 4 of Tokimatsu and Seed (1987) is then used to adjust for differences in the number of representative cycles of applied shear stress due to the Design Earthquake (~12 for Magnitude 7) and the 15 cycles used in Tokimatsu and Seed's studies. The dynamic settlement is calculated as the volumetric strain multiplied by the thickness of the nonplastic silts in the layer. Multidirectional effects of the earthquake are addressed by multiplying this result by 2, based on studies reported by Pyke, Seed, and Chan (1975).

The average cyclic shear stress developed in the field due to earthquake shaking is calculated as:

$$\tau_{avg} = 0.65 \bullet a_{max} \bullet \sigma_v \bullet r_d/g = 496 \text{ psf},$$

where: a_{max} = 0.67 g for the Design Earthquake
$\sigma_v = \gamma_{total} \bullet z$ above the groundwater table
γ_{total} = 80 pcf
z = depth below grade
r_d = stress reduction factor, which varies from 1.0 at z=0 to 0.9 at z=30'.

An iterative technique is used to determine the cyclic strain in the field due to the earthquake, γ_{field}. For an assumed value of the cyclic strain, G is calculated as $G_{max} \bullet G / G_{max}$, where G / G_{max} for the nonplastic silt is estimated using the curve for PI=0 presented in Figure 6 of Vucetic and Dobry (1991). G_{max} equals 1400 ksf, as indicated in SAR Table 2.6-1 for Layer 1. The following table presents the results of these iterations.

Private Fuels Storage Facility – Skull Valley, Utah
Determination of Cyclic Shear Strain Due to the Design Earthquake

Iteration No.	$\gamma_{assumed}$ x10^{-4} in./in.	G / G_{max}	G ksf	γ_{field} x10^{-4} in./in.	$\Delta\gamma$ %
1	10	0.250	350	14.2	30
2	15	0.200	280	17.7	15
3	20	0.166	232	21.4	6

The cyclic strain in the field, γ_{field}, is calculated as τ_{avg} / G. Note, it is approximately equal to the assumed cyclic strain for Iteration No. 3; therefore, additional iterations are not required, and γ_{field} is ~21 x 10^{-4} in./in., or 0.21%.

The volumetric strain due to compaction from 15 cycles is estimated as a function of this cyclic shear strain and N_1 of ~23 blows/ft based on Figure 13 of Tokimatsu & Seed (1987). This results in a volumetric strain, $\varepsilon_{c,N=15}$, of 0.17%.

The Design Earthquake is magnitude 7 (SAR Section 2.6.2.3). Table 4 of Tokimatsu & Seed (1987) indicates this corresponds to ~12 cycles of loading and that volumetric strain ratio, $\varepsilon_{c,N=12} / \varepsilon_{c,N=15}$, should be ~0.9. Therefore, the volumetric strain corresponding to the Design Earthquake is $\varepsilon_{c,N=12}$, which is 0.9 x 0.17%, or 0.15%.

$$\varepsilon_c = \frac{\Delta\rho_{dyn}}{\Delta H}$$

where $\Delta\rho_{dyn}$ is the dynamic settlement of the layer, and ΔH is the thickness of the layer.

The thickness of the nonplastic silts in the upper layer is conservatively estimated to be 20 ft, based on the discussion presented above. Therefore, for unidirectional shaking,

$$\Delta\rho_{dyn,1} = 0.36 \text{ inches} = 20 \text{ ft} \times 12 \text{ in./ft} \times \varepsilon_{c,N=12} / 100\%$$

The dynamic settlement is multiplied by 2 to account for multidirectional shaking due to the earthquake. This results in an estimated dynamic settlement of the nonplastic silts in the upper layer of 0.72 inches.

Examination of these soils, which are deposits from ancient Lake Bonneville, indicates the presence of numerous tiny shells (Ostracodes). Considerable void space was present under some of these shells, and it is believed that these voids are contributing to the high, in situ void ratio measured for the clayey silt.

Calcium carbonate is present in these soils, as evidenced by a vigorous reaction upon application of hydrochloric acid to these soils. Therefore, these soils are believed to be cemented, the result of carbonate cement bonding of the silt and clay-size particles, imparting cohesion to these soils.

The reviewer appears to conclude that the nonplastic silts in the upper layer of silt, silty clay, and clayey silt are "loose" to "very loose" based on the void ratio of 1.9 reported in SAR Section 2.6.1.11. However, this void ratio was determined on samples of the clayey silts from the upper layer, not the nonplastic silts, and as evidenced by the SPT data, the nonplastic silts are not loose. The dense nature of these soils, which is most likely the result of carbonate cement bonding of the silt particles, minimizes the potential for dynamically induced settlements due to the Design Earthquake. Ignoring this cementing, the total dynamic settlement is conservatively estimated to be less than ¾ of an inch.

This estimated dynamic settlement was determined based on the thickness of nonplastic silts in areas where the nonplastic silts are thickest, not on an average or median thickness, which conservatively overestimates the settlement. In addition, it conservatively neglects the fact that these nonplastic silts are stratified with layers of clay and clayey silt, which will minimize the potential for dynamically induced settlements. Thus, this estimated dynamic settlement is very conservative.

Dynamic settlements will be much less than this over most of the cask storage pad area, since most of the soils in this area are not nonplastic. Rather, these soils are sufficiently stiff and cohesive that they will not experience dynamic compaction due to the shaking caused by the Design Earthquake.

Dynamic settlements of this magnitude are not expected to adversely affect the performance of the facilities.

References

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CHAPTER 3—PRINCIPAL DESIGN CRITERIA

Section 3.2 Structural and Mechanical Safety Criteria

3-1 Provide design criteria for structures, systems, and components (SSCs) important to safety with respect to lightning strikes. Include intensity and duration of expected strike.

- Section 8.2.9.2 of the SAR states that lightning strikes would not affect integrity of the canister, even though no design criteria are given in Section 3.2 of the SAR.

RESPONSE

A lightning strike would have no adverse consequences on the safety functions of the SSCs that are classified as Important to Safety. Important to Safety SSCs are designed to perform their safety functions if they are struck by lightning. In addition, all PFSF SSCs that are exposed to lightning will be designed to withstand a lightning strike.

In lieu of specific criteria for lightning strike intensity and duration, the following criteria was used to ensure that the casks would adequately accommodate lightning strike current. EPRI EL-5036, Volume 5, "Grounding and Lightning Protection," expressed that structures with at least 5/16" thickness of steel that is electrically continuous, such as tanks, are self-protecting and need no lightning protection. That is, this thickness is sufficient to conduct lightning to ground. The steel shells on both casks is electrically continuous with a thickness in excess of 5/16" (3/4" outer liner and 2" inner liner for HI-STORM, and 2" inner liner for TranStor).

A lightning assessment was performed in accordance with NFPA 780. The results of the assessment showed that the PFSF has a moderate to severe risk of a lightning strike. Factors considered in the assessment were reinforced concrete structures housing operating equipment, tall steel poles, and a flat storage site. The assessment focused on high structures most likely to receive a strike, i.e. the Canister Transfer building, the storage casks, and the 120 ft tall light poles throughout the storage area. The mean annual number of days with thunderstorms for the site location in western Utah, based on the U.S. Meteorological Service isoceraunic map, shown in NFPA 780, was less than 40. Because of the possibility of a strike, the primary risk is the safety of personnel. Lightning protection and facility grounding are part of the Design Basis of the facility. The following sections of the SAR will be revised to include this information.

The text for SAR Chapter 3, Principal Design Criteria will be revised to include the following new section (this will be new Section 3.2.12):

> The design of the SSCs, that are exposed to lightning, i.e., outdoors, shall be designed to withstand the effects of a lightning strike such that a lightning strike will not impair their capability to perform their safety function or result in a radiological release. The light poles and perimeter fences will be connected to the facility grounding system for personnel safety in the event of lightning strikes. The Canister Transfer Building shall be provided with lightning protection in accordance with NFPA 780.

The text for SAR Section 4.2.1.5.1, Structural Design (for HI-STORM) will be revised to include the following new Section K:

> Lightning is addressed in HI-STORM SAR Sections 2.2.3.11 and 11.2.12. The HI-STORM storage system was evaluated for the effects of lightning striking the storage cask. The evaluation determined that when hit with lightning, the lightning will discharge through the steel shell of the storage cask to the ground. The lightning current will discharge through the storage cask and will not affect the canister, which provides the confinement boundary for the spent fuel. Therefore, the HI-STORM design meets the PFSF design criteria in Section 3.2.12 for lightning protection as required in 10 CFR 72.122(b).

The text for SAR Section 4.2.2.5, Structural Design (for TranStor), will be revised to include the following new Section K:

> Lightning is addressed in TranStor SAR Section 11.2.9. The TranStor storage system was evaluated for the effects of lightning striking the storage cask. The evaluation determined that even if a storage cask is hit by lightning, the primary path to ground would be from the steel concrete cask lid to the steel base plate via the steel cask liner and the steel air inlet ducts. The canister is surrounded by these steel structures and therefore, would not provide a path to ground. Therefore, a lightning strike would not affect the canister integrity. Any absorbed heat would be insignificant due to the very short duration of the event. If lightning enters or exits the cask through the concrete shell, some local spalling of concrete could occur, however, it would not be significant enough to affect the cask operation. Therefore, the TranStor design meets the PFSF design criteria in Section 3.2.12 for lightning protection as required in 10 CFR 72.122(b).

The text for SAR Section 4.3.2.2, Safety Considerations and Controls will be revised to include the following:

> In the event of a lightning strike, the most probable target is the 120 foot

tall light poles that provide the lighting for the storage area. The light poles are metal and therefore act as a conductor. The poles are grounded to ensure that the current from a lightning strike is properly conducted to ground.

The text for SAR Section 4.7.1.5.1, Structural Design (for the Canister Transfer Building), will be revised to include the following new Section H:

> The Canister Transfer Building is approximately 77 feet tall and is a possible lightning target. The Canister Transfer Building is designed with lightning protection features in accordance with NFPA 780.

See also changes to accident analysis for SAR Section 8.2.9, "Lightning" under RAI 8-9.

CHAPTER 3—PRINCIPAL DESIGN CRITERIA

Section 3.2 Structural and Mechanical Safety Criteria

3-2 Provide the site-specific evaluation of overturning stability of loaded concrete casks.

- Discussion is contained in SAR, Section 8.2.1, but no details are provided concerning the reference.

RESPONSE

As described in SAR Section 8.2.1, site-specific evaluations of overturning stability of loaded concrete casks were performed by both cask vendor's (Holtec and Sierra Nuclear Corp.) for their storage systems. In addition, an independent overturning stability analysis evaluation of both vendor's loaded concrete casks was performed by J. D. Stevenson, Consulting Engineer.

The text in SAR Section 8.2.1 describes the three analyses. The HI-STORM and TranStor storage casks were analyzed for the PFSF site specific design earthquake (DE), which is based on a seismological evaluation of the siting area as discussed in SAR Section 3.2.10. The DE is represented by response spectrum curves developed specifically for the site with a zero period acceleration of 0.67 g horizontal (two directions) and 0.69 g vertical. Both the HI-STORM and TranStor storage casks were analyzed for these conditions to assure structural strength of the cask and cask stability (Holtec calculations HI-971631 and HI-971574, and SNC calculations PFS 01.10.02.04 and PFS 01.10.02.05, respectively). The cask stability analyses show that the casks will not tip over or slide excessively in an earthquake. The vendor's calculations for the site-specific cask stability analyses were supplied in the calculation package submitted subsequent to the License Application (PFS letter, Parkyn to Delligatti, 'Submittal of Calculation Package', dated 7/14/98).

The analysis performed by J. D. Stevenson independently confirmed the cask stability conclusions of the vendor's analyses (Tipping Evaluation of Spent Fuel Storage Casks Subjected to Site Specific Earthquake Loading (ISFSI DE) for the Private Fuel Storage Facility, Revision 0, June 17, 1997). The J. D. Stevenson report is enclosed for your information and use.

CHAPTER 3—PRINCIPAL DESIGN CRITERIA

Section 3.2.3 Snow and Ice Loads

3-3 Explain the basis for the 10 pounds per square foot (psf) snow load.

- Reference ASCE 7–95 is inadequate to support a conclusion of 10 psf. Figure 7.1, is not sufficiently detailed to justify this load. Site-specific case studies may be warranted for most of Tooele County.

RESPONSE

The site is located very near the 10 psf (4800 ft. elevation) snow load contour on Figure 7-1 of ASCE 7-95, and as such, 10 psf was selected as the design basis snow load. The area outside the 10 psf contour is classified as CS, wherein a site-specific case study is required to establish ground snow loads. Upon further review, and to be conservative, the site location will be redesignated in the CS area.

The Tooele County Building Department was recently contacted to discuss the design snow load for the site. The entire state of Utah is required to have the design snow load established by the building official in accordance with the Uniform Building Code (UBC-97). The County Building Department said, based on the elevation of the Goshute Reservation (4600 to 4700 ft.), a ground snow load of 43 psf would be required to comply with the Uniform Building Code (UBC). The PFSF facility grade is an average elevation of 4465 ft., therefore the value provided by the County is conservative for the site. We propose to roundup the already conservative value of 43 psf to 45 psf for the snow load in lieu of performing an independent site-specific case study.

The 45 psf design snow load is still enveloped by the storage cask design basis snow load which is 100 psf as identified in SAR Section 4.2.1.5.1 (A) and Section 4.2.2.5.1 (A). Furthermore, the detailed design of the Canister Transfer Building will also use a ground snow load of 45 psf.

CHAPTER 3—PRINCIPAL DESIGN CRITERIA

Section 3.2.9 Water Level (Flood) Design

3-4 Justify the statement "all structures, systems, and components that are classified as important to safety are protected from the sheet flow associated with the basin II probable maximum flood by an earthen berm." (see also RAI 3-8)

RESPONSE

The watersheds near the site (Basins I and II) are described in SAR Section 2.4.1.2 and are shown in Figure 2.4-1. Flooding analyses are described in SAR Section 2.4.2.3. Flood design considerations for the facility are described in SAR Section 2.4.2.2. The earthen flood diversion berm is shown in Figure 2.1-2. The flooding accident analysis is described in SAR Section 8.2.3.

Basin II is a fairly small, local watershed, south of the site, which is associated with Hickman Knolls. Due to the gentle and uniform slope of the terrain toward the site, the runoff during the probable maximum flood (PMF) will flow as sheet flow. The depth of peak PMF sheet flow from Basin II is a maximum of 0.7 feet (8.4 inches) across the site (SAR page 2.4-11). The sheet flow will not result in long term standing water at the site since the entire area drains to the center of the expansive Skull Valley. The total duration of time that the PMF flow would be present across the site is calculated to be 14.2 hours (calculation no. 05996.01-G(B)-02). However, the intensity/duration is such that an elevation of 8 inches of water would be present for approximately 0.7 hours and an elevation of 3 inches of water would be present for approximately 3.7 hours.

These conditions would not compromise the safety of the storage casks, since the cask systems are designed to withstand severe flooding and full submergence. The condition of 100% blockage of air inlet ducts due to flooding is described in SAR Section 8.2.8.

The air inlet ducts on the HI-STORM storage casks would remain functional with partial exposure of 3.6 inches during the peak PMF flow level of 8.4 inches. Convective cooling would continue, even though SAR Section 8.2.8.2 shows that the air inlet ducts can be blocked for 92 hours without adverse effects.

The air inlet ducts on the TranStor storage casks would be covered during the peak PMF flow level of 8.4 inches. However, as described in SAR Section 8.2.8.2, the TranStor storage casks are capable of complete blockage of air inlet ducts for an unlimited time. Furthermore, the TranStor SAR Section 12.2.3.1.1 shows the complete blockage of both (air inlet and air outlet) ducts is acceptable for 30 hours.

PMF flows would not compromise the safety of the Canister Transfer Building, since the ground floor elevation will be located above the maximum elevation of the PMF sheet flow.

Nevertheless, it is not a desirable design condition to allow offsite storm runoff to traverse the site. So, in order to maintain a controlled environment onsite, an earthen berm will be constructed along the south and west sides of the facility to divert the PMF flow to the west and then north around the site and into the natural Skull Valley drainage system. A ditch will be provided along the base of the berm to intercept and channel the flow in the desired direction. Both the ditch and the berm are designed for flows associated with the PMF; they are classified as not important to safety. They are provided to minimize stormwater flowing across the site for ease of operations and maintenance activities.

The statement "all structures, systems, and components that are classified as important to safety are protected from the sheet flow associated with the basin II probable maximum flood by an earthen berm" means the sheet flow will not approach the area where these structures, systems, and components are located.

CHAPTER 3—PRINCIPAL DESIGN CRITERIA

Section 3.2.11.4 Canister Transfer Building Load Combination

3-5 Describe how the floor loading of stationary shipping casks, transfer casks, and storage casks have been included in the analysis of the Canister Transfer Building and Canister Transfer Building Foundation.

RESPONSE

The design criteria for the Canister Transfer Building floor and foundation include load combinations that analyze for the worst case placement and weight of the shipping, transfer, and storage casks. The analysis for the Canister Transfer Building floor and foundation will include the heaviest loaded canister configuration and stacked arrangements where the transfer cask is placed on top of the storage and shipping cask, which concentrates a large amount of weight in a small area. This will also include combinations where a transporter carrying a loaded storage cask is near other casks in the Canister Transfer Building.

The text for SAR Section 3.2.11.4.1, Canister Transfer Building Structure, will be revised to include the following:

> After the statement: "Live load shall include crane loads in accordance with ASME NOG-1 positioned to create a worst-case loading condition. All appropriate load combinations identified in ASME NOG-1, as shown in Section 3.2.11.5 herein, shall also be considered in the building design."
>
> The following statement will be added: "Live load shall also include shipping cask, transfer cask, and storage cask loads positioned with loaded canisters to create the worst-case loading on the Canister Transfer Building floor. Load combinations will account for "stacked arrangements" where the transfer cask is placed on top of the storage or shipping cask, side by side placement of the casks in a transfer cell, and when a transporter carrying a loaded storage cask moves adjacent to other loaded casks."

The text for SAR Section 3.2.11.4.2, Canister Transfer Building Foundation, will be revised to include the following:

> Following: "Live load shall include crane loads in accordance with ASME NOG-1 positioned to create a worst-case loading condition. All appropriate load combinations identified in ASME NOG-1 shall also be considered in the foundation design."
>
> Add: "Live load shall also include shipping cask, transfer cask, and

storage cask loads positioned with loaded canisters to create the worst-case loading on the Canister Transfer Building foundation. Load combinations will account for configurations where the transfer cask is placed on top of the storage or shipping cask, side by side placement of the casks in a transfer cell, and when a transporter carrying a loaded storage cask moves adjacent to other loaded casks."

The response to Question 4-0 addresses the RAI about the Canister Transfer Building analysis and provides the schedule for its completion.

CHAPTER 3—PRINCIPAL DESIGN CRITERIA

Section 3.4 Classification of Structures, Systems, and Components

3-6 Justify the classification of the cask transporter as "not important to safety" in Table 3.4.1, and discuss the consequences of its failure.

RESPONSE

In accordance with definitions from 10 CFR 72.3, structures, systems, and components (SSCs) that are classified as Important to Safety function to maintain the conditions required to store spent fuel safely, prevent damage to the spent fuel container during handling and storage, and provide reasonable assurance that spent fuel can be received, handled, packaged, stored, and retrieved without undue risk to the health and safety of the public.

As stated in SAR Section 4.7.5, the purpose of the cask transporter is to transport loaded storage casks between the canister transfer building and the concrete storage pads. The cask transporter is designed with several features that assure safety while transporting spent nuclear fuel. Potential failure mechanisms of the transporter could involve the drive-train, brakes, electrical system, or lift beam hydraulic ram. Of these potential failures, only those that could drop the cask have the possibility of damaging the cask and adversely affecting public health and safety. Because of this, the transporter is not permitted by design to lift a cask above the cask vendor's analyzed safe handling height.

As discussed in SAR Section 8.2.6, the transporter is designed with a mechanical stop that will prevent a cask from being raised above 10 inches, to be within the vendor's height limits of 10 inches and 18 inches (Reference HI-STORM SAR Section 12.3.14 and TranStor SAR Section 12.2.2.8, respectively). This Section also includes a discussion of the consequences of a drop in excess of the vendor's analyzed drop height and concludes that the cask would retain its confinement integrity and that there would be no release of radioactivity material or loss of shielding. Also, facility procedures will limit the lift height of a loaded storage cask to approximately 4 inches.

In addition, a Technical Specification is proposed to ensure that the casks will not be lifted above the vendor's analyzed safe handling height (See Technical Specifications or SAR Section 10.2.1.3). Therefore, a failure of the cask transporter will not damage the spent fuel storage system or adversely affect the health and safety of the public, which is the basis for the transporter classification as Not Important to Safety.

CHAPTER 3—PRINCIPAL DESIGN CRITERIA

Section 3.4 Classification of Structures, Systems, and Components

3-7 Justify the classification of the closed circuit television, radiation monitors, and temperature monitoring as not important to safety, and discuss the consequences of their failure.

- NUREG–1567 (Section 4.4.5), Operation Support Systems, states that the SAR should address a basis for determination that the regulatory requirements [10 CFR 122(l)] for instrumentation and control systems are under accident-level conditions.

RESPONSE

In accordance with definitions from 10 CFR 72.3, structures, systems, and components (SSCs) that are classified as Important to Safety function to maintain the conditions required to store spent fuel safely, prevent damage to the spent fuel container during handling and storage, and provide reasonable assurance that spent fuel can be received, handled, packaged, stored, and retrieved without undue risk to the health and safety of the public.

The function of the closed circuit television (CCTV) is to provide assessment of unauthorized penetration within the protected area as required per 10 CFR73.51 (proposed), "Requirements for the Physical Protection of Stored Spent Nuclear Fuel or High-Level Radioactive Waste," and NUREG-1497, "Interim Licensing Criteria for Physical Protection of Certain Storage of Spent Fuel." However, as stated in NUREG-1497, "adequate assessment may also be provided through onsite assessment by a watchman if an acceptable justification of timely assessment can be provided." A failure of the CCTV system would be discovered immediately by security personnel as indicated by a loss of continuously observed surveillance capabilities. Appropriate compensatory measures would then be initiated, eg, sending security personnel to CCTV observation locations to provide timely onsite surveillance.

As discussed in SAR Sections 5.1.6.4 and 7.3.5, the PFSF will utilize various types of radiation monitors including area monitors, thermoluminescent dosimeters (TLD), portable hand held monitors, personnel dosimetry, and portable airborne monitors. The purpose of the area radiation monitors is to detect and alarm high radiation conditions in the canister transfer building. The purpose of TLDs is to record radiation doses received at the radiation area boundary, owner controlled area boundary, and by PFSF personnel. The purpose of the portable hand held monitors is to provide surveillance of radiation levels near worker locations during transfer operations. The purpose of the personnel dosimetry, which is worn by all workers in the canister transfer area, is to measure worker accumulated dose while in the transfer area. The purpose of the portable airborne monitors is to ensure that, even though the canisters are sealed, no airborne radioactivity is present during transfer operations. Radiation monitors are not classified as Important to Safety since they are not needed to prevent or mitigate any credible accident that would adversely affect public health and safety. The use and presence of various types of monitors during facility operations provides defense in depth and will ensure that even if one fails, other monitors would detect high radiation conditions and alarm to provide safe working conditions for onsite personnel.

As discussed in SAR Section 5.1.4.4, the purpose of the temperature monitoring system is to provide continuous surveillance of each cask's temperature to ensure proper operation. In the event of a temperature monitor failure, the monitoring computer would not receive a signal. This would create an alarm informing personnel of a potential cask temperature problem. A temperature monitor system failure would alarm in the security monitoring area and security personnel would contact operations personnel. As noted in SAR Section 8.2.8, under worst case conditions, cask temperature increases occur over several days, which would give operation personnel ample time to assess and resolve the problem.

Therefore, the CCTV, radiation monitors, and the temperature monitoring system are not needed to maintain spent fuel storage safety, to prevent damage to the spent fuel container, or to preclude conditions that would adversely affect the health and safety of the public. Therefore, these SSCs are classified as Not Important to Safety.

CHAPTER 3—PRINCIPAL DESIGN CRITERIA

Section 3.4 Classification of Structures, Systems, and Components

3-8 (a) Provide and justify the safety classification of the flood-control berm.

(b) Discuss the consequences of its failure in relationship to the accident analysis provided in the SAR, Section 8.2.3.2.

RESPONSE

(a) The flood control berm is classified as not important to safety. Flooding due to PMF would not compromise the safety of the storage casks or the Canister Transfer Building if the berm were not installed or if it failed. The berm is provided to minimize stormwater flowing across the site for ease of operations and maintenance activities.

As described in response to RAI 3-4, the depth of peak PMF sheet flow from Basin II is a maximum of 0.7 feet (8.4 inches) across the site (SAR page 2.4-11). The sheet flow would not result in long term standing water at the site since the entire area drains to the center of the expansive Skull Valley. The total duration of time the PMF flow that would be present across the site is 14.2 hours per calculation no. 05996.01-G(B)-02. However, the intensity/duration is such that an elevation of 8 inches of water would be present for approximately 0.7 hours and an elevation of 3 inches of water would be present for approximately 3.7 hours. These conditions would not compromise the safety of the storage casks, since the cask systems are designed to withstand severe flooding and full submergence.

The condition of 100% blockage of air inlet ducts due to flooding is described in SAR Section 8.2.8. The air inlet ducts on the HI-STORM storage casks would remain functional with partial exposure of 3.6 inches during the peak PMF flow level of 8.4 inches, even though SAR Section 8.2.8.2 shows that the inlet ducts can be blocked for 92 hours without adverse effects. The air inlet ducts on the TranStor storage casks would be covered during the peak PMF flow level of 8.4 inches, but as described in SAR Section 8.2.8.2, the TranStor storage casks are capable of complete blockage of air inlet ducts for an unlimited time. Furthermore, the TranStor SAR Section 12.2.3.1.1 shows the complete blockage of both air inlet and outlet ducts is acceptable for 30 hours. PMF flows are mitigated in the Canister Transfer Building by locating the ground floor elevation above the maximum elevation of flood water.

(b) Failure of the flood control berm would not have an adverse effect on the structures, systems, and components that are classified as important to safety. As stated in the response to part (a) above, the maximum depth of sheet flow would be 0.7 feet (8.4 inches) across the site if the berm is not accounted for. This condition would not compromise the safety of the storage casks, since the casks are designed to withstand severe flooding and full submergence. The Canister Transfer Building would also be unaffected, as the floor elevation will be located above the maximum elevation of flood water. Forces due to flowing water would be insignificant due to its shallow depth.

CHAPTER 4—INSTALLATION DESIGN

General

4-0 Provide additional detail regarding the results of the structural analysis of the design of the Canister Transfer Building (Section 4.7.1), and Canister Transfer Cranes (Section 4.7.2).

RESPONSE

The project plan for the design of the PFSF is composed of two phases, a preliminary and detailed design effort. The preliminary design phase of the project developed and finalized the design criteria for the facility and is the basis for the License Application submitted for review under Part 72. The detailed design effort will prepare the necessary drawings and specifications (procurement and installation) for construction of the facility which under the present schedule will be starting in the near term. These two phases are more fully described below in terms of what has been completed to date and the present schedule for completion of the elements applicable to the information requested.

The first phase provided the necessary conceptual design drawings and complete design criteria to support the preparation of the License Application for review. The conceptual design of the Canister Transfer Building, including the Canister Transfer Cranes, is shown in SAR Figure 4.7-1 (3 Sheets). The design criteria for the Canister Transfer Building is described in SAR Section 3 and summarized in SAR Table 3.6-1. The methodology and reference standards to be used in the building seismic analysis under the detailed design phase of the project are described in SAR Section 3.2.10. Load combinations for the building design are shown in SAR Section 3.2.11.4. The design criteria for the Canister Transfer Building cranes is described in SAR Section 3 and summarized in SAR Table 3.6-1. Reference standards and load combinations for the crane design are shown in SAR Section 3.2.11.5. Additional information on the design and functional requirements of the Canister Transfer Building and Canister Transfer Cranes is contained in SAR Sections 4.7.1 and 4.7.2, respectively.

The second design phase of the project is to perform the detailed design calculations and drawings for the Canister Transfer Building and Canister Transfer Cranes. Detailed design of the Canister Transfer Building will include the preparation of a 3-dimensional computer model using the ANSYS computer program. Soil-structure interaction will be considered in the design by including the effects of the soil properties established during the site-specific geotechnical investigation program and as represented by discrete soil springs or a finite element layered system as described in ANSI/ANS 57.9. The computer model will be subjected to the loads and load combinations described in SAR Section

3.2.11.4, including dead loads, live loads, crane loads, wind loads, tornado loads, seismic loads, thermal loads, and accident loads. The seismic design will be based on the site-specific design response spectra curves anchored at 0.67 in two horizontal directions and 0.69 in the vertical direction. Seismic analysis methods used will be in accordance with standard methods as described in ANSI/ANS 57.9, NUREG-0800, and ASCE-4. The seismic response of the structure will be determined by calculating the response of the computer model to the prescribed seismic input. Once the results of all load combinations are computed, the structural members and foundations will be designed.

The detailed design of the Canister Transfer Cranes will be performed in accordance with ASME NOG-1 as Type I cranes. A Type I crane is designed and constructed so that it will remain in place and support the load during and after a seismic event and includes single failure proof features so that any credible failure of a single component will not result in the loss of capability to stop and hold the load. To ensure these qualities, ASME NOG-1 provides strict requirements be met that include:

- Dynamic analysis (e.g., response spectrum or time-history) to establish the response of the crane to a seismic event,
- Mathematical model represented by a three-dimensional system of nodes with a geometry that reflects the overall size, length connectivity, and stiffness of the various structural members: bridge girders, trolley frame, gantry legs, end ties, end trucks, and hoist components.
- Full range of load combinations that ensure that the crane components will perform as required for all operational loads, severe and extreme environmental loads, and abnormal event loads. These combinations are presented in SAR Section 3.2.11.5.
- Fatigue analysis for the critical parts of the mechanical components to ensure the fatigue strength and fatigue life of the components will meet the requirements for the service life and reliability of the crane and that no failure of any component will result in an uncontrolled movement of the load.
- 100% radiographic test or ultrasonic test of buttwelds per AWS 1.1, 10% magnetic particle test or dye penetrant test per AWS 1.1 of each weld that exceeds 10 in. in length, and ultrasonic volumetric tests of plate, wrought or forged material.
- Structural materials that are impact tested in accordance with the code.
- Proof load test of hooks per ASME/ANSI B30.10 and wire rope breaking strength test per Federal Specification RR-W-410.
- No load test to verify proper operation of all electrical, lubrication, instrumentation, and control systems.
- Full load test with 100% load placed mid-span and verification that all crane component operations perform as required including verification that each holding brake will stop and hold the load as required.

- Rated load test with 125% load hoisted the full range of movements to ensure the load can be supported by the crane and held by the hoist brakes.

The PFSF is committed to ensure these ASME NOG-1 qualities are incorporated into the crane designs.

The project plan has the bid specification for the cranes issued by June 1, 1998 and award a contract for the engineering/design by July 1, 1998. The selected crane supplier will be released to begin their dynamic analysis and design and to furnish certified loadings to be used as an input to the building structural analysis. It is anticipated the structural calculations and drawings for both of the cranes will be available for submittal by December 15, 1998. The Canister Transfer Building structural analysis will begin on June 1, 1998 and will conclude following the receipt and use of the crane loads. The schedule for submitting the structural calculations and drawings for the Canister Transfer Building is December 15, 1998.

CHAPTER 4—INSTALLATION DESIGN

Section 4.2.3 Cask Storage Pads

4-1 Provide the supporting analyses for the results given in Tables 4.2-7 and 4.2-8. Include discussion of assumptions, procedures, and results for shear deformation, bearing loads, etc.

RESPONSE

The supporting analyses for the cask storage pad design are contained in SAR Chapter 4, Reference 16 (Pad Analysis and Design Calculation No. SC(PO 17)-1, Revision 1). The calculation was supplied in the calculation package submitted subsequent to the License Application (PFS letter, Parkyn to Delligatti, 'Submittal of Calculation Package', dated 7/14/98).

The values summarized in SAR Tables 4.2-7 and 4.2-8 are obtained from the calculation on pages 77 and 178. Two of the values for maximum soil pressure in Table 4.2-7 (1.67 and 1.86 KSF) were conservatively derived in the calculation, as shown on page 159.

SAR Section 4.2.3 contains a summary of the pad analyses, including a description of the computer model and programs used in both the static and dynamic analyses. Major assumptions, procedures, and results are described in this Section.

The soil properties used in the analyses to account for soil structure interaction are contained in calculation No. 05996.01-G(PO5)-1 (Development of Soil and Foundation Parameters in Support of Dynamic Analysis) which was provided in the calculation package submitted. The allowable soil bearing criteria is contained in SAR Section 2.6.1.12 and the PFSF Geotechnical Design Criteria (provided to the CNWRA on 1/31/98 in response to the 12/23/97 NRC request for references).

CHAPTER 4—INSTALLATION DESIGN

Section 4.7.2 Canister Transfer Cranes

4-2 (a) Provide the detailed design analyses for the overhead and semigantry cranes that demonstrate they meet the criteria specified in ASME NOG-1.

(b) Provide the basis for the conclusion stated in SAR, Section 4.7.3.5.1(d), that it is assumed that "the crane would be connected to the cask throughout the transfer operation and therefore prevent the cask from toppling during a seismic event."

RESPONSE

(a) The structural analysis and detailed design of the Canister Transfer Cranes have not yet been completed. The cranes will be designed in accordance with ASME NOG-1 as Type I cranes. A Type I crane is designed and constructed so that it will remain in place and support the load during and after a seismic event and includes single failure proof features so that any credible failure of a single component will not result in the loss of capability to stop and hold the load. The PFSF is committed to ensure these ASME NOG-1 requirements are incorporated into the crane designs. The requirements for the cranes and a schedule for awarding the crane design contract and submittal of the crane analyses to the NRC for review is discussed in the response to NRC question 4-0.

(b) The text was not intended to convey that it was only an assumption that the crane would be connected to the cask throughout the transfer operation, but that because of a potential earthquake, it would be necessary to maintain this connection. The SAR will be revised to show there is no condition when the HI-TRAC transfer cask addressed in Section 4.7.3.5.1(d) is unsupported while on top of a shipping or storage cask. This means ensuring, through procedures, that regardless of the crane connection, the transfer cask will be secured to the cell walls with support struts as discussed for the same condition with the TranStor transfer cask in Section 4.7.4.5.1.D.

The text for SAR Section 4.7.3.5.1(D), Earthquake, will be revised to read as follows:

> The transfer cask has been evaluated for stability during a seismic event when in the stacked cask arrangement. The stacked cask arrangement occurs when the transfer cask is resting on top of the storage cask. It was concluded that during transfer operations, it is

necessary to ensure the transfer cask is supported throughout the transfer operation to prevent the cask from toppling during a seismic event. Therefore, facility procedures will ensure that the transfer cask be secured to the cell walls with struts when in the stacked arrangement to preclude a cask toppling accident.

The text for SAR Table 3.4-1, QA Classification of SSCs, will be revised to show that the Support struts are classified as Important to Safety, Category B.

The text for SAR Section 5.1.4.2, Table 5.1-1, and Figure 5.1-1 will be revised to show attachment of the seismic support struts to the HI-TRAC transfer cask.

CHAPTER 5—OPERATION SYSTEMS

Section 5.0 Operation Systems

5-1 In conformance with 10 CFR 72.44(c), provide the technical specifications (required per 10 CFR 72.24) for the SSCs categorized in Table 3.4–1.

- This is also recommended in NUREG–1567 (Section 4.4.2) whose use is described in Sections 5.1 through 5.6 and referenced in Section 10.2.5.

- NUREG–1567 (Section 4.4.2) states the design and design analysis for structural capabilities should be included for fuel handling SSCs important to safety. The cranes integral to the facilities and rigging (including attachments, wire ropes, spreaders, and hooks) are specifically identified.

RESPONSE

The attached matrix lists the SSCs categorized in Table 3.4–1 as Important to Safety. For each SSC a reference is provided for location of the following information in the PFSF SAR: Design Criteria, Design Description, Normal Operation, Operating Controls and Limits, and Surveillance Requirements. It should be noted that rigging (including attachments, wire ropes, spreaders, and hooks) are included with Associated Lifting Devices in Table 3.4-1.

The proposed Technical Specifications required by 10 CFR 72.44 (c) are provided in Appendix A of the License Application.

The information required by 10 CFR 72.24 (g) (an identification and justification for the selection of those subjects that will be probable license conditions and technical specifications) is included in SAR Chapter 10, Operating Controls and Limits.

In addition to the surveillance requirements provided in SAR Section 10.2.3, the PFSF will develop and implement an Equipment Maintenance and Testing Program prior to initial operation. All equipment designated as Important to Safety will have routine maintenance, inspections, and testing performed in accordance with approved written procedures. The program will include the requirements of the applicable equipment codes, regulatory requirements, vendors requirements/recommendations, standard industry practices, and any specific PFSF requirements deemed necessary. The goal of the program will be to ensure that all equipment that is Important to Safety is available to perform its intended function in a safe and reliable manner. Key elements will include operability, calibration, surveillance testing, post maintenance or modification testing as well as maintenance trending.

SSCs Important to Safety Location of Information					
SSCs	**Design Criteria**	**Design Description**	**Normal Operation**	**Operating Controls and Limits**	**Surveillance**
Spent Fuel Canister	SAR Chapter 3	SAR Chapter 4 Section 4.2.1/4.2.2	SAR Chapter 5	SAR Chapter 10 Section 10.2.1.2/10.2.2.1	SAR Chapter 10 Section 10.2.3
Concrete Storage Cask	SAR Chapter 3	SAR Chapter 4 Section 4.2.1/4.2.2	SAR Chapter 5	SAR Chapter 10 Section 10.2.1.3/10.2.1.6/ 10.2.2.2/10.2.2.3/ 10.2.2.4	SAR Chapter 10 Section 10.2.3
Transfer Cask	SAR Chapter 3	SAR Chapter 4 Section 4.7.3/4.7.4	SAR Chapter 5	SAR Chapter 10 Section 10.2.1.4/10.2.1.5	Maintenance Program
Associated Lifting Devices	SAR Chapter 3	SAR Chapter 4 Section 4.7.3/4.7.4	SAR Chapter 5	None	Maintenance Program
Canister Transfer Building	SAR Chapter 3	SAR Chapter 4 Section 4.7.1	SAR Chapter 5	None	Maintenance Program
Canister Transfer Overhead Bridge Crane	SAR Chapter 3	SAR Chapter 4 Section 4.7.2	SAR Chapter 5	None	Maintenance Program
Canister Transfer Semi-gantry Crane	SAR Chapter 3	SAR Chapter 4 Section 4.7.2	SAR Chapter 5	None	Maintenance Program
Seismic Support Struts	SAR Chapter 3	SAR Chapter 4 Section 4.7.1.4.1	SAR Chapter 5	None	Maintenance Program
Cask Storage Pads	SAR Chapter 3	SAR Chapter 4 Section 4.2.3	SAR Chapter 5	SAR Chapter 10 Section 10.2.1.6	Maintenance Program

CHAPTER 5—OPERATION SYSTEMS

Section 5.2.1.2 Spent Fuel Canister Handling

5-2 Demonstrate (including design and design analyses) that tools and gripping devices not specifically identified in cask specific SARs, have:

(a) Adequate margin of safety to prevent unacceptable damage to the shipping cask, canister, or storage cask during normal, off-normal, and accident conditions.

(b) Adequate control to prevent damage to the shipping cask, canister, or storage cask during normal, off-normal, and accident conditions.

RESPONSE

The only gripping devices and tools listed in Section 5.2.1.2 that are not identified in the vendors SARs are the Overhead Bridge Crane, the Semi-gantry Crane, Seismic Support Struts, and the Cask Transporter.

(a) As discussed in SAR Section 3.2.11.5, the canister transfer cranes (overhead bridge crane and the semi-gantry crane) shall be classified as Type I cranes in accordance with ASME NOG-1 since the cranes are used to handle critical loads. A Type I crane is defined as a crane that is designed and constructed to remain in place and support a critical load during and after a seismic event and has single-failure-proof features such that any credible failure of a single component will not result in the loss of capability to stop and/or hold the critical load. A critical load is defined as any lifted load whose uncontrolled movement or release could result in potential offsite radiation exposure. The single-failure-proof crane design shall meet the requirements of NUREG-0554, NUREG-0612, and ASME NOG-1.

As discussed in SAR Section 4.7.2, the design of the canister transfer cranes will be performed during the detailed design phase of the project. Detailed design of the cranes will be performed by the crane vendor. During the detailed design stage, design requirements will be specified that provide for the performance of testing, inspection, and maintenance activities on the cranes in accordance with 10 CFR 72.122(f). Inspection and acceptance of the cranes will be performed during fabrication, in accordance with the QA Program described in SAR Chapter 11, to ensure that the design requirements are satisfied.

Strict adherence to the design, testing, inspection, and maintenance criteria as noted above will ensure adequate safety margins are provided to prevent damage to the shipping cask, canister, or storage cask during normal, off-normal, and accident conditions. Specific design criteria for the cranes, a schedule for awarding the crane design contract, and submittal of the crane analysis to the NRC for review is discussed in the response to NRC RAI 4-0.

As discussed in revised SAR Section 4.7.1.4.1, the seismic support struts are classified as Important to Safety. The struts are designed to secure the shipping cask and transfer cask to the Canister Transfer Building transfer cell walls during transfer operations. In order to perform the transfer operation, the transfer cask will be placed on top of a shipping cask and storage cask.. In this position, struts are required to prevent the transfer cask from toppling during an earthquake. Each cask will utilize 2 struts, which will provide restraint in the x and y directions. The struts will be standard rigid support assemblies that conform to ASME III, NF requirements for Class 2 nuclear grade supports. As such, the struts will be subject to QA requirements per 10 CFR 50, Appendix B; material certification, design, and NDE per ASME III NF; and welder and weld qualifications per ASME IX.

As discussed in SAR Section 4.7.5, the cask transporter will be a commercial grade system and will be classified as not Important to Safety. Facility procedures will limit the lift height of a loaded storage cask to approximately 4 inches. A failure of any cask transporter components will not result in any safety concerns since the cask would only lower 4 inches back to the ground. Drops this small are within analyzed accident conditions presented in Section 8.2.6. The transporter will be designed to mechanically limit the lifting height of a canister to a maximum of 10 inches. The hydraulic lift cylinders will be equipped with double locking valves and a cam locking system engages and holds the load in the event a cylinder looses holding power. Indicator lights on the operating console tell if the cams are disengaged or engaged. Markings on the lift boom and a meter on the operating console give indication of the lifted height.

(b) Strict adherence to the design, testing, inspection, and maintenance criteria noted in (a) will also ensure the canister transfer cranes are provided with adequate controls to prevent damage to the shipping cask, canister, or storage cask during normal, off-normal, and accident conditions. The crane designs will include limit switches for prevention of bridge, trolley, and hook travel beyond safe operating positions, limits on bridge, trolley, and hook travel speeds, and provisions for lowering a load in the event of an overload trip. Periodic inspection and testing will be performed to keep the cranes certified to ASME NOG-1.

Additionally, facility operating procedures will provide instructions for all routine and projected contingency (off-normal) operations, including handling, loading, transporting, and storing of spent fuel, and for all other operations important to safety. Chapter 10 of the SAR provides a discussion on the required operating controls and limits. The requirements for certification of personnel operating equipment and controls Important to Safety will also be specified in the operating procedures.

The following SAR Chapter 8 scenarios demonstrate that damage is prevented to the shipping cask, canister, or storage cask during off-normal conditions. Section 8.1 of the SAR discusses two off-normal events associated with canister transfer operations: loss of electrical power and operator error. A total loss of external AC electric power is postulated to occur during canister transfer operations as the result of a disturbance in the offsite electric supply system. The operator error event consists of off-normal operator load handling errors that develop from the canister impacting against the inside of the shipping, transfer, or storage cask. Analysis of these events indicates that there are no resulting adverse safety or radiological consequences.

A hypothetical storage cask drop and/or tipover accident is analyzed in SAR Section 8.2.6.2 that would bound any potential accident associated with the transporter. Based on the results of this analyses, the cask/canister storage systems would retain their confinement integrity and there would be no release of radioactivity and no resultant doses in the event of hypothetical drop/tipover of a fully loaded storage cask.

CHAPTER 6—SITE-GENERATED WASTE CONFINEMENT AND MANAGEMENT

Section 6.4 Solid Waste

6-1 Describe the confinement, handling, and disposition used for solid waste generated in the course of using the transfer cask [NUREG–1567 (Section 6.5.5.2)].

RESPONSE

State-of-the art solid radwaste handling equipment and procedures will be used in handling any solid waste generated at the PFSF. The following is an example of the process. Final equipment will be selected and final procedures developed as the facility gets closer to operation.

Solid waste, that may be generated during canister transfer operations (including use of the transfer cask), such as smears, cloth rags, wipes, tape and similar decontamination materials, will be placed inside poly bags (yellow) that are inserted into 55-gallon drums. The poly bags will be placed so as to provide a clean surface for personnel to lift up and around to seal the material inside the bags. When the material is placed inside the bags the exposed surface will be tested (smeared) to insure that no loose surface contamination is present. To further insure that loose contamination is not transferred to the exterior of the drum, blotting material will be placed under the drum while material is placed into the poly bags. The poly bag will be double sealed in a reverse fashion whereby the bag is twisted and sealed then the sealed area is turned 180 degrees and sealed again.

The external surface of the 55-gallon drum will be smear tested to ensure no loose surface contamination is present prior to being transferred to a disposal facility. The drum will also receive a radiation survey to ensure that the radiological limits for transfer are met.

Protective clothing used during the decontamination efforts will be removed in a controlled area where there are placed sticky "step off pads" to minimize the potential for transfer of loose surface contamination to the surrounding areas. In this case the initial "step off pad" will be considered as "dirty" in a reverse fashion of commercial industry practices. However, additional "step off pads" will be available and appropriately marked to ensure a clean surface for personnel to exit the area. Training will ensure that personnel are knowledgeable of the difference in the practice and are capable of exiting the area without transferring contamination.

Used protective clothing will be placed in poly bags inside 55-gallon drums similar to the waste material. The handling of these drums will be performed in a similar fashion but will be transferred to a laundry facility for the cloth clothing and a waste facility for the disposable clothing.

The volume of solid waste is expected to be minimal since the occurrence of contamination would be due to an off-normal event. Due to limited expected volume of waste material, provisions are not considered necessary for the volume reduction of waste. However, waste materials will be separated at the source by use of separate containers for waste materials and protective clothing. The waste materials are not expected to require immobilization or change in composition since the expected materials are soft cleaning items that will not require these processes.

Full waste containers will be stored in the Low Level Waste Storage Room in the Canister Transfer Building. The concrete walls and ceiling of this room will provide shielding for the stored waste. This room will be considered a controlled area with restricted access. The use of a separate restricted storage area with concrete wall for shielding will maintain any exposures in the area ALARA. Waste material inside the drums is Low Level and is not expected to require the use of additional shielding materials around the drums.

CHAPTER 7—RADIATION PROTECTION

Section 7.2.1 Characterization of Sources and

Section 7.3.3.6 Dose Rates at Distances from the PFSF Array of Storage Casks and

Section 7.4 Estimated Onsite Collective Dose Assessment

7-1 Justify not using the bounding values for the assumed enrichment, burnups, and cooling times that describe the fuel for the calculation to show that the dose to workers will be less than the limits in 10 CFR 20.1201 and the dose to the off-site public will be less than the limits in 10 CFR 72.104.

The following specific assumptions should be justified:

(a) The assumption on page 7.2-2, first paragraph, fourth sentence, noting the assumed enrichments [3.7 percent for pressurized water reactor (PWR) fuel and 3.4 percent for boiling water reactor (BWR) fuel] are lower than the average enrichments normally used to obtain the burnups analyzed.

(b) The assumption in Section 7.3.3.5 stating the assumed burnup of 40 GWd/MTU represents a conservative burnup for a majority of the fuel stored at the PFSF.

- This is less than the maximum burnup for fuel that will be accepted (See reference in Section 10.2.1.1).

(c) The assumption in Section 7.4 showing the assumed burnup (35 GWd/MTU) and cooling time (20 yr) as indicative of the calculation of dose to workers during receipt and transfer operations.

- These values are not consistent with the burnup and cooling times assumed for the calculation of dose to the off-site public (40 GWd/MTU burnup and 10-yr cooling time) in Section 7.3.3.5.

RESPONSE

(a) The basis for this assumption is Section 5.2.2 of the HI-STORM SAR, which states the following:

"It is well known that the neutron source strength increases as enrichment decreases for a constant burnup and decay time. This is due to the increase in Pu content in the fuel which increases the inventory of other transuranium nuclides such as Cm. The gamma source also varies with enrichment, although only slightly. Because of this enrichment effect, and in order to obtain conservative source terms, an initial fuel enrichment of 3.4 w/o ^{235}U was chosen for the BWR design basis assembly, and 3.7 w/o ^{235}U for the PWR design basis fuel assembly. These enrichments are below the normal average enrichments used to obtain the burnups analyzed. Therefore, the source term calculations are conservative."

Also the basis for the assumption is supported by DOE Report DOE/RW-0184-R1, "Characteristics of Potential Repository Wastes, prepared for DOE's Office of Civilian Radioactive Waste Management (OCRWM), July 1992. Table 2.4-1 of this report provides three enrichments for each burnup specified, a low value, a mid-range value, and a high value. Section 2.4.2 of this report states that "The mid-range enrichment was determined from a regression on EIA historical and projected data and the enrichments used encompass most fuels of interest." Table 2.4-1 is broken into standard burnup cross sections and high burnup cross sections for both BWR and PWR fuels, and the high burnup cross section data is applicable to the design basis fuels which Holtec selected for the HI-STORM storage cask. The design basis fuels are identified in Section 7.2.1 of the PFSF SAR as 45 GWd/MTU, 5-yr cooled and 47.5 GWd/MTU, 6-yr cooled PWR fuel; and 45 GWd/MTU, 5-yr cooled BWR fuel. Table 2.4-1 of the OCRWM report identifies mid-range enrichments of 3.44% for 40 GWd/MTU and 3.74% for 50 GWd/MTU burnup BWR fuel. Therefore, Holtec's selection of 3.4% enrichment for their 45 GWd/MTU burnup design basis BWR fuel is seen to be below the average (mid-range) enrichment. Likewise, Table 2.4-1 of the OCRWM report identifies mid-range enrichments of 3.72% for 40 GWd/MTU and 4.26% for 50 GWd/MTU burnup PWR fuel. Holtec's selection of 3.7% enrichment for their 45 GWd/MTU burnup design basis PWR fuel is seen to be below the average (mid-range) enrichment.

(b) Section 7.3.3.5 of the PFSF SAR, "Dose Rates at Distances from the PFSF Array of Storage Casks", states the following:

"The spent fuel basis for these calculations is that all 4,000 casks contain 40 GWd/MTU burnup and 10-year cooled PWR fuel, with a low initial enrichment assumed for this burnup. A more realistic cooling time of 10 years (as compared to 5-year cooled reference fuel) is used since it is not reasonable to assume that 4,000 loaded storage casks are stored at the PFSF with an average cooling time of 5 years. This is based on the following: (1) the majority of the nuclear power plant spent fuel currently available to be stored at the PFSF is over 10 years old; (2) the vendors' minimum cooling time requirement for transporting 40 GWd/MTU PWR fuel is 10 years for the Holtec HI-STAR shipping cask system and 7 years for SNC's TranStor shipping cask system; and (3) the anticipated maximum storage cask loading rate at the PFSF is one cask per operating day or about 200 casks per year, which at this rate would take 20 years for the PFSF to be filled. Therefore, a 10-year cooling time is considered to be conservative for the 4,000-cask PFSF array since the actual average cooling time is expected to be much greater than 10 years. 40 GWd/MTU is considered to represent a conservative burnup for the majority of fuel stored at the PFSF."

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

The above paragraph from Section 7.3.3.5 of the PFSF SAR indicating that the TranStor shipping casks are capable of shipping PWR fuel with 40 GWd/MTU burnup and 7 year cooling time is based on Rev. 1 of the TranStor Shipping Cask SAR that is referenced in the PFSF SAR and was in effect at the time the PFSF SAR was prepared. Rev. 2 of the TranStor Shipping Cask SAR revised the minimum cooling time for PWR fuel with 40 GWd/MTU burnup to 8 years. Therefore, 8 years is the minimum cooling time of spent PWR fuel having 40 GWd/MTU burnup that could be transported to the PFSF in either the HI-STAR or TranStor shipping casks.

DOE's Energy Information Administration's Service Report entitled "Spent Nuclear Fuel Discharges from U.S. Reactors - 1994", published in February 1996, provides information regarding characteristics of spent fuel in the U.S. This report was reviewed to evaluate average burnups and cooling time associated with the spent fuel inventory at the end of 1994. At this time, the spent fuel inventory from pressurized water reactors (PWRs) was approximately 19,000 metric tons of uranium (MTU), and the inventory from boiling water reactors (BWRs) approximately 11,000 MTU, for a total inventory of approximately 30,000 MTU. This spent fuel inventory represents 75% of the capacity of the PFSF. While it is recognized that provisions already exist for storage of some of this spent fuel and the PFSF will not furnish storage for this entire inventory, data associated with this spent fuel is considered representative of fuel that the PFSF could be expected to receive. The weighted average burnup (weighted by MTU) for the BWR spent fuel inventory in the U.S. was calculated from Table 6 of the above referenced DOE Report to be approximately 23.8 GWd/MTU, and the weighted average burnup for the PWR spent fuel inventory in the U.S. was calculated from Table 7 of this report to be approximately 32.4 GWd/MTU.

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

Because of the large inventory of spent fuel taken into account (approximately 30,000 MTU), this is considered to be a reasonable representation of typical fuel that will be received at the PFSF.

Based on this evaluation of the spent fuel inventory in existence in the U.S. at the end of 1994, it is determined that use of the 40 GWd/MTU burnup and 10-year cooled PWR fuel assumed in the shielding analyses to evaluate dose rates at the RA fence and OCA boundary from the array of 4,000 casks is conservative.

(c) Different characteristics of the spent fuel were assumed in Sections 7.3.3.5 and 7.4 of the PFSF SAR. In Section 7.3.3.5, the conservative "average" fuel characteristics of 40 GWd/MTU burnup and 10-yr cooling time were assumed to demonstrate that dose rates at the RA fence and OCA boundary from the full array of 4,000 storage casks at the PFSF will be in compliance with the regulatory requirements of 10 CFR 20.1301 (dose rate less than 2 mrem/hr for unrestricted areas) and 10 CFR 72.104 (annual dose to an individual at the OCA boundary of less than 25 mrem). Section 7.4, "Estimated Onsite Collective Dose Assessment", provides the estimated occupational exposure to PFSF personnel involved in operations that include receipt of shipping casks, canister transfer operations from the shipping cask to the storage casks, movement of storage casks to the pads, and inspection and surveillance activities.

An objective of the occupational exposure estimate was calculation of more realistic and not necessarily conservative integrated doses that reflect expected personnel exposures. We purposely used more conservative fuel characteristics for the analysis in Section 7.3.3.5 to ensure that we are well below the 10 CFR 20 and 10 CFR 72 regulatory limits. Whereas the analysis for Section 7.4 is intended to be an estimate of actual doses. For this reason, the values of burnup and cooling time used in Section 7.3.3.5 to assess dose rates at boundaries from the array of 4,000 casks, shown to be conservative in the above response, were not applied to estimate worker integrated doses. As stated in PFSF SAR Section 7.4:

"Dose rate values include both gamma and neutron flux components, and are based on PWR fuel with 35 GWd/MTU burnup and 20-year cooling time. Fuel with these characteristics is considered to be representative of typical fuel that will be contained in canisters handled at the PFSF and dose estimates based on fuel with these characteristics are considered to be realistic."

Evaluation of weighted average burnups and cooling times of the nations' PWR and BWR spent fuel inventory, as discussed in the response to RAI 7-1(b) above, indicates an overall weighted average burnup (weighted by metric tons uranium) of approximately 32.4 GWd/MTU for PWR fuel and approximately 23.8 GWd/MTU for BWR fuel, with a weighted average cooling time for both types of fuel of approximately 23.0 years (assuming 30,000 MTU of spent fuel is received during the first 15 years of PFSF operation). Based on this evaluation, the 35 GWd/MTU burnup and 20-year cooling time spent fuel assumed in the onsite dose assessment is considered to be representative of typical fuel expected to be received at the PFSF whose use will result in reasonably accurate occupational exposure estimates.

CHAPTER 7—RADIATION PROTECTION

7-2 (a) Calculate the dose to worker clearing debris from the inlet ducts of the storage casks.

(b) Provide all assumptions made to calculate dose to worker, including location of worker relative to the duct, dose rate at this location, and time it will take for worker to clear the debris.

RESPONSE

The PFSF SAR discusses dose rates associated with clearing debris from inlet ducts in Sections 7.4 and 8.1.3.4 giving different dose rates associated with each case. This is not a discrepancy, but an intentional differentiation between integrated worker doses associated with routine clearing of a small amount of debris from around an assumed 200 casks per year (for the occupational exposure estimate) and worst case dose rates associated with clearing inlet ducts from a cask having half of its inlet ducts completely blocked (in the accident analysis). The bases for the different dose estimates are provided in the following paragraphs:

<u>Accident Analysis</u>

PFSF SAR Section 8.1.3.4 states the following:

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

Based on Section 11.1.2.3 of the TranStor SAR, it is assumed that a person spends 30 minutes in the radiation field clearing blocked inlet ducts. The TranStor SAR characterizes this as a "worst case estimate" for the time to clean the vents, assuming the person is kneeling next to the cask with his hands on the vent inlets the entire time. This is considered to be a conservative estimate for the time required to remove blockage from one-half the inlet ducts.

From PFSF SAR Table 7.3-1, which assumes design basis fuel in a HI-STORM storage cask (this table is used since dose rates are slightly higher than those associated with the TranStor storage cask, given in Table 7.3-2), the dose rate on contact with the bottom air inlet duct is 50 mrem/hr, and the dose rate on contact with the side of the cask is nearly 30 mrem/hr. These are maximum dose rates associated with design basis PWR fuel with 45 GWd/MTU burnup and 5-year cooled. It was estimated that surrounding casks contribute an additional 20 mrem/hr, which is an estimated average dose rate to a person located in the cask array, with surrounding casks loaded with the conservative "average" fuel (assumed to have 40 GWd/MTU burnup and 10 years cooling time). Thus, the total dose rate at the extremities involved in cleaning the blocked inlets is 50 + 20 = 70 mrem/hr, and the total dose rate at the whole body is 30 +20 = 50 mrem/hr. One-half hour exposure time produces integrated doses of 35 mrem to the hands and forearms and 25 mrem to the whole body.

The basis for the estimated 20 mrem/hr value from adjacent casks is supplied in the enclosed Calculation No. 05996.02-UR-5, entitled "Dose Rate Estimates from Storage Cask Inlet Duct Clearing Operations." The worker is assumed to be located near the center of a storage pad supporting eight HI-STORM (highest calculated side dose rates) storage casks, in contact with the affected cask and at the closest distance (approximately 1 meter) from the adjacent cask in the other column on the storage pad. This calculation determined that the dose rate from nearby casks whose canisters have a direct radiation path to the assumed worker location would be 33.9 mrem/hr if nearby casks are assumed to contain design basis fuel. This value is based on the conservative assumption that, if any portion of a nearby canister has a direct radiation path to the assumed worker location (ie, no intervening storage casks), dose rates at the worker location are calculated as if the entire canister has a direct path to the worker location. The assumption that nearby storage casks are loaded with design basis fuel was considered to be overly conservative. While it is assumed that the affected cask contains design basis fuel, it is assumed that nearby casks contain the conservative average PFSF fuel, with 40 GWd/MTU burnup and 10 year cooling time. The gamma and neutron dose rates were each scaled using the methodology described in Section 5.4.1 of the TranStor Storage Cask SAR, with source data obtained from the OCRWM Light Water Reactor Database (DOE/RW-0184-R1, Characteristics of Potential Repository Wastes, Office of Civilian Radioactive Waste Management, U.S. Department of Energy, July 1992). Scaling direct radiation dose rates from nearby casks to account for this "average" fuel instead of design basis fuel results in a calculated direct dose rate of 16.54 mrem/hr from nearby casks. This was multiplied by a factor of 1.25 to account for scattered radiation at the assumed worker location from canisters which do not have a line-of-sight radiation path to the assumed worker location, resulting in a total estimated dose rate from nearby casks of 20.7 mrem/hr. This calculation supports the 20 mrem/hr estimated contribution from nearby casks in the PFSF accident analysis.

Integrated Personnel Dose Assessment

PFSF SAR Section 7.4 states:

> "Conservatively assuming that 5 percent of the 4,000 casks require clearing of debris from the inlet ducts once a year at 10 minutes each, in a dose field of 15 mrem/hr, an additional annual dose of 0.5 person-rem is estimated."

This assessment is concerned with average dose rates from routine clearing of small amounts of debris from inlet ducts and not worst case conditions. It is assumed that quarterly visual surveillances of the storage casks and pads identify 200 casks each year that have accumulation of debris at the inlet ducts. It is considered reasonable that, on average, one worker with a hand tool (e.g. rake or shovel) can clean up debris near the inlet ducts of a single storage cask in 10 minutes. The 15 mrem/hr is an estimated average dose rate to a person located in the cask array, with both the affected cask and surrounding casks assumed to be loaded with typical fuel. The basis for the estimated 15 mrem/hr value is supplied in the enclosed Calculation No. 05996.02-UR-5, entitled "Dose Rate Estimates from Storage Cask Inlet Duct Clearing Operations." The worker is assumed to be located in the same position as in the accident analysis case, described above.

The dose rate at this point is calculated to be 64 mrem/hr due to direct radiation (canisters having line-of-sight radiation path to the assumed worker location) from the cask whose inlet ducts are being cleared and nearby casks, assuming the affected cask and nearby casks contain design basis fuel. In order to obtain a realistic dose estimate to workers performing routine tasks, it was assumed that the affected cask and nearby casks contain typical PFSF fuel, with 35 GWd/MTU burnup and 20 year cooling time. Dose rates were scaled from those associated with design basis fuel using the same methodology described above for the accident analysis case. Direct radiation dose rates from the affected cask whose inlet ducts are being cleared and nearby casks to account for this "cooler" fuel results in a calculated dose rate of 8.6 mrem/hr at the worker location. This was multiplied by a factor of 1.25 to account for scattered radiation at the assumed worker location from canisters which do not have a line-of-sight radiation path to the assumed worker location, resulting in a total estimated dose rate from the affected cask and nearby casks of 10.8 mrem/hr. Based on this calculation, the value of 15 mrem/hr estimated in Section 7.4 of the PFSF SAR for the integrated personnel dose assessment is determined to be reasonable, and somewhat conservative, for typical fuel.

CHAPTER 7—RADIATION PROTECTION

7-3 (a) Provide basis for the dose rates in Tables 7.4-1 and 7.4-2, which depict the dose to workers during receipt and transfer operations, and conclude that the dose limits of 10 CFR 20.1201 will not be exceeded.

(b) Provide the assumptions (e.g., work times, locations, etc.) used when considering a reduction in dose owing to temporary shielding.

RESPONSE

The bases for PFSF SAR Tables 7.4-1 and 7.4-2 are provided in the enclosed calculation (No. 05996.02-UR-6, entitled "Calculational Basis for PFSF SAR Tables 7.4-1 and 7.4-2, Estimated Personnel Exposures for Canister Transfer Operations"). The calculation provides the assumptions for the general location of workers for each operation identified in the tables, calculated dose rates at these locations, estimated times to perform each operation, and the estimated dose reduction due to temporary shielding for those steps where applicable. Revisions were determined to be necessary to the PFSF SAR Tables 7.4-1 and 7.4-2, as doses estimated in this calculation for some of the steps are different from those identified in these tables in Rev. 0 of the PFSF SAR. There are several reasons for these differences, as follows:

- Whereas it was considered in the original PFSF SAR tables that it would be practical to use temporary shielding to reduce dose rates to workers on top of the canister to levels of approximately 2 mrem/hr, this calculation indicates that gamma reduction would require up to 1.5 inches of lead, which is considered impractical due to the weight of the lead disc required (about 1 ton). Therefore, the revised dose rates reflect the use of temporary neutron shielding but not the much heavier gamma shielding. Consequently dose rates to workers on top of the canister are above the 2 mrem/hr originally assumed (7.1 mrem/hr for HI-STORM and 4.5 mrem/hr for TranStor).

- Sierra Nuclear Corporation revised the applicable design basis PWR fuel for their TranStor shipping cask in Revision 2 of the TranStor Shipping Cask SAR, from 45 GWd/MTU burnup and 8 year cooled to 40 GWd/MTU burnup and 8 year cooled. This resulted in different dose rates associated with this shipping cask with design basis fuel, as well as different scaling factors used to scale the gamma and neutron source strengths from the design basis fuel to the typical PFSF fuel (35 GWd/MTU burnup and 20 year cooled). The effect of this change was, in general, an increase in dose rates and integrated doses to workers for those tasks involving the TranStor shipping cask.

- General area dose rates associated with the HI-STAR shipping cask for workers involved in receiving and inspecting the shipment, measuring dose rates and removing the personnel barrier, had previously been estimated at 2.5 mrem/hr for typical PFSF fuel, but were recalculated to be approximately 4 mrem/hr.

Final integrated doses to operators and HP personnel involved in the canister transfer operations, with credit for temporary shielding on top of the shipping casks and canisters, increased from 176.6 mrem for HI-STORM and 182.9 mrem for TranStor (listed in Tables 7.4-1 and 7.4-2 of Rev. 0 in the PFSF SAR) to 198.7 mrem for HI-STORM and 208.9 for TranStor. PFSF SAR Tables 7.4-1 and 7.4-2 are being revised accordingly.

CHAPTER 7—RADIATION PROTECTION

Section 7.5 Radiation Protection Program

7-4 Describe how the radiation protection plan will ensure worker doses will be limited to less than the limits of 10 CFR 20.1201 in areas of the facility where area radiation monitors are not available.

RESPONSE

The radiation protection program as described in Chapter 7 ensures that occupational doses are below the limits required by 10 CFR 20.1201 as well as ensuring that occupational radiation exposures are as low as is reasonably achievable (ALARA). The use of area radiation monitors in the Canister Transfer Building is only one part of this program.

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

The Restricted Area (RA), is approximately 99 acres and is surrounded by a chain link security fence and an outer chain link nuisance fence with an isolation zone and intrusion detection system between the two fences. Access to the RA is controlled through a single access point in the Security and Health Physics Building (see SAR Figure 1.2-1, the PFSF General Arrangement). Personal dosimetry is issued and controlled in this building to individuals entering the RA. External radiation dose monitoring will be accomplished through the use of thermoluminescent dosimeters (TLDs) and self reading dosimeters (SRDs) or digital alarming dosimeters (DADs). The official record of external dose to beta and gamma radiations will normally be obtained from the TLDs, with SRDs or DADs used as a means for tracking dose between TLD processing periods and as a backup to TLDs. Self-reading dosimeters will be administered in accordance with the guidance in Regulatory Guide 8.4.

Periodic radiation surveys will be conducted of all areas inside the RA. Maps will be generated showing the radiation levels in all areas. Radiation work permits (RWPs) will be completed by qualified radiation protection personnel prior to any entry and will identify normal and unusual radiation readings. Workers will be required to read, understand and sign that they are aware of the conditions or unknowns. Personnel will be trained to use the appropriate radiation detection instruments or will be required to have a qualified radiation protection technician with them at all time while in the areas. Training will include responses to unusual readings and off-scale conditions. The Radiation Protection program will provide for the immediate reading of any individual's TLD if an unusual reading or off-scale condition occurs.

Implementation of the Radiation Protection Program procedures as outlined in Section 7.5.3 ensures that occupational doses are below the limits required by 10 CFR 20.1201 and are ALARA both in the Canister Transfer Building as well as other parts of the Facility where radiation monitors are not installed.

CHAPTER 7—RADIATION PROTECTION

Airborne and Environmental Monitoring

7-5 Describe in more detail the airborne and environmental monitoring program at the PFSF and operations. Include in this description the types of monitoring, monitoring locations, collection frequency, method of collection, and type of radionuclide analysis with lower limits of detection, as appropriate.

RESPONSE

As discussed in SAR Section 7.3.5, there are no credible events that could result in releases of radioactive material from within the canisters or unacceptable increases in direct radiation levels. Therefore area radiation and airborne radioactivity monitors are not needed in the storage area. Additionally, there are no liquid or gaseous effluent releases from the PFSF. Therefore a radioactive effluent monitoring system is not needed and routine monitoring for effluents is not performed.

During routine storage operations at the PFSF, the only radiological instrumentation in use in the storage area will be TLDs. TLDs will be used to record dose rates in the RA and along the OCA boundary fence. TLDs provide a passive means for continuous monitoring of radiation levels and provide a basis for assessing the potential impact on the environment.

As discussed in SAR Section 7.5.3, radiation protection requirements for all radiological work at the PFSF will be governed by radiation protection procedures. A procedure will be developed to perform contamination surveys to detect and remove any storage system contamination before the cask is placed in the storage area. A procedure will also be developed for determining radiation doses at the RA and OCA boundaries using TLDs.

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

For redundancy, each TLD location mentioned above will house a set of two TLDs. The TLDs will be retrieved and processed quarterly. The TLDs will primarily detect gamma radiation and have a lower limit of sensitivity of approximately 0.02 mrem.

CHAPTER 8—ACCIDENT ANALYSES

General

8-0 As indicated in RAI Section 8, provide the requested information needed for the NRC staff to conduct a review of the accident analysis.

- Regulatory Guide 3.48, "Standard Format and Content Guide for the Safety Analysis Report for an Independent Spent Fuel Storage Installation," NUREG–1567, "Standard Review Plan for Spent Fuel Storage Facilities," Regulatory Guide 3.61, "Standard Format and Content of Topical Safety Analysis Reports," and NUREG–1536, "Standard Review Plan for Dry Cask Storage Systems," provide detailed areas of review, review procedures, and acceptance criteria to be used in review of the SAR for this facility.

RESPONSE

The Section 8 RAIs are individually addressed in the following pages. It is our intention to provide the requested information for each response in sufficient detail that the RAI is adequately addressed.

CHAPTER 8—ACCIDENT ANALYSES

Section 8.0 Accident Analysis

8-1 (a) Provide the basis for selecting off-normal and accident conditions to ensure all relevant or potential scenarios were considered.

(b) Justify the exclusion of potential scenarios such as failure of the doors on the transfer casks during canister movement and external impacts from nearby facilities (e.g., the military training range). Otherwise, provide a discussion of such events or conditions and identify the appropriate bounding analysis.

- 10 CFR 72.24(d)(1) and 72.122(2)(l); NUREG–1567 (Sections 12.4.1, 12.4.3, and 12.5.1); and Regulatory Guide 3.48, Section 8.2, state the identification of off-normal and accident-level events and conditions should be based on a thorough review of what could reasonably occur and that a systematic analysis could be used to identify and assess potential hazards to minimize omissions.

RESPONSE

(a) ANSI/ANS 57.9, the regulatory guidance in Sections 12.4.1 and 12.4.3 of NUREG-1567, and the storage system vendor SARs were used as the basis for selecting off-normal and accident conditions to ensure all relevant or potential scenarios were considered.

Section 12.4.1 of NUREG-1567 indicates that examples of off-normal and accident conditions that should be considered in the SAR include those provided in ANSI/ANS-57.9. As described in the introduction to Chapter 8 and Section 8.2 of the PFSF SAR, ANSI/ANS-57.9 served as a basis for identifying accidents and classifying them into off-normal conditions or accidents. ANSI/ANS-57.9 Design Event II conditions are described in PFSF SAR Section 8.1. ANSI/ANS-57.9 Design Events III and IV are described in Section 8.2. The examples of off-normal occurrences and accidents provided in ANSI/ANS-57.9 are included in the PFSF SAR, where applicable to the PFSF.

The regulatory guidance of NUREG-1567 Section 12.4.3 was also used as a basis for selecting off-normal and accident conditions to ensure all relevant or potential scenarios were considered. Consideration of the eight bullet items listed in Section 12.4.3 included the following:

Site Characteristics

Consideration was given to the following in the analysis of the PFSF site characteristics:

- Thermal analyses of the effects of abnormally high ambient temperatures on the storage system considered climactic conditions of the area, and temperatures were selected to bound day/night average maximum temperatures that could occur over a period of several days (PFSF SAR Section 8.1.2).

- As described in PFSF SAR Section 8.2.1.2, a seismological evaluation of the PFSF siting area was performed. Although the HI-STORM and TranStor cask storage systems have been analyzed for generic design earthquakes (DE) selected by each vendor and described in their respective SARs, both storage systems were also analyzed for the PFSF site specific DE. The site specific DE is discussed in Section 3.2.10 and is represented by response spectrum curves developed specifically for the PFSF site.

- PFSF SAR Section 8.2.2.2 indicates that for the extreme wind accident, the maximum wind speed and pressure drop analyzed by the storage system vendors substantially envelopes the site specific requirements defined in Regulatory Guide 1.76 for tornado Region III. The Canister Transfer Building is designed to withstand the effects of the Region III tornado and pressure drop forces.

- In the case of flooding, PFSF SAR Section 8.2.3 assesses the site specific effects of flooding as well as providing information regarding capabilities of the vendor storage casks, which are designed to withstand severe flooding including full submergence.

- The explosion analysis in Section 8.2.4 considers the effects of explosion at the stationary rocket engine test facility located approximately 2.5 miles south-southeast of the PFSF site. The effects of explosion from a transportation accident on the Skull Valley Road, 2 miles from the nearest storage pad, were also evaluated.

- Fires evaluated in Section 8.2.5, and design measures to mitigate the effects of fires, are based on specific fire hazards associated with the PFSF.

- The hypothetical storage cask drop/tipover not only discusses the vendors' generic analyses, but also factors associated with site-specific storage pad concrete and soil parameters.

• PFSF SAR Section 8.3 states the following in regards to the potential for external impacts from nearby facilities, e.g. aircraft affecting the PFSF:

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

Automatic and Manual Safety Features

Analysis of the canister transfer operations identified the need for single-failure-proof canister and transfer cask lifting equipment which was incorporated into the design of lifting devices. As discussed in Section 8.1.1.3, the overhead bridge crane, semi-gantry crane, canister downloader and associated lifting devices used to handle shipping casks, transfer casks and canisters in the Canister Transfer Building are all designed to meet the criteria for single-failure-proof lifting devices and to hold the lifted load in place in the event of loss of electrical power. The cranes are seismically qualified to assure dropped loads will not occur in the event of a DE.

Necessary Instrumentation and Control Features

Analysis of the method for detecting blockage of the storage cask air paths at existing ISFSIs smaller than the PFSF (i.e., daily visual inspection of the cask vents) identified a desire for an alternative method that has a significant ALARA benefit. It was determined to use a cask temperature monitoring system that continuously monitors the temperature of the casks. This allows adequate monitoring of the cask thermal performance without subjecting operators to a daily radiation dose. To provide assurance of the availability and reliability of the temperature monitoring system, the following features are provided: backup power, procedures to periodically calibrate components and test the operability of the monitoring system, and a daily review of the monitoring output to detect trends of increasing cask temperature.

Although important for ALARA purposes, the temperature monitoring system is not classified as Important to Safety. In the event of failure of the system, a supervised alarm and detection system (which is a separate alarm from the high temperature alarm) will notify operators who will provide visual inspections of the affected cask(s) until the monitoring system is repaired.

The visual inspection of the cask vents to verify no blockage will still be performed, but on a quarterly basis. The use of the temperature monitoring instrumentation to reduce the frequency of visual inspections from a daily to a quarterly activity will greatly reduce the radiation exposure of personnel.

Sequences of Operations and Projected Contingency Actions

Analysis included a review of the sequence of operations associated with shipping cask receipt, canister transfer, and movement of the storage casks to the storage pads. The "stacked cask" configuration, where the transfer cask is supported on the shipping cask or storage cask, was thoroughly assessed and it was decided that the Canister Transfer Building should be qualified to withstand the effects of tornado winds and tornado-driven missiles to shelter this configuration in the operations sequence from the effects of tornadoes. PFSF SAR Section 8.2.1.2 identifies several requirements associated with canister transfer operations that were the result of consideration of a seismic event occurring at different stages of the canister transfer sequence. For instance, prior to disconnecting the crane from the shipping cask after it is placed in a canister transfer cell, seismic support struts are secured to the cask. The stacked cask configuration was evaluated for stability in the event of a DE. During the canister transfer operation, the crane is not disconnected from the transfer cask that is supported by the shipping cask or storage cask until seismic support struts are connected to the transfer cask to assure its stability.

The analysis considered the occurrence of fires (PFSF SAR Section 8.2.5) involving 1) 300 gallon fuel capacity heavy haul vehicle with a shipping cask in the cask load/unload bay; 2) 50 gallon fuel capacity cask transporter in a canister transfer cell; 3) cask transporter with a storage cask enroute to the storage pads; and 4) cask transporter with a storage cask on the storage pads. Several restrictions resulted from the assessment of potential fires, including the cask transporter not permitted in a canister transfer cell while transfer operations are in process and, for rail delivery/retrieval of shipping casks, the train locomotives are required by administrative procedure to stay out of the Canister Transfer Building.

The sequence of operations was also considered in regards to loss of electrical power as described in Section 8.1.1.3, which states:

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

(4) while the canister is being lowered from the transfer cask into the storage cask."

Characteristics of Facilities and Equipment

The location of the cooling air inlet ducts at the bottom of the storage casks, a characteristic of their design, gives rise to consideration for potential duct blockage due to buildup of material on the storage pads due to high winds, tornado, heavy snow, and flooding, evaluated in PFSF SAR Sections 8.1.3 and 8.2.8. PFSF SAR Section 8.1.4 evaluates bumping of the canister against the sides of the shipping or storage cask during canister transfer, which relates to characteristics of the cask configuration during this operation. Characteristics of the fuel and canister were accounted for in assessing the possibility and consequences of canister breach and canister pressurization accidents, discussed in PFSF SAR Sections 8.2.7 and 8.2.10. The characteristics of the heavy haul vehicle and cask transporter (fuel tank capacity) were considered in evaluating the consequences of postulated fires as discussed above. Vertical drop of a storage cask (PFSF SAR Section 8.2.6) considered characteristics of the cask transporter as well as the storage pads.

Consequences of Failures of Structures, Systems, and Components (SSCs)

Consideration of failure of lifting devices led to the decision to require that these devices meet single-failure-proof requirements to avoid the consequences of dropped casks and/or canisters.

Although it was determined that canister breach is not a credible event, the consequences of hypothetical failure of the canister confinement barrier are evaluated in Section 8.2.7. The consequences of this accident bound those of credible accidents that could occur at the PFSF.

Historical Considerations

Several accident conditions were evaluated not because they represent credible scenarios, but based on historical considerations - the fact that these conditions were considered in the licensing basis of other ISFSIs and/or in the PFSF storage cask vendor SARs. The hypothetical storage cask drop/tipover, hypothetical loss of confinement barrier, and hypothetical accident pressurization are examples of accidents considered partly as the result of historical reasons, which do not represent credible accident scenarios. Section 12.4 of NUREG-1567 states that "Credible accident level events and conditions should be analyzed (or bounded by design basis accidents) to demonstrate that the consequences do not exceed the limits of 10 CFR 72.106(b). (Design basis accidents are the subset of all credible accidents that bound the entire spectrum of accidents that could occur in terms of the nature and consequences of accidents.) Instead of providing analysis for every credible accident scenario, the SAR may choose to characterize and analyze the subset of design basis events." Section 12.4.1 of this NUREG states that "Credibility is the determinant for analysis and satisfaction of criteria for accident-level events and conditions." The PFSF SAR exceeds these requirements and analyzes several incredible accident scenarios, largely due to historical precedent.

Consequences of Human Error

Section 8.1.4 of the PFSF SAR assesses consequences of the occurrence of postulated operator error during the canister transfer operation. As stated in this section:

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

It is considered that the off-normal contamination release event could occur as the result of operator error. Section 8.1.5.1 states:

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

In addition to ANSI/ANS 57.9, and the regulatory guidance in Sections 12.4.1 and 12.4.3 of NUREG-1567, the storage system vendor SARs were used as a basis for selecting off-normal and accident conditions

Based on the above, the accident analysis in the PFSF SAR is based on a thorough review of a wide range of accident level events in accordance with the regulatory guidance in NUREG 1567. This approach ensured that relevant or potential off-normal and accident scenarios are considered in the PFSF SAR.

(b) As discussed in the response to RAI No. 8-1(a) above, external impacts from the Dugway Proving Grounds were excluded based on data obtained from the Dugway Proving Grounds and its associated Michael Army Air Field. As discussed in Section 2.2, the calculated probability of an aircraft impacting the PFSF is below the applicable guidance and therefore is not considered to be a credible event.

It is very unlikely that the transfer cask doors would fail due to the simplicity and inherent reliability of the door design and their opening/closing mechanism. The TranStor transfer cask doors slide open and closed along greased rails which support and align the doors. The TranStor system uses a hydraulic operator to provide the necessary force to open/close the transfer cask doors. The doors of the HI-TRAC transfer cask are equipped with multiple wheels that run along guided rails, enclosed, with no obstacles or protrusions. The doors are housed in such a way that they cannot come off the tracks. The reduced friction associated with the wheels enables the HI-TRAC doors to be manually opened and closed by the operators, and handles are provided on the doors for this purpose. The transfer cask doors of both vendors' transfer casks will be tested during the preoperational testing program to verify that they operate smoothly and there are no obstructions or misalignment that could cause jamming.

In the event the transfer cask sliding doors fail to close after the canister has been hoisted up into the transfer cask in preparation for a transfer operation, the transfer operation will cease, the canister will be lowered back down into the underlying cask, the transfer cask removed from the cask, the lid placed back on the cask, and the transfer cask doors or door operating mechanisms repaired.

In the event the transfer cask sliding doors are closed and fail to open when it is desired to lower the canister from the transfer cask into an underlying storage or shipping cask, then actions would be taken to make necessary repairs and open the doors with the canister in the transfer cask while the transfer cask is supported by the underlying cask. Dose considerations would be associated with this repair operation, as dose rates at the side of transfer casks are relatively high compared to dose rates on the sides of storage casks (as indicated in PFSF SAR Tables 7.3-1 through 7.3-4), and use of temporary shielding may be desirable. There is no hurry or time constraint associated with this operation, as the canister can remain housed in the transfer cask indefinitely without posing a safety concern. Corrective action would be carefully planned and executed in a deliberate controlled manner that assures doses to personnel involved are maintained ALARA. In the case of the TranStor transfer cask, if the hydraulic operator is broken and unable to open the sliding door, then the hydraulic operator would be repaired. This repair could either be performed in place, or the hydraulic operator mechanism could be removed, taken to another location (e.g. a low dose area) where the repair could be effected, then returned to the transfer cask, re-mounted and hydraulic force applied to open the sliding doors. In the case of the HI-TRAC transfer cask, if operators are unable to slide the doors open manually, a portable cable winch could be connected and used to provide the additional force necessary to slide the doors open even if a wheel is jammed or a wheel bearing seized.

CHAPTER 8—ACCIDENT ANALYSES

Sections 8.1 and 8.2 Off-Normal Operations and Accidents

8-2 Provide consequences of failures of those features relied upon for prevention or mitigation of events to ensure these failures would not result in an unanalyzed condition for the cask.

- 10 CFR 72.24(d)(2); NUREG–1567 (Sections 12.4.2, 12.4.3, and 12.6); and Regulatory Guide 3.48, Sections 8.1.1.1, 8.1.1.3(3), and 8.2.1.2(7), state the adequacy of SSCs provided for prevention of accidents and the mitigation of consequences of accidents should be evaluated. This includes a comprehensive review of the consequences of failures of these SSCs.

RESPONSE

Each of the SAR Chapter 8 events were reviewed to determine what features are relied upon to prevent or mitigate these events and the consequences associated with the failure of the features. The following discussion presents this information for each of the off-normal and accident scenarios described in SAR Chapter 8.

OFF-NORMAL OPERATIONS

Loss of External Electrical Power

SAR Section 8.1.1 postulates a total loss of electrical power event. There are no safety or radiological consequences for this event because loss of power does not affect the integrity of the canisters and does not result in the release of radioactive material. No PFSF spent fuel storage nuclear safety functions rely on electrical power for their accomplishment.

However, the emergency diesel-generator is provided as a backup power supply to maintain the operation for certain systems whose continued energization is desirable. These systems include the security system, emergency lighting, and cask temperature monitoring system. None of these systems are relied upon to maintain the operation of those systems classified as Important to Safety.

Three lifting devices that are Important to Safety and use electric power for normal operation are the overhead bridge crane, semi-gantry crane, and Holtec canister downloader. These are used during the shipping cask unloading/loading and canister transfer operations. As discussed in SAR Sections 4.7.2.1 and 4.7.3.5.1, the overhead bridge crane, semi-gantry crane, and canister downloader are all designed to meet the criteria for single-failure-proof lifting devices. The overhead bridge crane and the semi-gantry crane are designed to hold the lifted load in the event of loss of electrical power, with the brakes automatically actuated. The canister downloader is also designed to fail-as-is upon loss of electrical power, with two redundant sets of anti-drop cam locks. A loss of electrical power would delay the transfer operation but would not challenge the integrity of the canister or safe storage of the spent fuel in the canister. The overhead bridge crane, semi-gantry crane and canister downloader are all capable of supporting their rated loads indefinitely without electrical power.

Off-Normal Ambient Temperatures

SAR Section 8.1.2 postulates a high ambient temperature event. As discussed in the HI-STORM and TranStor SARs, the component temperatures during this event are all within the vendor temperature limits. The canister and storage cask temperatures pose no threat of fuel cladding failure, canister breach, or reduction in shielding provided by the storage cask. Normal weather monitoring and/or by evaluation of data from the storage cask temperature monitoring system are used to assess conditions. However, detection of off-normal ambient temperatures is not critical because there are no adverse safety consequences, i.e., the storage systems are designed to withstand such conditions.

Partial Blockage of Storage Cask Air Inlet Ducts

SAR Section 8.1.3 postulates a complete blockage of one-half of the air inlet ducts. The feature that is relied upon to prevent or mitigate this event is the surveillance of the cask vents. The cask vents are inspected quarterly to verify no blockage has occurred. However, detection of partial blockage of storage cask air inlet vents is not critical because there are no adverse safety consequences. As discussed in the HI-STORM and TranStor SARs, the component temperatures for this event are all within the vendor temperature limits. The canister and storage cask temperatures for this event pose no threat of fuel cladding failure, canister breach, or reduction in shielding provided by the storage cask.

Temperatures representative of the thermal performance of each storage cask are remotely monitored by the storage cask temperature monitoring system and trended. Increased temperatures indicate possible blockage of the natural convection air flow path, most likely at the air inlet ducts, and personnel are dispatched to inspect storage casks with high temperatures. Should blockage occur, it will be identified and removed before achieving the steady state temperatures considered in the vendor analyses. If the temperature monitoring system fails, the surveillance frequency is increased until the monitoring system is returned to normal operation.

Operator Error

SAR Section 8.1.4 discusses an operator error event. This event consists of off-normal operator load handling errors that result in the canister impacting against the inside of the transfer or storage cask. The features that are relied upon to prevent or mitigate the effects of this event include the canister, transfer cask, and storage cask designs. Off-normal handling events are evaluated in the HI-STORM and TranStor SARs and presented in SAR Section 8.1.4. The stresses on the canisters and casks were calculated and determined to be within design allowables. The analyses determined that the canister and casks would maintain their structural integrity and continue to perform their safety functions.

In addition, operating procedures will ensure movement of the canister and cask alignment is performed as required to prevent these conditions from occurring. Training procedures will also assist in ensuring that the crane operators can properly perform the necessary canister transfer as normal events.

Off-Normal Contamination Release

SAR Section 8.1.5 discusses an off-normal contamination release event. The features that would prevent this event include the transfer operations and procedures used while loading the spent fuel, surveys that are performed at the originating nuclear power plant and receipt surveys to verify cleanliness at the PFSF. However, for the purposes of assessing a worst-case scenario for this event, an analysis was performed to determine the consequences of the event. The event was assessed conservatively assuming removable contamination levels much higher than is possible for canisters received at the PFSF and above the removable surface contamination limit for accessible canister surfaces specified in Section 10.2.2.1. The doses for an individual located at the closest off-site point from the release point for the duration of the release were calculated to be well below the 10 CFR 72.106 criteria of 5 rem for accidents.

ACCIDENTS

Earthquake

SAR Section 8.2.1 discusses an earthquake event. The features that are relied upon to prevent or mitigate the effects of an earthquake include the storage casks, Canister Transfer Building, cask handling cranes, HI-TRAC downloader, and cask seismic support struts.

As described in SAR Sections 4.2.1.5.1 (H) (HI-STORM) and 4.2.2.5.1 (H) (TranStor), both storage casks were analyzed for a PFSF site specific design earthquake (DE), which is based on a seismological evaluation of the siting area, to assure structural integrity of the cask and cask stability. Cask stability analyses demonstrates that the casks will not tip over or slide excessively in an earthquake.

The overhead bridge crane and the semi-gantry crane are designed to withstand the DE, as is the Canister Transfer Building which provides the structural support for the cranes. As discussed in SAR Section 4.7.2 and 4.7.3, the cranes and HI-TRAC downloader are designed to meet the criteria for single-failure-proof lifting devices and are designed to withstand the DE. The cranes and canister downloader are capable of withstanding the DE during the critical lift without toppling or dropping the load. Therefore, a DE would not cause a load drop accident during lifting of either vendor's shipping cask, transfer cask, or a canister.

At various times during the canister transfer operation, the shipping cask is placed on end and the transfer cask is placed on the top of the shipping cask or the storage cask. In order to assure cask stability in the event of an earthquake, the crane is not disconnected from the shipping cask or transfer cask until seismic support struts are attached to the casks. As discussed in SAR Sections 4.7.3.5.1 and 4.7.4.5.1, the seismic support struts are physically connected to the walls of the transfer cell and are designed to resist forces resulting from the DE and maintain the transfer cask in its upright position. Therefore, these cask configurations are stable and will withstand the forces associated with a DE without a drop accident.

Extreme Wind

SAR Section 8.2.2 discusses extreme wind events. Extreme winds consist primarily of tornadoes and their effects, such as missiles. The features that are relied upon to prevent or mitigate the effects of this event include the storage casks and the Canister Transfer Building.

As shown in SAR Section 4.2, the HI-STORM and TranStor storage systems are designed to withstand loads associated with the most severe meteorological conditions, including extreme winds, pressure differentials, and missiles generated by a tornado. Analyses presented in the HI-STORM and TranStor SARs determined that the restoring moment far exceeded the overturning moment caused by extreme wind conditions and the storage casks would not tip over. Extreme winds in combination with tornado-driven missiles are not capable of overturning a storage cask nor of damaging a canister within a storage cask.

The Canister Transfer Building, which shields and protects SSC's in the canister transfer process, is designed to withstand the effects of the design basis tornado wind and pressure drop forces, as well as the effects of design base tornado missiles. The building provides this protection by means of reinforced concrete walls and roof of sufficent strength to withstand the design basis wind, pressure drop, and missile forces. Additional missile protection is provided by the interior reinforced concrete walls and missile / shielding doors and/or labyrinths.

Flood

SAR Section 8.2.3 discusses a flood event. The features that are relied upon to prevent or mitigate the effects of a flood are the storage systems designs, which are designed to withstand severe flooding, including full submergence. A flood would not adversely affect the storage systems (see response to RAI 3-4).

In addition, an earthen flood diversion berm is added to the site design to divert probable maximum flood sheet flow that could occur. Although the berm is not required to maintain the safe operation of the casks in the event of a flood, and therefore classified as not Important to Safety, it would prevent disruption of normal operation of the facility.

<u>Explosion</u>

SAR Section 8.2.4 discusses an explosion event. There are no features that are relied upon to prevent or mitigate this event since there are no facilities such as chemical processing plants, petroleum refineries, natural gas facilities, or munition depots that could contribute to the potential for significant explosions that are located within 5 miles of the PFSF. SAR Section 2.2 indicates that the only facility which could contribute to the potential for significant explosions within the 5 miles is the Tekoi Rocket Engine Test facility located approximately 2.5 miles south-southeast of the PFSF. This facility is used periodically to test engines mounted on stationary bases. However, due to the distance and natural terrain of the area between the PFSF and the Tekoi Test facility, overpressures resulting from the test facility would be substantially deflected and dispersed and would not produce significant overpressures at the PFSF.

The worst-case explosion potential at the PFSF is considered to be from an accident associated with the transportation of explosives along the Skull Valley Road (see SAR Section 8.2.4 analysis, based on Reg. Guide 1.91). Since the Skull Valley Road is 1.9 miles from the Canister Transfer Building and 2 miles from the nearest storage pad, explosions invoving vehicles travelling on this road would not produce significant overpressures at these locations.

As shown in SAR Sections 4.2.1.5.2 (I) and 4.2.2.5.2 (I), both storage systems are designed for overpressures substantially greater than the 1 psi given in Regulatory Guide 1.91 below which no significant damage would be expected. The Canister Transfer Building is designed to withstand extreme winds, pressure drops of 1.5 psi, and missiles associated with the design tornado. The effects of credible explosions occurring on the Skull Valley Road, with resultant overpressures less than 1 psi at the PFSF, would not challenge the Canister Transfer Building's structural integrity. Since there is no potential for significant overpressures occurring at the PFSF as a result of nearby explosions, there would be no damage to the cask storage or transfer systems.

Fire

SAR Section 8.2.5 discusses the consequences of potential fires at the PFSF. The features that are relied upon to prevent or mitigate the effects of a fire include the storage systems design, shipping cask design, and operating procedures. Fires were postulated at the storage pads and in the Canister Transfer Building where a canister would be located. The postulated fires involve worst-case scenarios, which were determined to be a spill and ignition of diesel fuel from the cask transporter or heavy haul vehicle tractor's saddle tanks in the vicinity of a storage or transportation cask.

As shown in SAR Sections 4.2.1.5.1 (J) and 4.2.2.5.1 (J), the storage system designs are highly resistant to the effects of fires. The thick concrete walls are not significantly affected by short-term exposure to fire induced temperatures, and any fire would be required to burn for many hours before much of the wall thickness would be affected. Analyses conclude that the effect of a fire accident on the canister temperature is negligible and that the ability of the storage systems to cool the spent fuel within design temperature limits is not compromised. Shipping casks demonstrate their ability to resist the effects of specified fires under 10 CFR 71 requirements with spent fuel remaining within temperature limits and showing that no breach of the confinement barrier occurs. Therefore, a fire would have a negligible effect on canister and fuel temperatures and cause no reduction in nuclear safety.

PFSF operating procedures will ensure that fuel volumes in excess of the analyzed amounts would not occur. The procedures require that train locomotives stay out of the Canister Transfer Building and the cask transporter cannot enter the transfer cell while a canister is in a transfer cask.

In addition, fire equipment (fire hydrants, fire truck, and sprinkler systems) are provided to aid in the suppression of fire. The fire protection system is not required to maintain the safe operation of the casks and is therefore classified as not Important to Safety. The fire protection system is described in SAR Section 4.3.8. The fire protection system reduces the potential damage of a fire by providing fire suppression to areas where critical components are used. Operation and training procedures also help ensure personnel avoid hazardous conditions and alert fire protection units if a fire is started.

Hypothetical Storage Cask Drop / TipOver

SAR Section 8.2.6 addresses a hypothetical drop / tipover of a storage cask The feature that is relied upon to prevent or mitigate the effects of this event is the storage cask design. The stability of the loaded storage casks in the upright position on the PFSF concrete storage pad is demonstrated in SAR Chapter 4. However, analyses of a hypothetical storage cask drop and/or tipover were performed and are documented in the HI-STORM SAR Chapter 3 and TranStor SAR Section 11.2.10.

Analyses presented in the Holtec and TranStor SARs determined that a hypothetical tipover of a storage cask at the PFSF would result in cask and canister accelerations and stresses bounded by the design accelerations and code allowables. The cask and canister would not sustain significant damage due to the hypothetical tipover event, and would continue to perform its safety functions. Holtec calculated the maximum drop height of the HI-STORM storage cask as 10 inches. For the TranStor storage cask, drops from heights up to 18 inches are not considered to be a concern. Since storage casks cannot be lifted above 10 inches by the cask transporter, end drop accidents at the PFSF will produce decelerations less than those analyzed by the storage system vendors.

Based on the results of the analyses described above, the cask/canister storage systems would retain their confinement integrity and there would be no release of radioactivity and no resultant doses in the event of hypothetical drop/tipover of a fully loaded storage cask.

Hypothetical Loss of Confinement Barrier

SAR Section 8.2.7 addresses a hypothetical loss of confinement of a canister. Since this is a hypothetical accident, there are no features relied upon to prevent or mitigate this event. Loss of the confinement boundary is considered to be a non-credible event, which will not occur over the life of the PFSF. Not withstanding, this hypothetical accident was analyzed and it was concluded that the radiation dose at the OCA boundary resulting from a hypothetical canister breach accident would be less than the 5 rem to the whole body or any organ as specified in 10 CFR 72.106 (b).

100% Blockage of Air Inlet Ducts

SAR Section 8.2.8 discusses a complete blockage of the air inlet ducts event. Since the HI-STORM storage casks have four air inlet ducts 90° apart and the TranStor storage casks have four air inlet ducts, with two located on opposing sides of the cask, it is highly unlikely that all air inlet ducts could become blocked. The feature that is relied upon to prevent or mitigate this event is surveillance of the cask vents to verify no blockage has occurred.

Temperatures representative of the thermal performance of each storage cask are remotely monitored by the storage cask temperature monitoring system and trended. Increased temperatures indicate possible blockage of the natural convection air flow path, most likely at the air inlet ducts, and personnel are dispatched to inspect storage casks with high temperatures. Should blockage occur, it will be identified and removed before achieving the steady state temperatures considered in the vendor analyses. If the temperature monitoring system fails, the surveillance frequency is increased until the monitoring is returned to normal operation.

The cask temperature monitoring system is described in SAR Section 5.1.4.4. The temperature monitoring system is classified as not Important to Safety. A failure of the temperature monitoring system would be the loss of temperature assessment of the storage cask. A loss of the system would not adversely affect cask operation safety. Maximum temperature levels caused by a total blockage of the cask ducts takes 4 to 5 days. In the event of a monitoring system failure, operators would be dispatched to perform visual surveillance of the cask vents to verify no blockage occurs.

To help maintain the reliability of the temperature monitoring system, the system is a supervised system capable of detecting monitor system failures and alarming. The system uses alarm management to provide system status, prioritized alarm information and maintenance activity on the system. In addition, procedures will require periodic tests to verify the operability of the monitoring system. The procedures will require performing instrument calibration and verifying proper alarm responses are detected.

Lightning

SAR Section 8.2.9 addresses a lightning strike event. There are no features that are relied upon to prevent or mitigate this event. However, lightning protection measures grounding the restricted area light poles are used to help deter lightning strikes from hitting the storage casks.

Lightning is evaluated by both the HI-STORM and TranStor SARs, which conclude that if hit, the storage system would not be adversely affected and the canister would retain its confinement integrity, such that there would be no releases of radioactivity.

Hypothetical Accident Pressurization

SAR Section 8.2.10 addresses accidental pressurization caused by a hypothetical breach of all fuel rods in the canister and subsequent release of their fission and fill gases to the canister interior. The features that is relied upon to prevent or mitigate this event is the canister design. The vendors' structural analyses evaluate the canister confinement boundary for this accident condition and show that stresses resulting from accident pressure are within applicable ASME BPVC Section III allowables. Since the analyses determined that the canisters would retain their integrity, there are no radiological consequences for this accident.

CHAPTER 8—ACCIDENT ANALYSES

Sections 8.1 and 8.2 Off-Normal Operations and Accidents

8-3 Provide an estimate of potential radiologic consequences for onsite personnel during off-normal and accident conditions.

- 10 CFR 72.24(e) and 72.24(k); NUREG–1567 (Sections 12.4.5 and 12.5.3) and Regulatory Guide 3.48, Section 8.1.2, state the analysis should consider onsite workers at several distances from the source, as well as individuals located at the boundary of the controlled area and the site boundary, and that worker doses potentially resulting from all actions for off-normal and accident-level events and conditions should be included in the analysis.

RESPONSE

Doses to onsite personnel from off-normal events and accidents were considered in the PFSF SAR. Dose rates from two sources of radiation were considered: 1) radiation from radioactivity released into the atmosphere; and 2) radiation from abnormal operations or occurrences resulting in exposures to operating personnel not considered in the PFSF SAR Section 7.4. The following paragraphs discuss treatment of these two radiation sources.

<u>Releases of Radioactivity</u>

Two of the events analyzed in PFSF SAR Section 8 involve release of radioactivity to the atmosphere, the off-normal contamination release (Section 8.1.5) and the hypothetical loss of confinement boundary (Section 8.2.7). Doses at the PFSF OCA boundary, which is the site boundary, were calculated for both accidents and shown to be within the limits specified in 10 CFR 72.106. In addition, a dose to onsite workers was calculated for the off-normal contamination release. PFSF SAR Section 8.1.5.3 states:

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

The χ/Q for the 150 meter distance was calculated in accordance with Regulatory Guide 1.145, based on the same assumptions used to calculate the χ/Q at the nearest distance from the Canister Transfer Building to the OCA boundary (1.0 m/sec wind speed, atmospheric stability class F, with no consideration for plume meander). The dose at the 150 meter distance was calculated as a representative dose at an intermediate distance between the release point and the OCA boundary (which is 500 meters from the assumed release location), that adequately characterizes effects on onsite workers. The doses to onsite workers from this accident are seen to be relatively low, and calculation of doses to workers at additional distances was not considered necessary. PFSF SAR Section 7.1.2 considers the effects of this event on workers inside the Canister Transfer Building, if the off-normal contamination release is postulated to occur inside the Canister Transfer Building and assuming the Co-60 mixes uniformly with the volume of air in this building, stating the following:

> "No surface contamination is expected on the outer surfaces of the canister since process controls are maintained during fuel loading into the canister at the originating nuclear power plants. Assuming the outer surfaces of a canister have removable Co-60 contamination at the maximum levels permitted by Section 10.2.2.1, and all of this is postulated to be released into the Canister Transfer Building atmosphere, general area radionuclide concentrations in the Canister Transfer Building would not exceed 10 CFR 20 Appendix B, Table 1, allowable airborne concentrations for occupational workers."

Dose rates to onsite workers from the hypothetical loss of confinement boundary accident were not evaluated in the PFSF SAR since it is not a credible occurrence.

Radiation from Abnormal Occurrences

Where off-normal events or accidents result in radiation exposure to onsite workers beyond that associated with normal operations (evaluated in PFSF SAR Section 7.4), doses were calculated and are presented in the SAR.

For the case involving blockage of one-half the air inlet ducts (PFSF SAR Section 8.1.3), doses were calculated to a worker conservatively assumed to spend 30 minutes next to the affected cask removing debris from the inlet ducts. The resulting doses presented in PFSF SAR Section 8.1.3.4 were calculated as described in the response to RAI No. 7-2. These dose rates were doubled to arrive at a conservative estimate of dose to workers that remove debris postulated to block 100% of the inlet ducts, discussed in PFSF SAR Section 8.2.8.3.

PFSF SAR Section 8.2.2.3 provides the estimated dose for repair of the TranStor storage cask in the event the concrete is damaged by a tornado-generated missile. The repair procedure involves the installation of grout to fill in the damaged area and return the concrete shield to its original effectiveness. The dose estimate considers the increased radiation intensity to repair workers from the loss of 5.69 inches of concrete, with this task assumed to take place in a radiation field with dose rates approximately 15 times greater than those calculated to be associated with an undamaged TranStor storage cask containing design basis fuel. As discussed in the response to RAI No. 8-6, the HI-STORM storage cask has a thick steel shell and it is likely that damage by a tornado-driven missile would not be repairable simply by grouting. For HI-STORM, corrective action may require returning the damaged storage cask to the Canister Transfer Building, transferring the canister to an undamaged cask, and moving this storage cask back to the pad.

PFSF SAR Section 8.2.6.3 discusses dose rates that could be associated with tipover of a TranStor storage cask, stating:

> "For tipover of a HI-STORM storage cask, it is considered that localized damage to the radial concrete shield and outer steel shell where the cask impacts the pad could result in an increased surface dose rate due to the damage. However, this would not produce a noticeable increase in the dose rates at the RA fence or OCA boundary because the affected area would likely be small (HI-STORM SAR, Section 11.2.3). The maximum concrete crush depth of 2 inches calculated for the TranStor storage cask would approximately double the dose rates in the localized area, but would not significantly affect the overall dose rates from the storage cask (TranStor SAR Section 11.2.10)."

Doses were not calculated to workers involved with recovery from this event since drop/tipover of a loaded storage cask is not a credible scenario.

For the remaining off-normal and accident conditions evaluated in the PFSF SAR, it is considered that there would be no significant dose consequences to onsite workers beyond those associated with normal operations and evaluated in Section 7.4. For the off-normal event involving loss of external electrical power, it is assumed that operators take necessary actions to minimize doses. As stated in Section 8.1.1.3:

> "In the event the transfer operation is interrupted due to loss of external electrical power, operators would take measures as necessary to assure adequate distance and/or shielding between themselves and the transfer casks to minimize doses until such time as electrical power is restored and the transfer process can resume."

Based on the above information, the PFSF SAR not only identifies dose consequences of off-normal and accident conditions to individuals located at the OCA boundary, but also adequately provides estimates of potential radiological consequences from these events to onsite personnel.

CHAPTER 8—ACCIDENT ANALYSES

Section 8.1.5 Off-Normal Contamination Release

8-4 Provide a basis for the assumption that all surface contamination is Co-60.

RESPONSE

Contamination of the canister at the originating nuclear power plant is unlikely because of the specific design features and operational procedures in place to preclude spent fuel pool water from contacting the canister. Surveys are performed at the originating nuclear power plant to assess removable contamination levels on the outside of the canister. Canisters having removable contamination levels in excess of specified limits are decontaminated before they are permitted to be shipped to the PFSF. However, for the purpose of postulating the accident scenario for this analysis, a conservative contamination source was assumed.

If contamination is present on the exterior surface of the canister, it is likely to come from the radioactive particulates suspended in the spent fuel pool water. Radioactive particulates in the pool at the time the spent fuel is loaded into a canister are mostly the long half-life corrosion products from the spent fuel surface, which might be dislodged during fuel movement. The most prominent corrosion products in the spent fuel pool are Co-60, Co-58, Fe-55, Fe-59, Mn-54, Cr-51, and Zn-65. Co-60 has the highest inhalation dose conversion factors and the longest half-life (5.27 years).

Other isotopes may be present in the spent fuel pool water at the nuclear power facility and could be considered as a potential source of contamination. However, many of these isotopes are volatile (such as I-129, I-130, I-131, I-132, I-133, etc.) and would release soon after the canister is removed from the pool. Others have short half-lives and would decay much sooner than Co-60. Some isotopes emit weak Beta radiation (Kr-85 and H-3) and as such do not provide a significant contribution to the exposure of personnel either by direct radiation or inhalation.

Co-60 is recognized by the NRC (NUREG-1536, Standard Review Plan for Dry Cask Storage Systems, Chapter 7, Table 7.1), as being present in the form of crud on fuel rods and is listed as the only nuclide which contributes significantly to the release that doesn't come from failed fuel. Co-60 is the predominant isotope of concern with corrosion and wear products in nuclear power plants. Therefore, the assumption that all the surface contamination on the spent fuel canister is Co-60 provides a conservative approach to assessing the potential effects of this accident scenario.

CHAPTER 8—ACCIDENT ANALYSES

Section 8.1.5 Off-Normal Contamination Release

8-5 (a) Clarify that the dose conversion factor given for intake represents only the inhalation pathway. Provide basis for not calculating an external dose from submersion or an ingestion pathway dose [10 CFR 72.24(e)].

(b) Revise the SAR to include respirable fraction consistent with the assumptions in Section 8.2.7.3. (see RAI 8-8)

RESPONSE

(a) Submersion Dose

Typically, submersion doses are calculated to an individual who is postulated to be immersed in a plume or cloud of airborne radioactivity that includes noble gases. In the case of noble gases, inhalation does not result in retention of the radionuclide in the body as the radionuclides are exhaled. Therefore the predominant dose is received from external radiation emitted by decay of the noble gases in the plume rather than by uptake in the body and internal exposure. In the case of particulates such as Co-60, the predominant dose results from inhalation of the radionuclide into the lungs and subsequent retention in the body, with activity settling in various organs and producing long term internal exposure to radiation.

The submersion dose from this accident was calculated using the equation for calculating the gamma dose rate in air from a semi-infinite cloud given in Regulatory Guides 1.3 and 1.4 (calculation no. 05996.02-UR-3, rev. 1, enclosed). The results of this calculation indicate that the submersion dose to an individual assumed to be located at the OCA boundary (500 meters from the source) for the duration of the contamination release is 3.78 E-5 mrem, and the submersion dose to an onsite worker assumed to be 150 meters from the source of the release is 2.73 E-4 mrem. Section 8.1.5.3 of the PFSF SAR will be revised to account for this external dose due to submersion in the plume and its contribution to the total effective dose equivalent (TEDE), which is the committed effective dose equivalent (CEDE, due to internal dose) plus the external dose.

Ingestion Dose

The ingestion pathway could apply where an individual could reasonably be expected to ingest water or food products contaminated with radioactivity released from the accident. While concentrations of radioactivity in the soil in unrestricted areas would be highest near the OCA boundary, it is not reasonable to consider ingestion occurring in this area due to land usage near the PFSF site. As discussed in Section 2.2.2 of the PFSF Environmental Report, some rangeland used for livestock grazing is located near the PFSF. However, there are no farm crops grown within 2 miles of the PFSF. Therefore, the closest location to the PFSF where ingestion could reasonably be expected to occur is the nearest residence located 2 miles southeast of the PFSF (Section 2.2.3.4 of the PFSF Environmental Report). Land use at this location is considered residential with the potential for back yard gardening activities and a dietary consumption as discussed in NUREG/CR-5512, Section 3.2.1. In addition, Section 2.5.1 of the PFSF Environmental Report states:

> "There are no public or private surface drinking water supplies in the PFSF vicinity. Potable water supplies for the Skull Valley Indian Reservation, and the few scattered ranches or farms along the east side of the valley, are wells drilled into the unconsolidated or semi-consolidated sediments that form the alluvial fan along the base of the Stansbury Mountains. Consequently, there is no potable surface water supply that could be subject to normal or accidental effluents from the facility."

There are no bodies of water in the vicinity of the PFSF OCA boundary.

The ingestion dose from this accident was calculated using conservative assumptions at the location of the nearest resident (calculation no. 05996.02-UR-7, enclosed). The results of this calculation indicate that the ingestion dose to an individual residing 2 miles from the PFSF site in the direction of the plume is negligible (5.3 E-7 mrem). Since the dose to an individual who could reasonably be expected to receive a dose from the ingestion pathway is negligible, inclusion of an ingestion pathway dose in the PFSF SAR is determined to be unnecessary.

(b) PFSF SAR Section 8.2.7.3 states:

> "Based on Table XX of Reference 25, 95 percent of Co-60 and Sr-90 particulates are greater than 10 microns aerodynamic diameter and are non-respirable. Therefore, a respirable factor of 0.05 was applied to these particulates to account for inhalation of those particulates having an aerodynamic diameter less than 10 microns."

PFSF SAR Section 8.1.5.3 will be revised to apply this respirable factor of 0.05 to determine uptake of Co-60 by inhalation in the off-normal contamination release event.

SAR CHAPTER 8—ACCIDENT ANALYSES

Section 8.2.2.3 Accident Dose Calculations [Extreme Wind]

8-6 Evaluate the other storage systems or otherwise explain why the TranStor system would be bounding.

- Only the consequences for the TranStor system are evaluated.

RESPONSE

The accident dose analysis presented in the SAR Section 8.2.2 evaluates both storage cask systems and determines that no radioactivity would be released in the event of a tornado and that dose rates at the OCA boundary would not be affected by damage to storage casks from tornado-driven missile strikes.

The dose calculations for repair of a TranStor storage cask after a missile strike are applicable to the TranStor storage cask only. It was not intended that this analysis be bounding for both storage cask vendors.

The missile strike for the HI-STORM storage cask is analyzed in Appendix 3.G of the HI-STORM SAR. The deformation results are summarized in Section 3.4.8.1. For the intermediate missile strike on the side of the cask, the damage is similar to that for the TranStor storage cask (the outer overpack shell is penetrated and the concrete is dented to a depth of 5.67 inches). The HI-STORM SAR does not discuss a repair procedure and the associated radiation dose from such a repair. Since the outer shell of the HI-STORM storage cask is constructed of ¾ inch thick steel, a simple grout repair similar to that described for the TranStor storage cask would not restore the cask to its original condition.

In lieu of a repair-in-place procedure the HI-STORM storage cask would be examined to determine the extent of damage. If required the MPC would be transferred to another HI-STORM overpack and the damaged overpack repaired or permanently removed from service. The dose that could be expected during the transfer from one storage cask to another would be similar to that presented in SAR Chapter 7, Table 7.4-1, Estimated Personnel Exposure For HI-STORM Canister Transfer Operations, of 198.7 person-mrem (this dose is based on revised SAR Table 7.4-1; see response to RAI 7-3).

The repair-in-place procedure for the TranStor storage cask results in a total dose of 150 person-mrem while the transfer operation for the HI-STORM storage cask (which would be performed in lieu of a repair) results in a total dose of 198.7 person-mrem.

SAR CHAPTER 8—ACCIDENT ANALYSES

Section 8.2.6 Hypothetical Storage Cask Drop/Tip-Over

8-7 Describe actions to be taken in response to a cask drop or handling accident.

- A surveillance requirement in technical specifications, generally found in ISFSI licenses and cask certificates of compliance, requires the return of fuel from a dropped cask to the spent fuel pool so that the cask can be evaluated for further use.

RESPONSE

As stated in SAR Section 8.2.6, the hypothetical drop / tipover of a storage cask is classified as Design Event IV as defined by ANSI/ANS-57.9. Storage cask tipover accidents, and storage cask vertical end drop accidents from heights greater than 10 inches, are hypothetical events, since there are no credible causes for the events. A storage cask tipover, and storage cask vertical end drop from 10 inches, are analyzed in order to assess potential consequences of such accidents. For drop and tipover accidents the canister would retain its integrity, and the canister and its internals would continue to perform their safety functions.

The storage cask and canister is design to withstand the loads produced by these events. However, a cask tipover accident may cause some localized damage to the storage cask radial concrete shield and outer steel shell (HI-STORM) where the cask impacts the surface. The damaged cask would be returned to the Canister Transfer Building where the canister would be transferred to another storage cask and returned to storage. The damaged storage cask would be repaired or discarded.

For a storage cask drop from below 10 inches or a side impact due to mishandling the canister would again retain its integrity, and the canister and its internals would continue to perform their safety functions. The storage cask would be examined by the PFSF staff for external damage. Repairs would be initiated as necessary with the canister in place or the canister would be transferred to another storage cask and returned to storage. The damaged storage cask would be repaired or discarded.

The hypothetical loss of confinement barrier (canister breach) is classified as Design Event IV as defined by ANSI/ANS-57.9. As discussed in SAR Section 8.2.7 this is not a credible accident at the PFSF. However, should cask drop or handling accident result in a hypothetical canister breach, a plan has been developed to recover from such an event. The recovery plan scenarios for a hypothetical canister breach are discussed in SAR Section 8.2.7.4.

CHAPTER 8—ACCIDENT ANALYSES

Section 8.2.7.3 Accident Dose Calculations [Hypothetical Loss of Confinement Barrier]

8-8 Provide basis for a respirable fraction of 5 percent for Co-60 and Sr-90.

- The respirable fraction should be consistent with Section 8.1.5 assumptions. (see RAI 8-5(b))

RESPONSE

PFSF SAR Section 8.2.7.3 states the following in regards to this respirable fraction for particulates:

> "Based on Table XX of Reference 25, 95 percent of Co-60 and Sr-90 particulates are greater than 10 microns aerodynamic diameter and are non-respirable."

Reference 25 of PFSF SAR Section 8 is SAND80-2124, Transportation Accident Scenarios for Commercial Spent Fuel, Sandia National Laboratories, dated February 1981. A section of this Sandia report entitled "From Environment to People", beginning on pg 8 of this report, states:

> "Once radioactive material has been released to the environment, a number of other factors become important in determining whether the radioactive material will reach people. Two important factors are the fraction of particles smaller than 10 microns aerodynamic diameter (particles less than this size are respirable) and the fraction of the material that becomes suspended in air. Table XX presents the values for these variables: volatiles, particulates and noble gases ... Particles released via the burst-rupture mechanism have been characterized in Reference 25. Table 42 in this reference indicates that no more than 3 percent of the particles released are smaller than 10 microns. So a value of 5 percent was assigned."

An indication of the level of conservatism of the 5% respirable fraction is shown by the NRC analysis that was performed to evaluate the effects of an explosion on a storage cask (FR Vol. 54, No. 86, p. 19379). The NRC determined the respirable fraction in this study to be 0.005%, a factor of 1000 smaller than that used for the PFSF SAR.

CHAPTER 8—ACCIDENT ANALYSES

Section 8.2.9 Lightning

8-9 Justify the statement in Section 8.2.9.2 that states that lightning strikes would not affect canister integrity.

RESPONSE

If a storage cask were hit by lightning, the path to ground would be through the steel shell of the storage cask. The steel shell acts as a mast type lightning protection system, which uses ground wires surrounding a structure to direct the current to ground. The mast type system is used in cases where it is desirable to prevent the lightning current from contacting any part of the protected structure (NFPA 780), in this case the canister. The canister is surrounded by the cask steel and is therefore, not a ground path. The cask shell would provide a direct path to ground. Since the effects of the lightning would be limited to the cask shell, a lightning strike would not affect canister integrity.

The text for SAR Section 8.2.9, Lightning, will be revised to read as follows:

8.2.9 Lightning

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

8.2.9.1 Cause of Accident

This event would be caused by meteorological conditions at the site. Lightning would probably strike one of the grounded metal light poles in the vicinity of the storage pads since they are substantially higher than the storage casks (approximately 120 ft high). However, since the light poles are approximately 500 feet apart, it is possible that lightning may strike a cask that is not within the zone of protection offered by the light poles. NFPA 780 specifies the zone of protection for a 20 foot high structure (storage cask) as a 75 foot radial area around a 120 foot high structure (light pole).

8.2.9.2 Accident Analysis

If a storage cask were hit by lightning, the path to ground would be through the steel shell of the storage cask. The canister is surrounded by the cask steel and is therefore not a ground path. Since the effects of the lightning would be limited to the cask shell, a lightning strike would not affect canister integrity. The absorbed heat would be insignificant due to the very short duration of the event.

If the lightning entered or exited the TranStor cask via the concrete shell, which is not fully surrounded by steel, some local spalling of concrete might occur; however, storage cask operation would not be adversely affected (Reference TranStor SAR Section 11.2.9.2). Since the concrete in the HI-STORM cask is completely encased by steel, the concrete would not sustain any damage from the lightning.

8.2.9.3 Accident Dose Calculations

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

CHAPTER 9—CONDUCT OF OPERATIONS

Section 9.1.2.1 Onsite Organization

9-1 Describe in more detail the plan to provide sufficient managerial depth for qualified backup staff in absence of an incumbent [NUREG–1567 (Section 13.4.1)].

- The general manager also functions as the chief operating officer. Section 9.1.2.1 of the PFSF states that the general manager will rotate the backup responsibility among the functional area leads to develop a senior capability for site direction.
- The personnel qualification requirements provided in Section 9.1.3 of the PFSF SAR note that only two of the functional area leads are required to have college degrees, and several of these individuals have narrowly specialized education and experience requirements.
- Similarly, it is not clear from the organization description in the PFSF SAR that the staff members in each functional area will have sufficient qualifications to backup the functional lead staff member.

RESPONSE

The statement in the second paragraph of Section 9.1.2.1 of the PFS Safety Analysis Report (SAR) that "The General Manager/Chief Operating Officer shall designate a lead person as a backup during his/her absence and rotate this responsibility among various leads to develop senior capability for site direction" was not intended to require that this rotation be among <u>all</u> leads or to suggest that individuals could be assigned such responsibility without possessing the appropriate background and experience to assume such duties. While minimum qualification requirements for each of the lead positions are listed in SAR Section 9.1.3, it is expected that most lead individuals will have education and experience levels beyond the minimum required. It is the intent of the PFS to encourage staff members to improve their education and experience levels while employed, and to develop as many of the lead staff as possible into individuals fully qualified to assume the General Manager/Chief Operating Officer position. It is quite possible that over a period of time many, if not all, persons in lead positions may possess sufficient education and experience to assume the duties of backup General Manager/Chief Operating Officer.

It will be a job expectation for the General Manager/Chief Operating Officer to develop the depth of the lead staff's qualifications. This will provide a wider group of people available for promotion should the General Manager/Chief Operating Officer position become vacant. Designating a single position or individual as backup would narrow the available talent for a replacement and demotivate other persons who may be interested and capable of advancing to senior management. The PFS desires to allow lead individuals who possess the appropriate background and experience to serve as backup General Manager/Chief Operating Officer in order to develop senior management resources. It is the intention of the PFS to allow the General Manager/Chief Operating Officer the discretion to designate his/her backup among those leads who have sufficient background and experience and who show the appropriate capability and potential to assume such duties. In the unlikely event that the initial PFS staff meet only the minimum qualification requirements specified in SAR Section 9.1.3, the backup General Manager/Chief Operating Officer duties will be limited to the Radiation Protection Manager or the Lead Nuclear Engineer until such time as other lead individuals acquire equivalent education and/or experience.

Similarly, it is expected that certain staff members within each functional area will have education and experience levels beyond the minimum required. It is also desired to develop supervisory skills in employees who demonstrate the appropriate ability and potential by allowing them to serve as a backup when their functional lead is off site for short periods.

CHAPTER 9—CONDUCT OF OPERATIONS

Section 9.1.2.2.2 Radiation Protection Manager

9-2 Clarify the responsibilities of the Radiation Protection Manager to provide consistency throughout the PFSF SAR.

- The operational organization presented in Figure 9.1-3 states that this individual will have responsibility for industrial safety. The functions, responsibilities, and authorities of this individual as presented in Section 9.1.2.2.2 of the PFSF SAR however, do not include industrial safety.

RESPONSE

The Radiation Protection Manager is identified on Figure 9.1-3 as having overall responsibility for the area of industrial safety because of the frequent intrusion of industrial accidents into the areas of radiological protection. Placing both the radiological and non-radiological human protection functions under the Radiation Protection Manager's control is an effective means of ensuring integration of the overall concern for personnel safety at the PFS. Section 9.1.2.2.2 of the PFS SAR will be revised to add the statement "The Radiation Protection Manager is also responsible for industrial safety at the PFSF."

CHAPTER 9—CONDUCT OF OPERATIONS

Section 9.1.2.2.10 Lead Nuclear Engineer

9-3 (a) Clarify requirements for a nuclear engineer onsite.

- According to Figure 9.1-3 and Section 9.1.2.2.10 of the SAR, all staff assigned to the Nuclear Engineering functional area, other than the Lead Nuclear Engineer, are located at offsite utility facilities. Therefore, plans for providing a qualified backup for the Lead Nuclear Engineer should be addressed.

(b) Indicate if a qualified nuclear engineer is required onsite to conduct operations and, if so, how this requirement will be satisfied. If not, explain why not.

RESPONSE

(a) As indicated in the response to the Question 9-3(b), below, the lead nuclear engineer is a position which was added to strengthen the facility staffing, but which is not necessary for day-to-day operations. There is no requirement to have a nuclear engineer on site, as there are no functions or responsibilities of this position which would require his/her presence at all times. The three additional nuclear engineering staff members are stationed at the site and are assigned as necessary for the off-site function of ensuring that the loading of canisters is properly conducted at the originating nuclear power plant, and for ensuring that the PFS "start clean/stay clean" philosophy is maintained. The three nuclear engineering staff members will rotate the off-site responsibilities, and it is not expected that all three nuclear engineering staff members will be off-site at the same time. It is intended that while the staff nuclear engineers are on site, they will assist the lead nuclear engineer in assuring that the oversight functions of procedure review, facility change monitoring, etc., as described in SAR Section 9.1.2.2.10, are enhanced and accurately completed. This would enable any one of the three nuclear engineers to provide a qualified backup for the lead nuclear engineer in the event of his/her absence.

(b) There is no NRC guideline or regulation that requires the on site presence of a nuclear engineer for operation of a 10 CFR Part 72 facility. PFS has decided that it would be prudent to have on staff, but not necessarily always on site, a qualified nuclear engineer to oversee the general technical direction of the facility and to review procedures and proposed facility changes. We are not establishing the requirement that a nuclear engineer be on site to conduct operations, including those important to safety, as there exists no corresponding functional role or reason. Normal operations are discussed in SAR Section 9.4, and the PFS staff will be appropriately qualified, trained and their fitness-for-duty ensured to allow the conduct of operations, independent of the presence or absence of a nuclear engineer. The lead nuclear engineer has his/her input and control at the level of equipment design and modification and procedure approval and preparation. The job responsibility outline in Section 9.1.2.2.10 emphasizes the overview nature of this position as one which steps back and reviews the full scope of site activity.

CHAPTER 9—CONDUCT OF OPERATIONS

Section 9.1.3.1.4 Lead Mechanic/Operator

9-4 (a) Provide justification for the scope of the functions, responsibilities, and authorities assigned to the Lead Mechanic/Operator.

- In addition to being the manager for this functional area, the Lead Mechanic/Operator must be qualified as a locomotive operator, a certified storage facility operator, and a certified welder.

(b) Justify why requiring the manager for this functional area to conduct welding operations on SSCs important to safety does not remove an important supervisory and oversight function for such operations.

RESPONSE

(a) The PFSF is structured to utilize a staff which is shared between maintenance and operations. It is the intent of the PFS to utilize the full capability of all its staff resources. This concept is emphasized in order to keep people active in tasks which complement each other. The roles of maintenance and operations at the PFS are such complimentary functions. Mechanics will perform predictive and periodic maintenance during times when fuel is not being transported or handled, and will perform functional operations during times when fuel is being transported or handled. The Lead Mechanic/Operator is generally a non-task performing supervisor, who is also trained and certified to perform the functions of the other mechanics. It is the intent of the PFS to use the Lead Mechanic/Operator to provide the supervisory role over the other mechanics while also maintaining his/her qualifications by performing the same tasks. The roles outlined for the Lead Mechanic/ Operator are consistent with a full-time position and it is anticipated that this configuration of duties will make excellent use of a highly qualified person.

(b) Welding activities on SSCs important to safety are expected to occur only on an infrequent basis due to the design of the facility and the packaging and storage systems. It is anticipated that some reparative welding may be necessary from time to time, and it is therefore desirable to have a certified welder on staff. Welding is more of a skill maintained in reserve at the PFS than a function performed with any expected frequency. Welding activities will be reviewed by Quality Assurance and Nuclear Engineering prior to their performance. Since the Lead Mechanic/ Operator is not only a manager but also a functional leader in these craft areas, it does not compromise the independence of the oversight function if the work is reviewed by someone in a separate department. Supervision of the Lead Mechanic/ Operator is provided by the General

Manager/Chief Operating Officer. Additional oversight of welding operations is provided by Radiation Protection for health physics and safety issues.

CHAPTER 9—CONDUCT OF OPERATIONS

Section 9.1.3.1.13 Emergency Preparedness Coordinator

9-5 Clarify the minimum qualification requirements for and the responsibilities assigned to the Emergency Preparedness Coordinator.

- Qualification requirements for the Emergency Preparedness Coordinator include "experience in providing training." Section 9.3.4 of the SAR also states that the Emergency Preparedness Coordinator will be the primary source for general employee training.

- A comparison of the general employee training topics presented in Section 9.3.2.1 with the qualification requirements for the Emergency Preparedness Coordinator presented in Section 9.1.3.1.13 indicates that the qualifications of this individual may not be sufficient for this assignment.

- Additionally, Section 9.1.2.2.14, which describes the functions, responsibilities, and authorities of the Emergency Preparedness Coordinator, requires this individual be a qualified radiation protection technician. Section 9.1.3.1.13 does not include this requirement.

RESPONSE

The responsibilities of the PFS Emergency Preparedness Coordinator necessitate only a part-time assignment of a staff member. It was therefore decided to review which other PFS functional job specialty would most enhance and compliment the skills of the Emergency Preparedness Coordinator. With experience in the nuclear power industry and in radiation protection, as described in SAR Section 9.1.3.1.2, it was evident that this individual's skills would complement the training oversight function. Of the eleven topics covered in general employee training, as outlined in SAR Section 9.3.2.1, the area of radiation control procedures and practices was considered the most significant, with the emergency plan and procedures also included. Therefore, the Emergency Preparedness Coordinator was chosen to be the normal instructor providing these areas of training, based on this individual possessing the requisite expertise. The Emergency Preparedness Coordinator will thus be the primary instructor for general employee training in terms of the total volume of training given.

As stated in SAR Section 9.3.4, the Emergency Preparedness Coordinator is responsible for the "administration" of the training program and maintaining the training records. It was not intended to imply that the Emergency Preparedness Coordinator will be giving training in all of the GET areas. The Emergency Preparedness Coordinator will coordinate training by ensuring appropriate, qualified instructors from on-site or contracted training services conduct all training activities. The Emergency Preparedness Coordinator will ensure that appropriate instructors are assigned as per Section 9.3.4 for areas other than those in which he/she has special expertise.

The responsibilities of the Emergency Preparedness Coordinator include the performance of the health physics technician role as assigned based on facility need, emergency preparedness coordination, and on-site training coordination. SAR Section 9.1.3.1.13 will be revised to add the statement: "This individual shall also have a minimum of four years of working experience in radiation protection."

CHAPTER 9—CONDUCT OF OPERATIONS

Section 9.1.4 Liaison with Outside Organizations

9-6 Provide justification for the statements in the last paragraph of Section 9.1.4 regarding the responsibilities of the PFSF facility staff to oversee and monitor the fabrication and storage/ transfer/transportation technology for the canisters.

- It is not clear from the qualification requirements presented in Section 9.1.3 that facility staff will be capable of these responsibilities. The specific staff positions assigned responsibilities should be identified so the sufficiency of the qualifications can be evaluated.

RESPONSE

The PFS Nuclear Engineering and Quality Assurance staff will be responsible for oversight of the fabrication of all PFS hardware. This staff will consist of both degreed engineers and quality assurance/quality control staff experienced in QA programs that meet the requirements of 10 CFR Part 72, Subpart B. The oversight of the fabrication will be augmented by contract personnel on an as needed basis. Contract personnel will include experts in the fields of quality assurance/quality control, welding, materials, the ASME Boiler & Pressure Vessel Code, non-destructive examinations, etc.

The referenced statement is admittedly unclear. The last paragraph in Section 9.1.4 of the SAR will be revised to read:

> "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 Quality Assurance 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 Quality Assurance staff."

CHAPTER 9—CONDUCT OF OPERATIONS

Section 9.2.1 Administrative Procedures for Conducting Test Program

9-7 Provide a complete and consistent statement of test procedure review responsibilities.

- Section 9.2.1 states that test procedures will be reviewed and approved by the responsible line manager. Section 9.1.1.2.3 assigns this authority to the Safety Review Committee. Section 9.1.2.2.1 gives this responsibility to the General Manager/Chief Operating Officer. Section 9.2.1 provides that review and approval of procedures involving SSCs important to safety are performed by the Operations Review Committee.

RESPONSE

It is agreed that one of the referenced statements is incorrect and that two other statements require clarification. The Safety Review Committee consists of a minimum of 5 members whose responsibility is to advise the Board of Managers on all matters related to SSCs important to safety, as set forth in SAR Section 9.1.2.1.1. The Operational Review Committee (ORC) consists of a quorum of 8 members whose responsibility is to support the General Manager/Chief Operating Officer in the review and assessment of site operations, as set forth in SAR Section 9.1.1.2.3. The ORC reviews and approves procedures involving SSCs important to safety, as stated in SAR Section 9.2.1. The statement in the third sentence of the second paragraph of SAR Section 9.1.1.2.3 regarding the Safety Review Committee is incorrect. This sentence will be revised to remove the words "and approving," as the Safety Review Committee has no approval authority or responsibility.

The General Manager has final approval authority for all procedures, and does so after all other reviews, including ORC approval for SSCs important to safety, have been completed. The seventh sentence in SAR Section 9.1.2.2.1 will be revised to clarify this protocol, and state: "The General Manager/Chief Operating Officer provides final approval of procedures for facility operations, maintenance, equipment inspections, administration and security, after all other required approvals have been obtained."

The statement in the second sentence of the second paragraph of SAR Section 9.2.1 regarding approval of test procedures by responsible line management, is correct though perhaps not absolutely clear if read independently. This sentence will be revised to read: "Review and approval of test procedures by the responsible line manager is required before submission for final approval or ORC review (if required)."

CHAPTER 9—CONDUCT OF OPERATIONS

Section 9.2.2 Pre-operational Test Plan

9-8 Provide justification for the statement in Section 9.2.2 that the PFSF will meet the general design criterion of 10 CFR 72.122(f) because preoperational tests will be performed in accordance with approved procedures to be developed and implemented in accordance with the PFSF quality assurance (QA) program.

- The design criterion in 10 CFR 72.122(f) specifies that systems and components important to safety must be designed to permit inspection, maintenance, and testing. Preparing and implementing procedures in accordance with an approved QA program does not, of itself, guarantee that systems and components were designed to meet this regulatory requirement.

RESPONSE

The last sentence in SAR Section 9.2.2 incorrectly references 10 CFR 72.122(f), and will be revised to reference 10 CFR 72.122(a). SAR Table 4.4-1 will also be revised to add reference to Section 9.2.2 for 10 CFR 72.122(a) and delete the same reference from 10 CFR 72.122(f).

CHAPTER 9—CONDUCT OF OPERATIONS

Section 9.2.3 Operational Readiness Review Plan

9-9 Include nuclear safety in the list of areas to be examined in the operational readiness review plan discussed in Section 9.2.3.

- NUREG–1567 (Section 13.4.2.2) recommends nuclear safety be included in the areas to be examined in the operational readiness review plan.

RESPONSE

SAR Section 9.2.3 will be revised to add the following additional bullet to the areas covered by the ORR:

- "Nuclear Safety - There exist no unresolved safety questions regarding the facility or facility operation. All criticality controls and fuel accountability controls will be approved and distributed in an appropriate procedural form. All procedures for the loading of fuel into canisters at the originating power plants will be ready and approved by the steps prescribed in the Quality Assurance Program."

CHAPTER 9—CONDUCT OF OPERATIONS

Section 9.3 Training and Certification of Personnel

9-10 (a) Per the requirements of 10 CFR 72.190, describe the operator requirements for the equipment and controls that have been identified as important to safety.

(b) Per the requirements of 10 CFR 72.192, provide information on the training program to show a systematic approach to training, proficiency testing, and certification of personnel.

(c) Per the requirements of 10 CFR 72.194, provide information on the program for the certification of the physical condition and the general health of personnel who will operate equipment and controls that are important to safety.

RESPONSE

(a) For equipment and controls that have been identified as important to safety, a designated check-off list of the required training will be prepared along with the operating procedures. Personnel operating or supervising the operation of equipment or controls important to safety will be certified in such operation. The method of certification will be by an approved training and requalification program maintained by the facility. The prerequisites for procedures effecting equipment and controls important to safety will state specifically that the person performing the procedure must hold the applicable certification.

(b) The training program will be developed using a job and task analysis to define the various functions which must be performed by different staff members. Each prescribed task will be covered in a training module which will be presented by a person who is qualified to perform the procedure or to train persons on the procedure. The person performing the training will follow a defined curricula which will outline the steps that are required to perform the task. The procedures to be used in performing the task will be used as a basis of the training curricula, so that persons being trained to perform a task (particularly those which are important to safety) will be trained in the precise manner and steps used in performing the task. At the conclusion of task training, proficiency testing will be administered to ensure that proper understanding has been achieved by the person being trained. A test will be prepared and graded by the instructor and will be retained in the facility records for a two year period.

The certification of personnel in functional areas such as fuel handling will be documented in a personnel training file, which will include an outline of the tasks they are authorized to perform, the date training was provided, the person providing the training, and a certification indicating that a suitable proficiency test was passed. Retraining and refresher training will be provided at intervals that are appropriate to the specific task. Retraining will involve a review of the basic tasks plus special attention to those items within the task which have undergone change. When new equipment is added or modifications in existing equipment of a significant nature are made, procedures will be modified and retraining on the revised procedure and equipment by a qualified instructor will be provided to those persons already certified prior to operation of that equipment.

(c) The program to ensure the certification of the physical condition and general health of personnel who will operate equipment controls important to safety will require all staff members designated as operators of equipment or controls important to safety to have a physical examination by a licensed physician every two years in accordance with NRC Form 396. Observation of continued fitness-for-duty will be controlled by procedures and training of staff personnel. These procedures and training will include the information necessary to ensure that staff reporting for the performance of work which involves the operation of equipment or controls identified as important for safety are capable of performing such duties without impairment. The procedures and training will include the authority for any trained staff member to deny the right of a person to operate equipment or controls important to safety should they fail to meet the standards of fitness-for-duty. Permanent conditions of staff members that could cause impaired judgment or motor coordination will be considered for accommodation by the physician performing the physical examination using NRC Form 396. Temporary conditions causing impaired judgment or motor coordination will be considered in the procedures as a possible cause for restricted performance of these duties if, in the opinion of trained personnel, further evaluation by a physician is required.

CHAPTER 9—CONDUCT OF OPERATIONS

Section 9.4.1.2 Procedure Preparation

9-11 (a) Clarify the content of procedures to be developed for activities important to safety.

- An illustrative procedure format and synopsis should be provided to present the proposed depth of procedure coverage as recommended by NUREG–1567 (Section 13.4.4.1).

(b) Justify why the following specific procedural components discussed in NUREG–1567 (Section 13.4.4.1) have been omitted.

- Specification of calibration requirements
- Identification of preceding and follow-on actions
- Specification of physical or operating limits to be observed during procedure execution
- Notifications required before and after procedure execution

RESPONSE

NUREG-1567, Section 13.4.4.1 was utilized, as applicable, in drafting SAR Section 9.4.1.2. Nonetheless, in order to ensure that the use of this guidance is made clear, Section 9.4.1.2 will be revised to read:

> Procedures will be generated for all activities important to safety, and will include the following format and depth of coverage:
>
> - Purpose and role in broader scope function
> - Personnel required per shift by staff position and general function (e.g., function performance, QA, radiation monitoring)
> - Continuous or single (or double) shift operation
> - Prerequisites for readiness, such as
>   - calibrations to be performed or checked
>   - instrumentation to be on hand
>   - tools and special equipment to be on hand
>   - notifications (with lead times)
>   - check/set equipment controls (e.g., physical travel limits for overhead crane)
>   - check environmental or other monitors for acceptable range
>   - identification of subject(s) of function (e.g., canister to be transferred, cask to be retrieved)
>   - log and forms to be completed on hand
>   - preceding function

- Series of operations, including results, projected times, projected instrument and gauge readings, controls to be used in performance (e.g., torque, time at pressure, and threshold limits requiring contingency actions (such as hold, initiating a contingency sequence, notification))
- Records to be completed during operation and distribution
- Record and notification upon Completion
- Identification of following function

CHAPTER 9—CONDUCT OF OPERATIONS

Section 9.4.2.1 Records Management System

9-12 Provide clarification of the responsibilities and authority of the Technical Support Manager.

- The Technical Support Manager is not identified in the text or organization charts presented in Section 9.1, Organizational Structure, of the SAR; however, the position is discussed in Section 9.4.2.1.

RESPONSE

The Technical Support Manager referenced in SAR Section 9.4.2.1 is actually the Administrative Assistant, as shown on Figure 9.1-3. The first sentence of SAR Section 9.4.2.1 is therefore incorrect, and will be revised to read:

> "Records relating to the historical operation of the facility will be maintained by the Administrative organization, under the responsibility of the Administrative Assistant."

CHAPTER 9—CONDUCT OF OPERATIONS

Emergency Plan Section 1 Facility Description

9-13 (a) Provide additional PFSF Emergency Plan (EP) information as specified in Appendix C, Section C.4.1.1 of NUREG–1567.

(b) Justify why the following facility description information from NUREG–1567 is missing from the EP:

- Onsite routes for transferring spent nuclear fuel to and from storage
- Specific locations of PFSF gates
- Locations of homes on the reservation

RESPONSE

(a) The following changes to the EP figures have been made:

- Figure 1-1 has been revised to show the location of the Intermodal Transfer Point.
- Figure 1-2 has been revised to include a north indicator.
- Figure 1-3 has been revised to include a bar scale.
- Figures 1-4 has been revised to label all gates, show the diversion berm and retention basin, and a bar scale.
- Figure 1-4 has also been revised to show typical onsite routes for transferring material to and from storage. (It should be noted that the storage area is an open area, designed to allow access to the casks from several directions).

These new figures will be incorporated into the EP.

(b) Information on onsite routes for transferring spent nuclear fuel to and from storage and specific locations of PFSF gates is addressed in (a), above. EP Figures 1-1 and 1-2 provide the information recommended in NUREG-1567, Section C.4.1.2, for a general map (approximately 10 mile radius) and a USGS topographical map including population centers within 1 mile. There are no population concentrations or facilities within a mile of the PFSF. Most of the homes on the reservation are concentrated in the Goshute Village, which is labeled on Figure 1-2. There are also two homes located approximately 2 miles southeast of the PFSF. Figure 1-2 has been revised to show their approximate location from the site.

CHAPTER 9—CONDUCT OF OPERATIONS

Emergency Plan Section 4 Organization

9-14 Provide a discussion in the EP explaining how radiation monitoring teams and the fire brigade will be staffed by available staff during an alert.

The EP provides insufficient information regarding the staffing of radiation teams and the fire brigade. Staffing requirements for the Emergency Response Organization below the supervisory positions for both normal working hours and off-hours should be provided to support an NRC evaluation of whether or not sufficient staffing is available for functions such as radiological assessment, fire fighting, and security control, among others.

RESPONSE

The schedule for submitting the response to this RAI is June 15, 1998.

CHAPTER 9—CONDUCT OF OPERATIONS

Emergency Plan Section 9.5.2 Emergency Planning Records

9-15 Provide the information missing from Section 9.5.2 of the EP.

Section 9.5.2 terminates in an incomplete sentence on page 9-4. The sentence should be properly ended and the remaining information provided.

RESPONSE

EP Section 9.5.2 was incomplete as previously submitted, and will be revised to read:

"Emergency Plan implementing procedures will detail requirements for identification, storage, traceability and length of time emergency planning records are to be retained. Emergency planning records include, but are not limited to:

- Training and retraining (including lesson plans and test questions),
- Drills, exercises, and related critiques,
- Inventory and locations of emergency equipment and supplies,
- Maintenance, surveillance, calibration, and testing of emergency equipment and supplies,
- Letters of agreement with offsite support organizations,
- Reviews and updates of the Emergency Plan,
- Notification of onsite personnel and offsite response organizations affected by an update of the Emergency Plan or its implementing procedures,
- Incident Reports and associated documentation, including:
  - cause of the incident
  - personnel and equipment involved
  - extent of injury and damage (onsite and offsite) as a result of the incident
  - locations of contamination with the final decontamination survey results
  - corrective actions taken to terminate the emergency
  - actions taken or planned to prevent a recurrence of the incident
  - onsite and offsite assistance requested and received
  - any program changes resulting from a critique of emergency response activities

CHAPTER 10—OPERATING CONTROLS AND LIMITS

Section 10.2.1.1 Fuel Characteristics

10-1 (a) Provide a reference for allowable decay heat for Zircaloy cladded PWR and BWR fuels.

(b) Revise the text to include a reference that provides similar information for Zircaloy fuel assemblies.

- Table 2.1.8 (in Reference 1 cited in the SAR) provides allowable decay heat values for stainless steel fuel assemblies.

RESPONSE

(a) The allowable decay heat for Zircaloy clad PWR and BWR fuel assemblies is provided in the HI-STORM SAR (Reference 1 in the PFSF SAR) and the TranStor SAR (Reference 2 in the PFSF SAR). Specific references are provided in (b) below.

(b) The text for SAR Section 10.2.1.1, Decay Heat, will be revised to list the references for the decay heat values as shown below.

HI-STORM 100 System:

PWR: Zircaloy ≤ 1.177 kW per assembly (See Table 2.1.6 of HI-STORM SAR)

Stainless steel ≤ 0.662 kW per assembly (See Table 2.1.8 of HI-STORM SAR)

BWR: Zircaloy ≤ 0.3989 kW per assembly (See Table 2.1.6 of HI-STORM SAR)

Stainless steel ≤ 0.079 kW per assembly (See Table 2.1.8 of HI-STORM SAR)

TranStor Storage System:

PWR: ≤ 1.083 kW per assembly (See Table 12.2-1 of TranStor SAR)

BWR: ≤ 0.426 kW per assembly (See Table 12.2-1 of TranStor SAR)

CHAPTER 10—OPERATING CONTROLS AND LIMITS

Section 10.2.1.1 Fuel Characteristics

10-2 (a) Clarify the discrepancy in cooling time (≥5 yr) and maximum initial fuel enrichment (≤4.2) requirement values specified in the SAR compared to the values presented in Table 2.1-8 of Reference 1 cited in the SAR.

(b) Provide justification if there is deviation in the specified limits.

- Table 2.1.8 (in Reference 1 cited in the SAR) specifies minimum cooling time of 10 yr for stainless steel and initial BWR fuel enrichment of 4 wt. percent max. for HI-STORM and 4.4 wt. percent for TranStor storage systems.

RESPONSE

(a) The following clarification is provided for the HI-STORM 100 System, Section 10.2.1.1, "Initial Enrichment" and "Cooling Time". The SAR text will be revised to incorporate this clarification.

Initial Enrichment: HI-STORM 100 System:

PWR: See HI-STORM SAR Table 2.1.6 (Zircaloy) or Table 2.1.8 (stainless steel)
BWR: See HI-STORM SAR Table 2.1.6 (Zircaloy) or Table 2.1.8 (stainless steel)

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

PWR: See HI-STORM SAR Table 2.1.6 (Zircaloy) and Table 2.1.8 (Stainless steel)
BWR: See HI-STORM SAR Table 2.1.6 (Zircaloy) and Table 2.1.8 (Stainless steel)

(b) There is no deviation in the specified cooling time or initial enrichment limits. The text has been clarified in (a) above.

CHAPTER 10—OPERATING CONTROLS AND LIMITS

Section 10.2.1.2 Canisters Authorized for Use at the PFSF

10-3 (a) Describe the procedure to verify that loading and shipping documentation provided by the originating power plant contains the required information to assure that the as-received fuel and the storage canisters meet the vendor specifications.

(b) Revise this section of the SAR by incorporating brief descriptions of review procedures for the shipping documents and the associated procedure to validate these documents.

- It is not clear in the SAR what review procedures will be used at the PFSF site as the basis for accepting or rejecting the canisters for storage.

RESPONSE

(a) The procedure utilized to verify that the as-received fuel and the storage canisters meet the vendor technical specifications will be part of the overall receipt inspection plan for incoming shipping casks and canisters and will be performed in addition to the security inspection and the survey for dose rates and surface contamination discussed in SAR Section 5.1.4.

A copy of the records required by 10 CFR 71.91 will be included with each shipment. A PFSF requirements document will be developed to provide the originating power plant with the detailed fuel and canister requirements. The originating nuclear power plant will be responsible for providing the information necessary to document that all fuel and cask/canister conditions are met. The information provided by the originating power plant will be developed in accordance with their approved quality assurance program. It will include signatures from appropriate plant management and QA personnel attesting to the completeness and accuracy of the information. Although this information will be provided by the originating nuclear power plant in accordance with their QA program, it will be the responsibility of the PFSF nuclear engineers to ensure that the loading of the canisters is properly conducted and in compliance with PFSF procedures and specifications (see response to RAI SAR 9-3).

Prior to shipment, the PFSF staff will review the information developed by the originating power plant to ensure that the cask/canister and contents are in compliance with the PFSF license. The PFSF staff will then authorize the shipment and provide the originating power plant with a unique shipment identification number for confirmation of the shipment release.

Following shipment and upon arrival at the PFSF, the receipt inspection procedure is initiated. The purpose of the PFSF receipt inspection procedure will be to provide a list of specific documentation attributes which must be reviewed and verified, and provide a list of items that require visual inspection/verification at the PFSF to ensure the as-received fuel and the storage canisters meet the vendor technical specifications. The procedure will be prepared, reviewed and approved, and training on the procedure provided as discussed in SAR Section 9.4.1. Records will be maintained per the requirements of SAR Section 9.4.2.

The receipt inspection procedure will contain a check list of specific attributes which must be reviewed and verified. Receipt inspection forms will be provided with appropriate signature blocks for operations personnel to sign after verification of each attribute. A list of typical attributes which will be included on the receipt inspection forms is provided below. The exact information required will be determined by the conditions imposed on the cask vendor Certificate of Compliance and the PFSF license/technical specifications.

- Verify the tamperproof device on the shipping cask is in place

- Verify the unique shipment identification number was assigned and agrees with PFSF records

- Verify that the radiation survey is complete

- Visually inspect shipping cask and canister to verify ID/serial number agrees with list previously supplied by vendor

- Review the fuel loading manifest to verify the fuel meets Technical Specification requirements. Required information is as follows:
    1. Name and address of shipper
    2. Date of shipment
    3. Fuel Assembly ID/Serial Number
    4. Type/condition
    5. Fuel cladding
    6. Initial enrichment
    7. Burnup
    8. Cooling time
    9. Assembly dimensions and weight
    10. Number of assemblies loaded
    11. Map showing location, by ID/serial number, of each assembly in the canister

- Verify the following reports are provided and that reported results meet applicable Technical Specification requirements (HI-STORM).
    1. Canister and shipping cask dryness verification

2. Canister helium backfill
3. Canister removable surface contamination
4. HI-STAR 100 shipping cask dose rates
5. Canister top end dose rates
6. Helium leak rate testing of Canister lid confinement welds
7. Helium leak rate testing of the HI-STAR 100 mechanical seals
8. HI-STAR 100 shipping cask helium backfill pressure
9. Field weld Liquid Penetrant examination
10. Pressure rise leak rate testing of Canister vent and drain port cover plate welds

- Verify the following reports are provided and that reported results meet applicable Technical Specification requirements (TranStor).
    1. Maximum permissible Canister leak rate
    2. Maximum Canister removable surface contamination
    3. Canister vacuum pressure during drying
    4. Canister helium backfill pressure
    5. Test of Canister shield and structural lid seal welds

- Verify that the documentation package includes a certificate of conformance signed by an authorized representative of the nuclear utility attesting to the accuracy and completeness of the information provided.

As stated in SAR Section 10.2.1.2, if canisters are received at the PFSF that do not conform with the above requirements, arrangements shall be made for return of the canisters to the originating nuclear power plant. Nonconforming canisters shall not be removed from the shipping cask.

(b) The following description of the receipt inspection procedure will replace the last sentence of the first paragraph of SAR Section 10.2.1.2, Specification:

"A PFSF receipt inspection procedure will be utilized to verify that the as-received fuel and the storage canisters meet the PFSF technical specifications. This procedure will be part of the overall receipt inspection plan for incoming shipping casks and canisters and will be performed in addition to the security inspection and the survey for dose rates and surface contamination.

The purpose of the PFSF receipt inspection procedure will be to provide a list of specific documentation attributes which must be reviewed and verified, and provide a list of items that require visual inspection/verification at the PFSF to ensure the as-received fuel and the storage canisters meet the PFSF technical specifications. The receipt inspection procedure will contain a check list of specific attributes which must be reviewed and verified. The exact information required will be determined by the conditions imposed on the cask vendor Certificate of Compliance and the PFSF license/technical specifications. Receipt inspection forms will be provided with appropriate signature blocks for operations personnel to sign after verification of each attribute."

The procedure will be prepared, reviewed and approved, and training on the procedure provided as discussed in SAR Section 9.4.1. Records will be maintained per the requirements of SAR Section 9.4.2.

CHAPTER 10—OPERATING CONTROLS AND LIMITS

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

10-4 Provide details or an appropriate reference for the minimum operating temperature limits of 0° and 32 °F established for handling the HI-TRAC transfer cask.

- Revise this section by incorporating details or a reference to justify that 0 °F is above the nil ductility temperature for the HI-TRAC transfer cask material, as it is made for the TranStor transfer cask in Subsection 10.2.1.4.

- There is no explanation of the thermal analysis to be performed to operate below 32 °F (concern about water freezing) in the HI-TRAC transfer cask.

RESPONSE

- The minimum operating temperature limits of 0° and 32°F established for handling the HI-TRAC transfer cask are specified in Chapter 12, Subsection 12.3.12 of the HI-STORM SAR.

  The minimum operating temperature limits are postulated as a constant ambient temperature in the Canister Transfer Building caused by extreme weather conditions. To determine the effects of these temperatures, it is conservatively assumed that these temperatures persist for a sufficient duration to allow the HI-TRAC transfer cask to achieve thermal equilibrium. Because of the large mass of the HI-TRAC transfer cask with its corresponding large thermal inertia and the limited duration for the temperatures and duration of use for the HI-TRAC transfer cask, this assumption is conservative. The HI-TRAC transfer cask is composed of ferritic steel materials subject to impact loading in a cold environment and, therefore, must be evaluated and/or tested for their propensity for brittle fracture.

  The HI-STORM SAR (Subsection 3.1.2.3) discusses nil ductility transition temperature, Charpy V-notch testing, and the structural analysis performed to assure prevention of brittle fracture failure of the HI-TRAC transfer cask. The HI-TRAC transfer cask top flange, lifting trunnion block, pocket trunnion, and 125-ton HI-TRAC pool lid outer ring are composed of SA350-LF3 (the 100-ton HI-TRAC pool lid is composed of SA203-E) and all have thicknesses greater than 4 inches. SA350-LF3 and SA203-E were specifically chosen for their ductility at low temperatures. All other steel structures in the HI-TRAC transfer cask are made of

SA516-70. Table 3.1.18 of the HI-STORM SAR, provides a summary of impact testing requirements. These requirements ensure that HI-TRAC transfer cask meets all brittle fracture requirements for an ambient temperature of 0°F.

The text of Section 10.2.1.5 will be revised as shown below to incorporate reference to the HI-STORM SAR.

Basis: The HI-TRAC thermal analysis is based on an upper ambient temperature of 100°F. Operating the HI-TRAC at or below 32°F may lead to freezing and subsequent damage to the neutron shield jacket. Handling the HI-TRAC below an ambient temperature of 0°F may present a risk of brittle fracture as discussed in the HI-STORM SAR (Reference 1), Chapter 12, Section 12.3.12.

- The minimum operating temperature specified for the HI-TRAC transfer cask is 0°F and the HI-TRAC is conservatively assumed to reach 0°F throughout the structure. For ambient temperatures from 0° to 32°F, a 25% ethylene glycol solution may be added to the demineralized water in the water jacket to prevent freezing. An alternate method to ensure that the water in the neutron shield tank does not freeze is perform a thermal analysis to specify the minimum decay heat required. The thermal model provided in Chapter 4, Section 4.5 of the HI-STORM SAR, would be utilized to determine the minimum decay heat required to ensure that the temperature of the HI-TRAC transfer cask does not drop below 32°F with an ambient temperature of 0°F.

  It is not expected that the HI-TRAC transfer cask will be exposed to ambient temperatures below 32°F at the PFSF since it is used exclusively inside the Canister Transfer Building. Rather than perform a thermal analysis as discussed above, PFSF will add a 25% ethylene glycol solution to the demineralized water in the water jacket to prevent freezing in the unlikely event that the HI-TRAC may be exposed to ambient temperatures below 32°F. This will protect the HI-TRAC from freezing temperatures whether or not it is loaded. SAR Section 10.2.1.5 will be revised as follows:

  Action: a. If the HI-TRAC transfer cask is exposed to an ambient temperature below 32° F, the neutron shield water jacket shall be drained and replaced with a 25 percent solution of ethylene glycol and demineralized water.

CHAPTER 10—OPERATING CONTROLS AND LIMITS

Section 10.2.2.2 Concrete Storage Cask External Dose Rate

10-5 (a) Provide justification or additional references for the different values adopted for allowable external radiation dose rates at various locations for HI-STORM and TranStor storage casks for Zircaloy and stainless steel clad fuels.

(b) Provide an explanation for selecting different dose values for Zircaloy and stainless steel clad fuels.

- The specification should indicate acceptance criteria for the external dose rate for both types of casks at comparable locations, and a reference should be included to justify the specified values.

RESPONSE

The dose rates specified by each vendor are intended to provide an additional means of verifying that the cask has been loaded with fuel meeting the technical specification requirements and that the system is functioning properly. The dose rate limits were selected by the vendors to ensure that personnel exposure is maintained ALARA and that off-site dose rates meet 10CFR72 requirements. The limiting conditions for operation provided in SAR Section 10.2.2.2 have been formatted to be consistent with the vendors SARs and with the expected Certificates of Compliance issued by the NRC for each vendors storage system.

Specifically, Holtec and SNC say the following in their SARs for criteria for cask dose rates:

> HI-STORM SAR, Section 2.3.5.2, "Dose rates in the immediate vicinity of the cask are important in consideration of occupational exposure. A design objective for the maximum radial surface dose rate has been established as 35 mrem/hr. Areas adjacent to the inlet and exit vents which pass through the radial shield are limited to 50 mrem/hr. The average dose rate at the top of the overpack is limited to below 10 mrem/hr."

> TranStor SAR, Section 2.3.5.2, "The TranStor storage cask and other components are designed to minimize radiological dose rates to the general public and plant personnel. The design dose limits one meter from the cask surface are selected as 15 mrem/hr for the side (30 mrem/hr for SS-clad fuel) and 200 mrem/hr for the cover lid centerline. ....The calculated one meter dose rates (approximately 10 mrem/hr at the side and 135 mrem/hr at the top)

are well within their design limits and the actual measured data shows even lower values."

It should be noted that SAR Chapter 7, Table 7.3-1 and Table 7.3-2, provide maximum dose rates (calculated in the vendors shielding analysis) on contact and at one meter from the side and top of each vendors cask, as well as top and bottom vent contact doses. The calculated values are seen to be within the limits specified in SAR Section10.2.2.2.

(a) Maximum external surface dose rates for the TranStor system are provided in the TranStor SAR, Chapter 12, Section 12.2.1.3. The HI-STORM system dose rates are provided in the HI-STORM SAR, Chapter 12, Section 12.3.16. These references are currently provided in the "Basis" portion of Section 10.2.2.2.

(b) SAR Chapter 7, Section 7.2.1.1, provides a discussion of the methodology used by both vendors to determine the fuel region gamma source originating from fission products, actinides, and activated materials in the active fuel region. The TranStor shielding analysis determined that the gamma source from Co-60, as a result of activated stainless steel fuel cladding, would result in higher dose rates than those for zircaloy clad fuel. Additional detail on the basis for these limits is provided in the shielding analysis presented in Chapter 5 of the HI-STORM and TranStor SARs.

CHAPTER 10—OPERATING CONTROLS AND LIMITS

Section 10.2.2.3 Concrete Storage Cask Air Outlet Temperature-Initial Installation

10-6 Provide a reference or data to support the choice of the limiting temperature values for TranStor and HI-STORM storage casks.

- Revise the text by providing supporting documentation for the specified temperature limits to avoid degradation of fuel, canister, and concrete materials for TranStor and HI-STORM storage casks.

RESPONSE

The text of SAR Section 10.2.2.3 will be revised as shown below to add the appropriate references.

Specification: The equilibrium air temperature, after initial installation, at the outlet of a loaded storage cask shall not exceed ambient by more than 125° F for the HI-STORM (Reference 1, Chapter 12, Section 12.3.17) storage cask and 100° F for the TranStor (Reference 2, Chapter 12, Section 12.2.1.2) storage cask.

CHAPTER 10—OPERATING CONTROLS AND LIMITS

Section 10.2.2.3 Concrete Storage Cask Air Outlet Temperature-Initial Installation

10-7 Provide maintenance and calibration requirements for temperature monitoring instruments to ensure reliable operation.

- As specified in 10 CFR 72.164, the licensee will establish measures to ensure that instruments and other testing devices are properly calibrated at specified periods to maintain accuracy within necessary limits. Revise the section by providing maintenance and calibration intervals.

RESPONSE

The temperature monitoring instruments are passive electronic components that will not require any maintenance. The PFSF is committed to the calibration of temperature monitoring instrumentation, which will be defined by the monitor manufacturer. Each temperature instrument will be tested when the storage casks are placed in service. The temperature monitoring system will include automatic calibration and self-testing features. Additional calibration requirements, as recommended by the manufacturer, will be incorporated into facility procedures.

SAR Section 9.4.1.1.3, Maintenance and Surveillance Procedures, discusses procedures that will be used to ensure that various activities such as calibrations will be performed to preclude the degradation of PFSF equipment.

ENCLOSURES