April 1, 1998

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

SUBJECT: REQUEST FOR ADDITIONAL INFORMATION (TAC NO. L22462)

Dear Mr. Parkyn:

The staff of the Nuclear Regulatory Commission (NRC) has completed its request for additional information (RAI) regarding Private Fuel Storage, L.L.C.'s (PFS's) application for a license to construct and operate an independent spent fuel storage installation (ISFSI) on the reservation of the Skull Valley Band of Goshute Indians (see enclosure). Please note that this RAI only includes information associated with the Holtec Hi-Storm and Sierra Nuclear TranStor cask systems to the extent that this information is directly necessary for the staff's review of the subjects covered in the safety analysis report which are not cask-specific. The Hi-Storm and TranStor cask systems are being reviewed separately and are not the focus of this RAI. While the staff has attempted to be as thorough as possible, should additional safety, technical, regulatory, or financial issues come to light, this RAI may be supplemented.

In the course of its review, the staff noted that PFS has committed to design according to certain recognized national standards. The staff requests that PFS committ to fabricate as well as design to these standards which are identified in Safety Analysis Report Chapter 3, Principal Design Criteria.

The staff notes that the PFS 10 CFR Part 71 Quality Assurance Program (QAP), Revision 1, which the staff approved on September 17, 1996, has been incorporated by reference into this PFS application. Successful implementation of the QAP and associated procedures for the activities proposed in this license application would be demonstrated to the staff during inspections of licensed activities. The staff does not review and approve quality assurance procedures as part of the licensing process.

Please reference the Docket No. and the above TAC No. in future correspondence related to this request. Please inform us, within 30 days of the date of this letter, of your schedule for responding to this RAI.

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

Sincerely,

ORIGINAL SIGNED BY /s/

Mark S. Delligatti, Senior Project Manager
Spent Fuel Licensing Section
Spent Fuel Project Office
Office of Nuclear Material Safety
and Safeguards

Docket 72-22

Enclosure: As stated

cc: PFS Service Lists

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REQUEST FOR ADDITIONAL INFORMATION

This document, titled Request for Additional Information (RAI), contains a compilation of additional information requirements, identified to-date by the U.S. Nuclear Regulatory Commission (NRC) staff, during its review of the Private Fuel Storage, L.L.C. (PFS), License Application (LA) and Safety Analysis Report (SAR). This RAI follows the same format as the applicant's SAR.

Each individual RAI describes information needed by the staff for it to complete its review of the LA and/or the SAR and to determine whether the applicant has demonstrated compliance with the regulatory requirements. Where an individual RAI relates to the applicant's apparent failure to meet one or more regulatory requirements or where an RAI specifically focuses on compliance issues associated with one or more specific regulatory requirements (e.g., specific design criteria or accident conditions), such requirements will be specified in the individual RAI.

This RAI is organized as follows:

LICENSE APPLICATION

Financial Assurance and Decommissioning Funding Assurance

The following regulatory requirements are applicable to the RAIs in this section: 10 CFR 72.11, 72.22, 72.30, 72.54, 72.130, 72.236(l), and 10 CFR 61.55. It should be noted that other regulatory requirements may be applicable to this section.

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.

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.

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.

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).

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.

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.

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.

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.

Physical Protection

LA Chapter 9

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.

SAFETY ANALYSIS REPORT

CHAPTER 1—INTRODUCTION AND GENERAL DESCRIPTION OF INSTALLATION

The following regulatory requirements are applicable to this chapter: 10 CFR 72.2(a)(1); 72.11; 72.22; 72.24(a), (b), (c)(3), (j) and (n); 72.28(a); 72.40(a)(3) and (5); and 72.236(a) (Nuclear Regulatory Commission, 1997). It should be noted that other regulatory requirements may be applicable to this chapter.

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.

CHAPTER 2—SITE CHARACTERISTICS

The following regulatory requirements are applicable to this chapter: 10 CFR 72.24(a); 72.90; 72.92; 72.94; 72.96(a); 72.98; 72.100; 72.102; 72.104; 72.106; 72.108; 72.122(b); and Appendix A to 10 CFR Part 100 (Nuclear Regulatory Commission, 1997). It should be noted that other regulatory requirements may be applicable to this chapter.

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.

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.

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.

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.

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.

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.

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.

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.

CHAPTER 3—PRINCIPAL DESIGN CRITERIA

The following regulatory requirements are applicable to this chapter: 10 CFR 72.24(c); 72.40; 72.82(a); 72.106(a), (b) and (c); 72.120(a) and (b); 72.122(a), (b), (c), (d), (e), (f), (g), (h), (j) and (k); 72.124(a), (b) and (c); 72.126(a); 72.128(a) and (b); 72.130; 72.182(a); and 72.236 (e), (f), (g) and (k) (Nuclear Regulatory Commission, 1997). It should be noted that other regulatory requirements may be applicable to this chapter.

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.

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.

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.

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)

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.

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.

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.

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.

CHAPTER 4—INSTALLATION DESIGN

The following regulatory requirements are applicable to this chapter: 10 CFR 72.11; 72.24(b), (c), (d), (l) and (l)(2); 72.26; 72.40; 72.44(c); 72.70; 72.82(c); 72.106; 72.120(a); 72.122 (a), (b), (c), (d), (f), (g), (h), (k) and (l); 72.146; 72.154; 72.162; and 72.236 (Nuclear Regulatory Commission, 1997). It should be noted that other regulatory requirements may be applicable to this chapter.

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).

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.

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."

CHAPTER 5—OPERATION SYSTEMS

The following regulatory requirements are applicable to this chapter: 10 CFR 72.24(b), (d)(l)(2) and (f); 72.40(a)(1), (a)(5) and (13); 72.44(c)(1); 72.104(b); 72.122(f), (g), (h), (l), (j), (k) and (l); and 72.236(j) (Nuclear Regulatory Commission, 1997). It should be noted that other regulatory requirements may be applicable to this chapter.

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.

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.

CHAPTER 6—SITE-GENERATED WASTE CONFINEMENT AND MANAGEMENT

The following regulatory requirements are applicable to RAIs in this chapter: 10 CFR 72.104; 72.122; 72.126; and 72.128 (Nuclear Regulatory Commission, 1997). It should be noted that other regulatory requirements may be applicable to this chapter.

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)].

CHAPTER 7—RADIATION PROTECTION

The following regulatory requirements are applicable to the RAIs in this chapter:
10 CFR 72.24(e), (l)(1), (2), and (m); 72.40(a), (5) and (13); 72.92(c); 72.94(c); 72.104; 72.106(a) and (b); 72.122(h)(3) and (5); 72.128; 72.130; 10 CFR 20.1101; 20.1201; 20.1207; 20.1208; 20.1301; 20.1302; 20.1501; 20.1502; 20.1601(a), (b), (c), (d) and (e); 20.1602; 20.1701; 20.1702; 20.1801; 20.1802; and 20.2106 (Nuclear Regulatory Commission, 1997). It should be noted that other regulatory requirements may be applicable to this chapter.

Section 7.2.1 Characterization of Sources and

Section 7.3.3.5 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.

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.

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.

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.

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.

CHAPTER 8—ACCIDENT ANALYSES

The following regulatory requirements are applicable to this chapter: 10 CFR 72.11; 72.24(a), (d), (e), (k) and (m); 72.26; 72.32; 72.40(a)(1) and (13); 72.44(c); 72.92; 72.94; 72.102(c), (d) and (f); 72.104; 72.106(a) and (b); 72.120(a); 72.122(b), (d), (g), (h), (l), (j) and (l); 72.124; 72.126(d); 72.128; and 72.236 (Nuclear Regulatory Commission, 1997). It should be noted that other regulatory requirements may be applicable to this chapter.

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.

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.

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.

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.

Section 8.1.5 Off-Normal Contamination Release

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

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)

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.

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.

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))

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.

CHAPTER 9—CONDUCT OF OPERATIONS

The following regulatory requirements are applicable to this chapter: 10 CFR 72.24(e), (h), (I), (j), (k) and (p); 72.28(a), (b), (c) and (d); 72.30(d)(1); 72.32(a); 72.40(a)(4), (9), (11) and (13)(I); 72.44(b)(4) and (5); 72.144(d); 72.190; 72.192; 72.194; and 73.21(a), (b)(I), (iii), (v), (viii), (x), and (xii) (Nuclear Regulatory Commission, 1997). It should be noted that other regulatory requirements may be applicable to this chapter.

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.

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.

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.

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.

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.

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.

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.

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.

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.

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.

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

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.

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

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.

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.

CHAPTER 10—OPERATING CONTROLS AND LIMITS

The following regulatory requirements are applicable to this chapter: 10 CFR 72.11; 72.24 (g); 72.26; 72.44(c); 72.44(d); 72.104; 72.106; 72.164; 72.172; 72.234(a); and 72.236 (Nuclear Regulatory Commission, 1997). It should be noted that other regulatory requirements may be applicable to this chapter.

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.

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.

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.

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.

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.

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.

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.

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