Private Fuel Storage, LLC

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John D. Parkyn, Chairman of the Board

September 21, 1999

U.S. Nuclear Regulatory Commission
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Washington, D.C. 20555-0001

LICENSE APPLICATION AMENDMENT No. 7
DOCKET NO. 72-22/TAC NO. L22462
PRIVATE FUEL STORAGE FACILITY
PRIVATE FUEL STORAGE L.L.C.

The purpose of this letter is to submit Amendment 7 to the Private Fuel Storage Facility (PFSF) License Application (LA). The Safety Analysis Report and LA are being revised to reflect Private Fuel Storage L.L.C. information provided in responses to NRC requests for additional information (RAIs) and commitment resolution letters that has not been previously incorporated into these licensing documents.

If you have any questions regarding this submittal, please contact me at 608-787-1236 or Mr. J. L. Donnell, Project Director, at 303-741-7009.

Sincerely,

John D. Parkyn, Chairman
Private Fuel Storage L.L.C.

JDP:JRJ
Enclosures

PREFACE

PRIVATE FUEL STORAGE FACILITY

LICENSE APPLICATION

AMENDMENT 7

Enclosed are the following revisions to the Private Fuel Storage Facility License Application documents:

Safety Analysis Report - Revision 7

License Application - Revision 4

The revisions are provided in the replacement page format. If a page has changed, either due to an actual text change or due to a shift in text caused by insertion of text, then the revision number of that page has been changed in the upper right hand corner. The location of text changes is noted by a side bar in the right hand margin.

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The PFSF project has been developed on a phased basis. Steps I and II, which involved preliminary investigations, predated the formation of the PFSLLC. Step III began with the formation of the PFSLLC and concluded with the filing of the License Application. This step was funded by direct payments to the PFSLLC from member utilities pursuant to Subscription Agreements. Step IV includes the NRC licensing proceeding as well as detailed design and preparation of bid specifications. The budget for Step IV is approximately $10 million, including contingencies, to be funded by direct payments to the PFSLLC from the member utilities pursuant to Subscription Agreements. These Step IV payments will be made on a quarterly basis. Given the relatively small size of this payment for any participating utility, there is the reasonable assurance that the PFSLLC will obtain Step IV funding.

Step V represents the construction of the PFSF. The budget for this phase is $100 million and includes site preparation; construction of the access road, administration building, security and health physics building, operations and maintenance building, canister transfer building and storage pads; procurement of canister transfer and transport equipment; and transportation corridor construction. The Step V budget also includes necessary personnel costs, licensing fees, and host benefits, as well as a contingency amount.

Step V will be funded through several mechanisms. An additional $6 million in equity contributions is planned from PFSLLC members pursuant to Subscription Agreements. The bulk of the Step V costs is expected to be funded through Service Agreements with PFSF customers (including both PFSLLC members and non-members). Payments under each Service Agreement will be spread out over the period of time from construction through spent fuel delivery. No construction will proceed unless Service Agreements committing for a significant quantity of spent fuel storage have been signed. The nominal target is 15,000 MTU of storage commitments. Raising the non-

equity portion of Step V costs through Service Agreements will allow the PFSLLC to avoid financing costs for construction. The PFSLLC, however, retains the option to finance the non-equity portion of Step V costs through debt financing secured by Service Agreements. As with direct financing from customers, no construction will take place without the commitment through Service Agreements for a significant quantity of spent fuel storage. Unless PFSLLC members and non-members have committed to a significant quantity of storage, construction of the PFSF will not begin. Thus, there will be reasonable assurance that the PFSLLC will obtain Step V funding.

Step VI, the operational phase of the PFSF, will also be funded through the Service Agreements. The significant costs of this phase will include procurement and/or fabrication of canisters ($432 million) and storage casks ($134 million). These components will be obtained on an as-needed basis, to coincide with the schedule for moving spent fuel to the PFSF. All capital costs associated with the storage of any spent fuel will be paid by the customer pursuant to the Service Agreement prior to the acceptance by the PFSLLC of that spent fuel. Since the PFSF will not accept spent fuel for storage without prior payment through Service Agreements of the necessary capital costs for transportation and storage, there is reasonable assurance that the PFSLLC will obtain the necessary Step VI costs.

The on-going operations and maintenance cost for spent fuel in storage at the PFSF will be paid by the customer on an annual basis as required by the Service Agreements. The annual operations and maintenance cost is estimated to be $49 million for a 20-year facility operating life and $31 million for a 40-year life. 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. All dollars expressed are in current year dollars at the time of the license application submittal (1997).

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.

The Service Agreements will provide assurance for the continued payment of these costs by requiring the customers to provide annual financial information, meet creditworthiness requirements, and, if necessary, provide additional financial assurances (such as an

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CHAPTER 4

DECOMMISSIONING COST ESTIMATE

The decommissioning cost estimate is based on a 40,000 MTU facility. The size of the storage facility affects only approximately 6 percent of the overall decommissioning cost. The total decommissioning cost is highly contingent upon the shipping casks, the Canister Transfer Building, and the transfer casks, none of which are dependent on the size of the storage facility. The cost to decommission each storage cask is funded separately before an individual cask is utilized, as described in Section 5.1. The only variance in the decommissioning cost related to the size of the storage facility is the area of the concrete storage pads and the assumed amount of decontamination and disposal costs associated with that area.

Decommissioning the PFSF will be a multiphased effort, with portions completed during the operational phase. The amount of decontamination required and the extent of decommissioning efforts will be based on the usage and history of the facility. The cost of decommissioning major portions and components of the facility is outlined here as a means to estimate the total cost of decommissioning the facility.

The philosophy of operating the PFSF is "start clean, stay clean." Thus the intention is to maintain the facility free of radiological contamination at all times. During the operational phase of the facility, all radioactive contamination will be removed immediately upon its discovery. The cost estimate for decommissioning nonetheless conservatively assumes that certain areas and components will require decontamination. The areas of possible contamination concern and the projected decontamination and decommissioning costs are discussed below.

Shipping Casks: The shipping casks will not become activated because of the relatively short duration of their exposure to the spent fuel canisters. In the event a shipping cask becomes contaminated, the cost of decontamination is estimated to be $5,000. Four shipping casks from each of the two vendors, for a total of eight casks, will require $40,000 for decommissioning. The basis for this estimated cost is provided below.

Surveys to determine removable and fixed contamination levels on each shipping cask will cost $200 based on two technicians at $25/hr each for four hours.

Decontamination of each Shipping Cask is estimated to cost $1,000, based on general decontamination cost of $1 per square foot (s.f.) for approximately 400 s.f. of interior surface area and 600 s.f. of external surface area. The cost for general decontamination efforts of $1 per square foot is based on actual experience at a nuclear power plant undergoing decontamination in 1997 (La Crosse), including labor and materials.

Waste disposal is estimated to cost $1,100, based on 3 cubic feet (c.f.) of compacted low level waste at $300/c.f. for disposal plus $200 for transportation.

Ten percent of the shipping cask internal surface area is assumed to have fixed contamination, or 40 s.f. In removing all the fixed contamination, it is assumed that one-inch of material removed, which will generate approximately 3.5 c.f. of waste. $150/c.f. is estimated for cutting and removal of contaminated portions, based on two workers plus a health physics technician at $25/hr each for 6 hours, plus $75 in materials, for a total of $525. $100/c.f. is estimated for packaging based on two workers plus a health physics technician at $25/hr each for 4 hours, plus $50 in materials, for a total of $350. Disposal of the 3.5 c.f. of low level waste is estimated to cost $1,250, based on $300/c.f. for disposal plus $200 for transportation.

The cost to decontaminate each shipping cask is therefore estimated to be $4,425, which is rounded to $5,000, resulting in a total for 8 shipping casks of $40,000.

Storage Casks: The storage casks vendors have indicated there will be no anticipated activation of cask materials. Measures will be taken at the originating reactors and upon arrival of the canisters at the PFSF to ensure the canisters will have surface contamination levels below specified limits before being loaded into storage casks, thereby minimizing the possibility of contaminating the storage casks. It is therefore anticipated that the storage casks will have no radioactive contamination or activation. In order to conservatively account for the unlikely event that a storage cask is found to have contamination or activation levels above the applicable NRC limits for unrestricted release, an estimate has been made of costs to decontaminate and dispose of a storage cask.

The inside surface of a storage cask is 365 square feet and the initial decontamination is estimated to cost $365 plus waste disposal costs of $550. Waste disposal cost of $550 associated with decontamination of each Storage Cask is based on an estimate of 1.5 cubic feet of compacted low level waste at $300/c.f. for disposal plus $100 for transportation. If surveys show the cask has fixed contamination or activation, a series of three core borings at an estimated total cost of $850 will be performed to determine the nature and extent of activation or fixed contamination, i.e., whether it is the steel liner, concrete shielding or both. Core boring costs are estimated at $850 based on two workers plus a health physics technician at $25/hr each for 8 hours plus $250 for miscellaneous tools and supplies.

If the steel liner is activated, it will be removed and sectioned for shipment off site to a licensed disposal facility. It will cost an estimated $3,000 for dismantlement and

packaging efforts. This cost of $3,000 is based on the assumption that 20% of the internal surface area (365 sq. ft.), or 73 sq.ft., will have activation or fixed contamination. The steel liner is 2 inches thick, based on manufacturer's specifications and drawings. The volume of steel liner to be dismantled and packaged is therefore 12 cubic feet (c.f.). Estimate $150/c.f. for dismantlement, based on two workers plus a health physics technician at $25/hr each for 20 hours, plus $300 for miscellaneous tools and supplies, for a total of $1,800. Estimate $100/c.f. for packaging, based on two workers plus a health physics technician at $25/hr each for 10 hours, plus $450 in materials, for a total of $1,200.

Shipping cost for Storage Cask low level waste is estimated to be $1,400 based on 12 c.f. at $100/c.f. plus $200 for miscellaneous expenses. Disposal cost for the low-level waste for each Storage Cask is based on 12 c.f. of material at $300/c.f., for a total of $3,600.

If the storage cask concrete is activated, it will be scabbled at an estimated cost of $1,970. This cost is based on two workers plus a health physics technician at $25/hr each for twenty hours plus $200 in tools and materials. Disposal cost of $270 is estimated for scabbled concrete from each Storage Cask based on 1/8 inch of material removed from 73 s.f., or 0.9 c.f. of material at $300/c.f. The total cost to decommission a storage cask is estimated to be less than $17,000.

<u>Site Characterization Survey</u>: At the end of facility operations, a radiological survey of the entire PFSF site will be performed in order to verify the absence of contamination and to identify any areas requiring decontamination. The cost of this survey is estimated to be $250,000, which is based on 2,500 data points at $100 per sample.

Canisters: The spent fuel canisters will be shipped off-site prior to the commencement of facility decommissioning. These activities are considered part of PFSF operations, and the associated costs are therefore not included in the decommissioning cost estimate.

Transfer casks: There will be four transfer casks; two for each vendor design, one of which will be used at the PFSF and the other which will be used at the various reactor sites. The transfer casks will not become activated due to their relatively short duration of exposure to the spent fuel canisters, but they may become contaminated. Using the same assumptions as for the shipping cask, the final decontamination and dismantlement of the transfer casks is estimated to cost $5,000 per cask in labor and material disposal costs, for a total of $20,000.

Canister Transfer Building: For the purpose of preparing a decommissioning cost estimate, the Canister Transfer Building operations area of 46,000 square feet is assumed to require decontamination. The cost of decontamination is estimated to be $5 per square foot for labor, materials and waste disposal. This cost is based on $1/s.f. for general decontamination efforts plus an additional $4/s.f. to perform a more intense cleaning of those areas with potentially higher contamination levels. These areas will require a reduction in cleaning rate per time unit and a corresponding increase in the unit cost. The total estimated cost to decommission the Canister Transfer Building is $230,000.

Storage pads: The concrete storage pads will only be used for sealed storage casks and it is not anticipated that they will become activated or contaminated. The only mechanism which could result in contamination of a storage pad is by having a contaminated canister which was not detected prior to insertion in a storage cask. The possibility of such an occurrence is remote, but is addressed for decommissioning

purposes by assuming up to 10 percent of the storage pad area will require surface decontamination. The maximum number of storage pads is 500, with each having an area of 64 ft by 30 ft, for a total area of 960,000 square feet. Ten percent of this area is 96,000 square feet, which takes no credit for the area protected by the bottom of each storage cask. A storage pad decontamination cost of $1/s.f. is utilized based on actual experience at the La Crosse nuclear power plant undergoing decontamination in 1997, including labor and materials. Therefore decontamination of this area is estimated to cost $96,000. Storage pad waste disposal cost from decontamination efforts is estimated to be $145,000 based on $100/c.f. for packaging, plus $100/c.f. for transportation, plus $300/c.f. for disposal of an assumed 290 c.f. of low level waste. The total estimated cost to decontaminate the storage pads is $241,000.

Final Site Survey: A final site survey will be performed to verify decontamination and decommissioning efforts and the absence of radioactive materials. A final site survey is estimated to cost $260,000 based on essentially re-performing the characterization survey, with an additional $10,000 contingency.

Independent Verification Survey: This survey, to be performed by a contractor selected by the NRC, is a validation of the results of the final site survey. An independent verification survey is estimated to cost $50,000 based on sampling 20 % of the areas covered in the final survey.

The total estimated cost of PFSF decommissioning is estimated to be $1,631,000 plus $17,000 per cask for each storage cask actually utilized.

CHAPTER 5

DECOMMISSIONING FUNDING PLAN

The method of funding for decommissioning activities consists of two components: prepayment of the costs for decommissioning the storage casks into an escrow account and a letter of credit coupled with an external sinking fund for the costs of decommissioning the remainder of the facility and site. These financial assurance mechanisms will be prepared in conformance with the guidance of NRC Regulatory Guide 3.66.

5.1 Storage Cask Decommissioning Funding Plan

The service agreement with each customer (reactor) shall require at least $17,000 to be deposited into an externalized escrow account prior to shipment of each spent fuel canister to the PFSF. The full amount of potential decommissioning costs will thus be collected in a segregated account prior to the receipt of each spent fuel canister at the PFSF. This method of funding provides for prepayment of the storage cask decommissioning costs prior to any potential exposure of the storage cask to radiation or radioactive material, and therefore prior to the need for any decommissioning. This funding method complies with the requirements of 10 CFR 72.30(c)(1).

Storage cask decontamination and decommissioning may be performed at any time following the removal of the canister and its shipment off site. This will allow individual storage cask decommissioning to be an ongoing effort, which can potentially be completed by the end of canister shipping operations. As storage cask

decommissioning is completed, the amount of funds in the escrow account will be adjusted periodically to reflect the remaining decommissioning efforts. The escrow amount and the per-canister fee will be reviewed and adjusted annually to account for inflation and any changes in the estimated cost of storage cask decommissioning.

5.2 Facility and Site Decommissioning Funding Plan

A letter of credit will be obtained in the amount of $1,631,000 to cover the estimated facility and site decommissioning costs, exclusive of the storage casks. This amount includes $250,000 for a site characterization survey, $200,000 for decommissioning of four transfer casks, $400,000 for decommissioning of eight shipping casks, $230,000 for decontamination of the Canister Transfer Building, $241,000 for storage pad decontamination, $260,000 for a final release survey, and $50,000 for an independent verification survey. This letter of credit will be coupled with an external sinking fund into which customers will be required under the service agreements to pay the costs to decontaminate any portion of the facility for which they may be responsible for contaminating. When the actual costs of decontamination and decommissioning are paid into the external sinking fund, the letter of credit may be reduced by an equivalent amount.

The amounts in the external sinking fund and the letter of credit will be reviewed and adjusted annually to account for inflation and any changes in the scope or cost of decommissioning. 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. This funding method complies with the requirements of 10 CFR 72.30(c)(3).

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CHAPTER 2

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2.2 NEARBY INDUSTRIAL, TRANSPORTATION, AND MILITARY FACILITIES

The PFSF site is situated in the northwest corner of the Skull Valley Indian Reservation in Tooele County, Utah. The Reservation consists of approximately 18,000 acres, of which the PFSF site area is approximately 820 acres, or less than 5% of the reservation area. The PFSF site location was selected by the Skull Valley Band of Goshute Indians in order to avoid disruption of tribal roads, housing or cultural facilities. Figure 1.1-1 shows the facilities and locations addressed in this section.

The area surrounding the PFSF site is very sparsely populated, with the nearest residence 2 miles southeast of the site. The Skull Valley Band of the Goshute Village, with a population of about 30, is 3.5 miles east-southeast of the PFSF site. Terra, a small residential community with a population of 120 (Tooele County Commission, 1995), is located 10 miles east-southeast of the PFSF.

2.2.1 Hazards from Facilities and Ground Transportation

The only industrial, transportation or military facility within 5 miles of the PFSF is the Tekoi Rocket Engine Test facility, located about 2.5 miles south-southeast of the PFSF. This facility is used periodically to test engines mounted on stationary bases. Hickman Knolls, with an elevation of approximately 4873 ft, is situated directly between the PFSF (approximate elevation 4465 ft) and the Tekoi Test facility (elevation 4600 ft). The relative location of Hickman Knolls between the PFSF and Tekoi Test facility, and the distance of 2.5 miles would substantially deflect and disperse overpressures from an explosion at the Tekoi Test facility, precluding any hazard to the PFSF. There are no other facilities which could present the threat of an explosion or other hazard within 5 miles of the PFSF.

Interstate Highway 80 and the Union Pacific Railroad main line are located 24 miles north of the PFSF site. Any events associated with either the interstate highway or the railroad will not present a hazard to the PFSF due to the relatively large distance involved. The Skull Valley Road runs essentially north-south between Interstate 80 and the town of Dugway, population 1,700, 12 miles south of the PFSF. Dugway is a residential community supporting the nearby Dugway Proving Ground and has no facilities which could present a hazard to the PFSF.

The U.S. Army's Dugway Proving Ground is a 1,315 square mile range and test facility located west of the town of Dugway. The Dugway Proving Ground performs testing of all types of military equipment in chemical and biological environments, as well as smoke, obscurant and incendiary testing, and munitions testing. Open air testing is not permitted by law, and there have been no accidents or releases of toxic gas from the facility or associated transportation activities. The Proving Ground has a mean elevation of 4,350 ft above sea level and is surrounded on three sides by mountain ranges. The Cedar Mountains, with an elevation of 5,300 ft or greater, lie between the Proving Ground and the PFSF. The activities and materials at Dugway Proving Ground will therefore present no credible hazard to the PFSF, because of their relative distance and the intervening Cedar Mountains.

The Dugway Proving Ground receives and ships conventional Army weapons approximately 95 times a year. Some of these shipments could travel the Skull Valley Road, which present the only credible potential for an explosion near the PFSF. An accident associated with the transportation of explosives along the Skull Valley Road would be a minimum of 1.9 miles from the canister transfer building and 2 miles from the nearest cask storage pad. Based on the methodology of Regulatory Guide 1.91, the Skull Valley Road is located much further from the PFSF than the distances

required to exceed 1 psi overpressure for detonation of explosives transported by highway.

The Tooele Army Depot facilities, where toxic gas munitions are stored and incinerated, are located west and south, respectively, of Tooele City. The North Tooele Army Depot is 17 miles east-northeast of the PFSF and the South Tooele Army Depot is 21 miles east-southeast of the PFSF. The Stansbury Mountains, with an elevation of approximately 8,000 feet, lie between the PFSF and the Tooele Army Depots. The activities and materials at the Tooele Army Depots will therefore present no credible hazard to the PFSF, because of their relative distance and the intervening Stansbury Mountains.

2.2.2 Hazards from Air Crashes

Aircraft flights in the vicinity of the PFSF take place to and from Michael Army Airfield on Dugway Proving Ground, on and around the Utah Test and Training Range (UTTR), and on federal airways J-56 and V-257. While there are no civilian airports within 25 miles of the PFSF, general aviation aircraft may also transit the region. The average annual probability of an aircraft crashing into the PFSF has been calculated to be less than 1 E-6 per year and qualitative factors indicate that the true probability of an aircraft impacting the PFSF is less than 1 E-7 per year. (PFS Aug. 1999) This is an extremely low probability, below the guideline of NUREG-0800 (1 E-7) , and well below the regulatory standard the NRC has promulgated for above ground facilities at the Yucca Mountain geologic repository (1 E-6) (64 Fed. Reg. 8,640, 8,652 (1999)). Therefore, aircraft crashes do not present a credible hazard to the PFSF and the facility does not need to be designed to withstand the impact of an aircraft crash.

2.2.2.1 Michael Army Airfield and Airway IR-420

Michael Army Air Field is located on the Dugway Proving Ground, 17 miles south-southwest of the PFSF. This military airfield has a 13,125 foot runway, and can accommodate all operative aircraft in the Department of Defense inventory, although the majority of the aircraft flying to and from Michael AAF are large cargo aircraft such as the C-5, C-17, and C-141. The airspace over the Dugway Proving Ground is restricted. Military airway IR-420 passes over the PFSF site area. The methods of NUREG-0800 Section 3.5.1.6 were used to estimate the probability of an aircraft impacting the PFSF from this airway, using the equation:

$$P = C \times N \times A / w, \quad \text{where}$$

P = probability per year of an aircraft crashing into the PFSF
C = in-flight crash rate per mile
N = number of flights per year along the airway
A = effective area of the PFSF in square miles
w = width of airway in miles

NUREG-0800 states the in-flight crash rate as 4 E-10 per mile, which is appropriate to apply to the types of aircraft flying to and from Michael AAF. Information provided by the Dugway Proving Ground states that there are approximately 414 flights annually at this airfield. The average effective areas of the PFSF cask storage area and Canister Transfer Building are 0.0924 mi^2 and 0.0264 mi^2, respectively, calculated using Department of Energy (DOE) formulas. (DOE 1996) The width of the airway is 10 nautical miles (nm), or 10nm x 1.15 mile/nm = 11.5 miles. The probabilities of an aircraft impacting the cask storage area and Canister Transfer Building are therefore 1.3 E-9 and 3.9 E-10 per year, respectively. (PFS Aug. 1999)

2.2.2.2 Utah Test and Training Range

The UTTR is an Air Force training and testing range over which the airspace is restricted to military operations. It is divided into a North Area, located on the western shore of the Great Salt Lake, north of Interstate 80, and a South Area, located to the west of the Cedar Mountains, south of Interstate 80 and northwest of Dugway Proving Ground. (Cole 1999) The airspace over the UTTR extends somewhat beyond the range's land boundaries and is divided into military operating areas (MOAs) and restricted areas. The MOAs on the UTTR are located on the edges of the range, adjacent to the restricted areas. The PFSF site is located over 18 statute miles east of the eastern land boundary of the UTTR South Area and 8.5 statute miles northeast of the northeastern boundary of Dugway Proving Ground. The site lies within the Sevier B MOA, two statute miles to the east of the edge of restricted airspace. (PFS Aug. 1999)

Military aircraft flying in or around the UTTR South Area comprise three groups: 1) F-16 fighter aircraft flying down Skull Valley en route to the range; 2) aircraft conducting training in the restricted airspace on the range; and 3) aircraft departing the range via the Moser Recovery (Section 2.2.2.2.3). Aircraft flying in or around the UTTR North Area pose no credible hazard to the PFSF because of the distance from the facility.

2.2.2.2.1 F-16s Transiting Skull Valley

F-16 fighter aircraft fly north to south down Skull Valley, within Sevier B MOA, en route from Hill Air Force Base, near Ogden, Utah, to the UTTR South Area. The F-16s use the eastern side of Skull Valley as their predominant route of travel and typically pass approximately five miles to the east of the PFSF site. The U.S. Air Force has indicated that the F-16s fly between 1,000 and 4,000 ft. above ground level (AGL) and that in 1998 3,871 such flights passed through Skull Valley.

Because of the distance to the PFSF, the low altitude at which the F-16s fly, and the fact that Air Force pilots are instructed to avoid ground facilities in the event of a mishap in which the pilot retained control of the direction of the aircraft, it is not credible that a crashing F-16 would impact the PFSF. Nevertheless, an impact probability was calculated, using two methods, both of which conservatively assumed that the F-16 flights are uniformly distributed within the Sevier B MOA airspace in the vicinity of the PFSF.

First, the F-16 impact probability was calculated using the NUREG-0800 method. (PFS Aug. 1999) The Sevier B MOA airspace in the vicinity of the PFSF was treated as an airway with a width of 10 miles. Given the flight characteristics of the F-16, the PFSF has an average effective area of 0.063 mi^2 (cask storage area) and 0.0116 mi^2 (Canister Transfer Building). The number of flights through the valley was taken to be 3,871 per year, but because of the low altitude at which the F-16s fly through Skull Valley and the distance that an F-16 could glide after suffering an in-flight mishap, only 70 percent of those aircraft (uniformly distributed within the Sevier B MOA airspace in the PFSF vicinity) were determined to be capable of reaching the PFSF in the event of a crash. Thus, the number of aircraft used in the NUREG-0800 formula was 2,710. (PFS Aug. 1999) The crash rate for the F-16 was calculated from Air Force data to be 2.736 E-8 per mile. Accordingly, the average annual crash impact probabilities for the F-16s in Skull Valley were calculated to be 4.67 E-7 for the cask storage area and 8.57 E-8 for the Canister Transfer Building.

Second, the F-16 impact probability was calculated using the method by which the Department of Energy (DOE) calculated the crash impact probability for the Device Assembly Facility at the Nevada Test Site and for above ground facilities at the Yucca

Mountain geologic repository. (Kimura et al. 1998) The crash impact probability is given by:

$$P = N \times C \times (4/\pi^2) \times (1/R_p) \times A, \text{ where}$$

P = annual crash impact probability

C = in-flight crash rate per mile

N = number of flights per year

R_p = the effective radius of the facility plus the maximum distance a crashing aircraft could glide after suffering an in-flight mishap

A = effective area of the facility in square miles

The value for R_p is calculated as $(A/\pi)^{1/2} + g \times h$, where g is the glide ratio of the aircraft and h is the aircraft's starting altitude.

It was again assumed that the F-16s in Skull Valley were uniformly distributed by both altitude and lateral distance within the Sevier B MOA airspace in the vicinity of the PFSF. Under this methodology, F-16s flying at a distance greater than R_p from the PFSF would not impact the site. Because the effective area, A, for the Canister Transfer Building and for any particular cask configuration is fixed, R_p is a function of the altitude of the aircraft (assumed to be uniformly distributed) and the glide ratio. The glide ratio for the F-16 under ideal conditions is approximately 7, but to test the sensitivity of the results to the glide ratio, crash impact probabilities were calculated using glide ratios of 5, 7.5 and 10. Accordingly, average annual crash impact probabilities for the F-16s transiting Skull Valley were calculated to be:

Glide Ratio	Impact Probability	
	Storage Area	Canister Transfer Bldg.
5	4.77 E-7	8.72 E-8
7.5	4.31 E-7	7.86 E-8
10	4.04 E-7	7.36 E-8

2.2.2.2.2 Aircraft Training on the UTTR

According to the Air Force, 8,284 sorties were flown over the UTTR South Area in 1998. (PFS Aug. 1999) Those aircraft conducted a variety of activities, including air-to-air combat training, air-to-ground attack training, air-refueling training, and transportation to and from Michael Army Airfield (which is located beneath UTTR airspace). Hazards posed by aircraft flying to and from Michael Army Airfield are addressed in Section 2.2.2.1 above. Of the remaining aircraft, only fighter aircraft conducting air-to-air training represent a potential hazard to the PFSF, in that aircraft conducting air-to-ground attack training do so over targets that are located more than 20 miles from the PFSF site and aircraft conducting air refueling training do so on the far western side of the UTTR, over 50 miles from the site. The Air Force indicated that of the 8,284 sorties flown on the UTTR South Area in 1998, one-third, or approximately 2,118, involved fighter aircraft conducting air-to-air training.

The crash impact probability for fighter aircraft conducting air-to-air training on the UTTR was calculated as follows:

$P = C_a \times A_c \times A/A_p$, where

P = annual crash impact probability

C_a = total air-to-air training crash rate per square mile on the UTTR

A_c = the area of the UTTR from which aircraft could credibly impact the PFSF in the event of a crash

A = effective area of the PFSF in square miles

A_p = the footprint area, in which a disabled aircraft could possibly hit the ground in the event of a crash

The total air-to-air training crash rate per square mile on the UTTR, C_a, was calculated from the total number of hours flown in air-to-air training on the UTTR South Area (2,468), the crash rate per hour for fighter aircraft (the F-16) in maneuvering flight (i.e., combat training) (3.96 E-5), the distribution of air operations over the sectors of the UTTR nearest the PFSF, and the ground areas of those sectors. (PFS Aug. 1999) The area from which an aircraft could credibly impact the PFSF in the event of a crash, A_c, was taken to be the portion of the UTTR within 10 miles of the PFSF, in that a crashing aircraft more than 10 miles from the site would have to be under control of the pilot in order to glide and reach the site, and the pilot would guide any such aircraft away from the site, which is outside the land boundaries and the restricted airspace of the UTTR. The site effective area, A, was determined as in Section 2.2.2.2.1 above. The footprint area, A_p, was calculated by assuming that a crashing aircraft could glide in any direction up to a distance equal to the product of its starting altitude above ground and its glide ratio. Accordingly, the aircraft conducting air-to-air training over the UTTR were divided into altitude bands and an impact probability calculated for each band. As in Section 2.2.2.2.1 above, aircraft too low to glide to the PFSF in the event of a mishap would have no chance of impacting the site. The total average annual air crash impact probability for aircraft conducting air-to-air training on the UTTR South Area was calculated from the sum of impact probabilities of the altitude bands to be 1.63 E-7 for the cask storage area and 3.01 E-8 for the Canister Transfer Building.

2.2.2.2.3 Aircraft Using the Moser Recovery

Most aircraft returning to Hill Air Force Base from the UTTR South Area exit the northern edge of the range (away from the PFSF) in coordination with air traffic control. However, some aircraft returning to Hill from the UTTR South Area may use the Moser recovery route, which runs from the southwest to the northeast, approximately two miles from the PFSF site. (PFS Aug. 1999) The Moser route is only used during marginal weather conditions or at night under specific wind conditions which require the use of Runway 32 at Hill AFB. Based on information from local air traffic controllers, conservatively estimated, the Moser recovery is used by less than five percent of the aircraft returning to Hill. Thus, out of the 3,871 aircraft per year that fly from Hill AFB to the UTTR South Area (see Section 2.2.2.2.1 above), less than 194 aircraft per year would use the Moser recovery on their return flights.

The average annual crash impact probability for aircraft flying the Moser recovery was calculated using the methods used in Section 2.2.2.2.1 to calculate the crash impact probability for F-16s flying down Skull Valley. Using the NUREG-0800 method, the Moser recovery is defined as an airway with a width, w, of 10 nautical miles (11.5 statute miles) (equal to the width of military airway IR-420). The number of aircraft, N, is conservatively taken to be 194, the crash probability, C, is equal to 2.736 E-8 per mile, and the average effective areas of the site are 0.063 mi^2 (cask storage area) and 0.0116 mi^2 (Canister Transfer Building). Thus, the average annual crash impact probabilities are conservatively estimated to be 2.91 E-8 (cask storage area) and 5.4 E-9 (Canister Transfer Building).

Using the DOE method (see Section 2.2.2.2.1 above), N, C, and A are the same as for the NUREG-0800 method. The glide ratio, g, used in the calculation was conservatively

assumed to be 5 (impact probability increases with decreasing glide ratio) and the altitude was taken to be 10,500 ft., since aircraft flying the Moser recovery do so at 15,000 ft. MSL, which is 10,500 ft. AGL at the PFSF site. Accordingly, the average annual crash impact probabilities for F-16s flying the Moser recovery were conservatively calculated to be 1.34 E-8 (cask storage area) and 2.49 E-9 (Canister Transfer Building).

2.2.2.3 Aircraft Flying Federal Airways

Federal airway J-56 runs northeast to southwest at a distance (from the airway centerline) of 11.5 miles north of the PFSF. (PFS June 1999) Local air traffic controllers have indicated that fewer than 12 aircraft per day use the airway. The crash impact probability for aircraft on the airway was calculated for the PFSF using the method of NUREG-0800. Using the standard width for federal airways, J-56 is 8 nautical miles (9.2 statute miles) wide and the closest edge of J-56 is 6.9 miles from the PFSF. For facilities outside an airway, the effective width of the airway, w, is equal to the actual width plus twice the distance from the facility to the closest edge. Thus, J-56 has an effective width of 23 miles. The number of aircraft, N, is conservatively taken to be 12 per day, the crash rate, C, from NUREG-0800 is 4 E-10 per mile, and the average effective area of the PFSF for commercial airliners (the most common aircraft on the airway) is 0.113 mi^2 (cask storage area) and 0.031 mi^2 (Canister Transfer Building). Accordingly, the average annual crash impact probabilities are 8.4 E-9 (cask storage area) and 2.2 E-9 (Canister Transfer Building). (PFS June 1999, PFS Aug. 1999)

Federal airway V-257 runs north and south at a distance (from the airway centerline) of 19.5 miles east of the PFSF. (PFS June 1999) Local air traffic controllers have indicated that fewer than 12 aircraft per day use the airway. The crash impact probability for aircraft on the airway was calculated for the PFSF using the method of

NUREG-0800. V-257 is 12 nautical miles (13.2 statute miles) wide and its closest edge is 12.6 miles from the PFSF. Thus, V-257 has an effective width of 39 miles. The number of aircraft, N, is conservatively taken to be 12 per day, the crash rate, C, is 4 E-10 per mile, and the average effective area of the PFSF is 0.113 mi^2 (cask storage area) and 0.031 mi^2 (Canister Transfer Building). Accordingly, the average annual crash impact probabilities are 5.3 E-9 (cask storage area) and 1.4 E-9 (Canister Transfer Building). (PFS June 1999, PFS Aug. 1999)

2.2.2.4 General Aviation

There are no civilian airports within 25 miles of the PFSF. Thus it is highly unlikely that a general aviation aircraft would crash into the facility. (Cole 1999) Nevertheless, a crash impact probability for general aviation aircraft was calculated using National Transportation Safety Board (NTSB) crash data and the population of general aviation aircraft in the state of Utah. (PFS June 1999) The crash impact probability is equal to C_a x A, where C_a is the crash rate per square mile and A is the effective area of the PFSF. In 1996, the 162,342 general aviation aircraft in the United States suffered 20 fatal accidents, for a rate of 1.2 E-4 accidents per aircraft per year. There are 1,218 general aviation aircraft in the state of Utah, which covers an area of 82,076 mi^2. Thus the overall crash rate in Utah is equal to 1.78 E-6 per mi^2. NTSB crash data indicate, however, that only 4.1 percent of all general aviation crashes occur during the cruise mode of flight, which, because there are no airports nearby, is the mode in which general aviation aircraft would be flying near the PFSF. Thus the crash rate at the PFSF site would be equal to 7.3 E-8 per mi^2. The average effective area of the PFSF with respect to general aviation aircraft crashes is 0.055 mi^2 (cask storage area) and 0.0097 mi^2 (Canister Transfer Building). Accordingly, the average annual crash impact probabilities for general aviation aircraft are 4.0 E-9 (cask storage area) and 7.1 E-10 (Canister Transfer Building). (PFS June 1999, PFS Aug. 1999)

2.2.2.5 Cumulative Air Crash Impact Probability

The cumulative air crash impact probability is given in the table below.

Aircraft Crash Impact Probabilities		
Aircraft	**Annual Average Probability**	
	Cask Storage Area	**Canister Transfer Building**
Skull Valley F-16s	4.04 to 4.77 E-7 (DOE) 4.67 E-7 (NUREG)	7.33 to 8.7 E-8 (DOE) 8.57 E-8 (NUREG)
Aircraft Using the Moser Recovery	1.34 E-8 (DOE) 2.91 E-8 (NUREG)	2.49 E-9 (DOE) 5.4 E-9 (NUREG)
UTTR Aircraft	1.6 E-7	3.0 E-8
Aircraft on Airway J-56	8.4 E-9	2.21 E-9
Aircraft on Airway V-257	5.3 E-9	1.4 E-9
General Aviation Aircraft	4.0 E-9	7.1 E-10
Aircraft on Airway IR-420	1.3 E-9	3.9 E-10
Cumulative Probability	6.75 E-7 (NUREG)	1.26 E-7 (NUREG)

The table shows that the cumulative air crash impact probability is less than 1 E-6 for the cask storage area and the Canister Transfer Building. Qualitative factors discussed below show further that the true impact probability for both facilities is less than 1 E-7. Thus, air crash impact does not pose a credible hazard to the PFSF and the PFSF does not need to be designed to withstand the effects of air crash impacts.

2.2.2.6 Projected Growth in Air Traffic

The Federal Aviation Administration projects that the number of commercial aviation flights in the United States will increase by approximately 66 percent between 1998 and 2025, that the number of general aviation flights will increase by approximately 14 percent over the same period, and that the number of military flights will not increase during this period. (FAA 1999) Because most of the air traffic near the PFSF site is military, the growth in commercial and general aviation projected by the FAA will have no material effect on the air crash impact probability calculated for the facility.

2.2.2.7 Conservatism in the PFSF Air Crash Impact Probabilities

While the calculated total average annual air crash impact probabilities for the cask storage area and the Canister Transfer Building at the PFSF are 6.75 E-7 and 1.26 E-7, respectively, qualitative factors indicate that the true probabilities of an aircraft impacting either the cask storage area or the Canister Transfer Building are significantly lower, less than 1 E-7 per year. With respect to the F-16s transiting down Skull Valley en route to the UTTR South Area, these factors include the fact that, according to the U.S. Air force, the predominant route of choice for the F-16s is the east side of the Valley, approximately five miles from the site. Thus, the uniform distribution assumed in calculations in Section 2.2.2.2.1 is highly conservative. Further, the calculations are conservative in that they assume that a crashing aircraft could glide for several miles with the pilot taking no action to avoid the PFSF site. Realistically, the pilot would avoid the site or any inhabited area. The calculations also assume that a pilot would not eject from the aircraft, which would cause the aircraft to fall to the ground after traveling a short distance, but assume instead that the pilot would stay with the plane until it crashed. In fact, pilots are directed not to delay ejection from disabled aircraft below

2,000 ft. AGL. Finally, the Skull Valley F-16 calculations assume that F-16s will crash at the 10-year average rate rather than the more recent and lower 5-year average rate.

The calculations of the crash impact hazard posed by other aircraft are conservative as well. The calculations assume that aircraft conducting air-to-air training on the UTTR could hit the PFSF site from a distance up to 10 miles away, without considering that a pilot would consciously seek to steer the plane away from the PFSF and other built-up/populated areas or eject and cause the aircraft to hit the ground before it left the range. Similarly, the calculations do not assume that disabled civilian aircraft would attempt to fly to an airfield or an open area away from the PFSF to make an emergency landing. They also assume that the density of flight operations involving air-to-air training near the edge of the UTTR (near the PFSF) is the same as it is near the center, when in fact it is lower. Therefore, the true crash hazard from those aircraft is significantly lower than the calculated value and in fact is insignificant.

Finally, while no credit was taken for the resistance to the effects of an air crash impact provided by the concrete storage casks in which the spent fuel canisters will be located, the cask construction is robust enough that a significant fraction of the potential air crash impacts at the PFSF would not cause a release of radioactivity. (Davis et al. 1998) The casks could withstand the direct impact of a jet fighter or commercial airliner at a speed up to 340 knots, which is significantly greater than typical air crash impact velocities, and could withstand the impact of the great majority of general aviation aircraft altogether. (PFS June 1999; PFS Aug. 1999)

2.2.3 The Use of Ordnance on the UTTR

As discussed in Section 2.2.2.2, military aircraft conduct air-to-ground attack training using air-delivered ordnance on the UTTR South Area. Military aircraft also conduct

weapons testing, including the testing of cruise missiles. (Cole 1999; PFS June 1999) The use of air-delivered ordnance on the UTTR does not pose a significant hazard to the PFSF. The PFSF site is located 18 miles to the east of the easternmost land boundary of the range. Aircraft flying over Skull Valley are not permitted to have their armament switches in a release capable mode, and all switches are "safe" until the aircraft are inside DOD land boundaries. Weapons use on the UTTR is strictly controlled and the UTTR has never experienced an unanticipated munitions release outside of designated launch/release areas. Furthermore, the targets on the UTTR are all over 20 miles from the PFSF site and no run-in headings for weapons delivery cross Skull Valley. Any aircraft with hung ordnance are directed to Michael Army Airfield on a flight path that is not near Skull Valley. In addition, weapon systems that have a capability of exceeding range boundaries, such as cruise missiles, are required to have a Flight Termination System (FTS) installed prior to testing on the UTTR. FTSs are designed to destruct the weapons and terminate the weapons' flight paths in the event of an anomaly. The UTTR has never experienced an FTS failure. Therefore, weapons use on the UTTR does not pose a credible hazard to the PFSF and the facility does not need to be designed to withstand a weapon impact.

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Two major watersheds have been identified which can contribute runoff to the PFSF site area as described in Section 2.4.1.2. A relatively large watershed from the lower Stansbury Mountains in the east to the Lookout Mountain in the south is identified as Basin A and a relatively smaller watershed from the lower Cedar Mountains in the west is identified as Basin B (see Figure 2.4-1). Basin A is separated from Basin B by an earthen berm (PMF Berm) which will be constructed at the PFSF to control runoff from these offsite sources. This berm will ensure that there is no cross flow between basin A and B.

Analyses of the probable maximum precipitation (PMP) were performed to determine a probable maximum flood (PMF) for stormwater drainage Basins A and B. For an extremely conservative PMF (Q_{PMF} = 85,000 cfs), the Basin A PMF water elevation predicted at the southeast and northeast corner locations of the site is 4,468.8 and 4,456.7 feet, respectively. The site grade elevations at these locations are 4,476 and 4,462 feet, respectively, which are higher than the predicted flood elevations. Consequently, all SSCs that are classified as Important to Safety are located above the Basin A PMF flood plain.

Basin B stormwater runoff from the lower Cedar Mountain drains as a sheet flow toward the PFSF site. An earthen berm and drainage ditch system will be constructed on the south and west sides of the PFSF storage site to divert the PMF stormwater flows around the site and into the Skull Valley natural drainage system. Flood diversion berms will be constructed to resist erosive forces by using compacted soil with shallow side slopes (3:1 for the access road PMF diversion berm and 4:1 for the site PMF diversion berm). The berms will be seeded with a mixture of grasses and shrubs to provide soil stability. Ditches lined with riprap will be provided along the base of the flood diversion berms where stormwater is collected and conveyed. Consequently, all SSCs that are classified as Important to Safety are protected from the sheet flow associated with the Basin B PMF by the earthen berm. Therefore, forces due to flood waters and flood protection measures need not be considered in the design of SSCs that are classified as Important to Safety.

3.2.10 Seismic Design

The design of SSCs classified as Important to Safety shall consider loadings associated with the ISFSI design basis ground motion, which was determined by a probabilistic seismic hazard analysis as discussed in Section 2.6. Probabilistic analysis does not result in the determination of a unique Design Earthquake, such as is the case for a deterministic analysis. Instead, various scenarios and models are used to estimate the likelihood of earthquake ground motions at a site and systematically take into account uncertainties that exist in various hazard parameters. The results are in the form of hazard curves that express the mean annual probabilities or frequencies with which various levels of fault displacement and ground motion are expected to be exceeded. Regulatory Guide 1.29 (Reference 10) was used to define the SSCs that are required to withstand the loadings associated with the ISFSI design basis ground motions. These SSCs are identified in Regulatory Guide 1.29 as seismic Category I.

3.2.10.1 Input Criteria

Tooele County is located west of the Rocky Mountain Front, which is defined in 10 CFR 72.102 as approximately 104° west longitude. As described in Section 2.6, a probabilistic seismic hazard analysis was performed to establish the appropriate seismic design basis for the facility. This analysis applies the guidance in Regulatory Guide 1.165 (Reference 25) to the PFSF site. A return period of 2,000 years was determined to be appropriate (References 26 and 29).

In addition, a site-specific geotechnical investigation was performed to ensure the geological characteristics and soil are stable under earthquake conditions as described in Section 2.6.

codes and standards to ensure compatibility with SSCs that are Important to Safety and to maintain a level of quality that shall ensure that they will mitigate the effects of off-normal or accident-level events as required.

The cask transporter is classified as not Important to Safety but 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. In addition, a Technical Specification is proposed to ensure that the casks will not be lifted above the vendor's analyzed safe handling height. 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.

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 was not constructed or if it failed since the cask systems are designed to withstand severe flooding and full submergence. The berm is provided to minimize stormwater flowing across the site for ease of operations and maintenance activities. Complete blockage of air inlet ducts due to flooding is described in SAR Section 8.2.8 , which shows that the HI-STORM inlet ducts can be blocked for 92 hours without adverse effects and the TranStor storage cask inlet ducts can be blocked for an unlimited time. PMF flows are mitigated in the Canister Transfer Building by locating the ground floor elevation above the maximum elevation of flood water. In addition, forces due to flowing water would be insignificant and would not affect the stability to the casks due to the shallow depth of the flow across the site.

The closed circuit television (CCTV) is classified as not Important to Safety. The function of the CCTV is to assist in assessment of unauthorized penetration within the protected area as required per 10 CFR 73.51 (Reference 30). As noted in NUREG-1497 (Reference 31), adequate assessment may also be provided through onsite assessment by security personnel 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.

The PFSF radiation monitors are classified as not Important to Safety since they are not needed to prevent or mitigate any credible accident that would adversely affect public health and safety. 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, although the canisters are sealed, no airborne radioactivity is present during transfer operations. 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.

The temperature monitoring system is classified as not Important to Safety. 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 discussed 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.

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24. NUREG/CR-6407, (INEL-95/0551), Classification of Transportation Packaging and Dry Spent Fuel Storage System Components According to Importance to Safety, 1996.

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26. PFS Letter, Parkyn to Delligatti (NRC), Request for Exemption to 10 CFR 72.102(f)(1), dated April 2, 1999.

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29. PFS Letter, Parkyn to Delligatti (NRC), Request for Exemption to 10 CFR 72.102(f)(1), dated August 24, 1999.

30. 10 CFR 73.51, Requirements for the Physical Protection of Stored Spent Nuclear Fuel or High-Level Radioactive Waste.

31. NUREG-1497, Interim Licensing Criteria for Physical Protection of Certain Storage of Spent Fuel, November 1994.

TABLE 3.6-1
(Sheet 1 of 5)

SUMMARY OF PFSF DESIGN CRITERIA

DESIGN PARAMETERS	DESIGN CONDITIONS	APPLICABLE CRITERIA AND CODES
GENERAL		
PFSF Design Life	40 years	PFSF Specifications
Storage Capacity	40,000 MTU of commercial spent fuel	PFSF Specifications
Number of Casks	approximately 4,000 casks	PFSF Specifications
SPENT FUEL SPECIFICATIONS		
Type of Fuel	See Tables 3.1-1 and 3.1-2	HI-STORM SAR TranStor SAR
Fuel Characteristics	See Table 3.1-3	HI-STORM SAR TranStor SAR
STORAGE SYSTEM CHARACTERISTICS		
Canister Capacity	<u>HI-STORM</u> 24 PWR assemblies/canister 68 BWR assemblies/canister <u>TranStor</u> 24 PWR assemblies/canister 61 BWR assemblies/canister	HI-STORM SAR, Section 1.1 TranStor SAR, Section 1.1
Weights (maximum)	<u>HI-STORM</u> Storage Cask - 268,334 lbs. Loaded Canister - 87,241 lbs. Transfer Cask - 152,636 lbs. Shipping Cask - 153,080 lbs. <u>TranStor</u> Storage Cask - 222,200 lbs. Loaded Canister - 84,020 lbs. Transfer Cask - 126,230 lbs. Shipping Cask - 160,900 lbs.	 HI-STORM SAR, Table 3.2.1 " HI-STORM SAR, Table 3.2.2 Shipping SAR, Table 2.2.1 TranStor SAR, Table 3.2-1 " " Shipping SAR, Table 2.2-1

TABLE 3.6-1
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SUMMARY OF PFSF DESIGN CRITERIA

DESIGN PARAMETERS	DESIGN CONDITIONS	APPLICABLE CRITERIA AND CODES
STRUCTURAL DESIGN		
Wind	90 mph, normal speed	ASCE-7
Tornado	240 mph, maximum speed 190 mph, rotational speed 50 mph, translational speed 150 ft, radius of max speed 1.5 psi, pressure drop 0.6 psi/sec rate of drop	Reg. Guide 1.76
Tornado Missiles (at 84 mph)	1800 kg automobile 125 kg 8" armor piercing artillery shell 1" solid steel sphere	NUREG-0800, Section 3.5.1.4
Flood	N/A - PFSF is not in a flood plain and is above the PMF elevation	PFSF SAR Section 2.3.2.3
Seismic	0.53g, horz.(both directions) & 0.53 g vert. Design basis ground acceleration	10 CFR 72.102, Reg. Guide 1.165
Snow & Ice	P(g) = 45 psf	ASCE-7/County
Allowable Soil Pressure	Static = 4 ksf max Dynamic = Varies by footing type/size	PFSF SAR Section 2.6.1.12
Explosion Protection	N/A - PFSF is located beyond distances from transportation routes from where cargo explosions could cause overpressures > 1 psi.	Reg. Guide 1.91
Ambient Conditions	Low Temperature = -35°F Max. Annual Average Temp. = 51°F Average Daily Max. Temp. = 95°F Humidity = 0 to 100 %	NOAA Data-Salt Lake City UT Climate Data UT Climate Data
HI-STORM 100 Cask System Load Criteria	Canister: Internals: Storage Cask: Transfer Cask: } See HI-STORM SAR, Table 2.2.6	ASME III, NB ASME III, NG ASME III NF, ACI-349 ASME III NF, ANSI N14.6

CHAPTER 4

FACILITY DESIGN

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BWR assembly (HI-STORM SAR Tables 2.1.7 and 2.1.8). The analysis assumed HI-STORM storage casks are in an array, subjected to an 80° F annual average ambient temperature, with solar radiation. The annual average temperature takes into account both day and night, summer and winter temperatures throughout the year. The annual average temperature is the principal design parameter in the storage system design analysis because it establishes the basis for demonstration of long-term spent nuclear fuel integrity. The long-term integrity of the spent fuel cladding is a function of the averaged ambient temperature over the entire storage period, which is assumed to be at the maximum average yearly temperature in every year of storage for conservatism in the cladding service life components. The results of this analysis are presented in Tables 4.4.9 and 4.4.10 of the HI-STORM SAR for MPC-24 and MPC-68 canisters, respectively. The results, summarized in Table 4.2-3 of this SAR, indicate that temperatures of all components are within normal condition temperature limits.

Holtec considered stainless steel clad fuels in the thermal analysis, as discussed in HI-STORM SAR Section 4.3.1. Stainless steel cladding is less conductive than zircaloy clad fuel and the net thermal resistance of a basket full of stainless steel clad fuel is greater, which would result in higher cladding temperatures for stainless steel fuel assemblies having the same decay heat generation rate as zircaloy clad fuel. However, the design basis decay heat for stainless steel clad fuel is significantly lower than that of zircaloy clad fuel, as noted previously, and the allowable temperature limit for stainless steel cladding is considerably higher than for zircaloy cladding. Holtec determined that the reduction in heat duty is much more pronounced than the nominal increase in the resistance to heat transfer, and concluded that the peak cladding temperature for stainless steel clad fuel will be bounded by the results for zircaloy clad fuel and a separate analysis for stainless steel clad fuel was not required.

HI-STORM SAR Section 11.1.2 evaluates temperatures of the HI-STORM storage system for a maximum off-normal daily average ambient temperature of 100° F, an

increase of 20° F from the normal conditions of storage discussed above. This off-normal temperature conditions is based on a 24 hour average solar load in accordance with 10 CFR 71, which represents extreme environmental conditions or off-normal conditions. The maximum off-normal temperatures were calculated by adding 20° F to the maximum normal temperatures from the highest component temperature for MPC-24 and MPC-68. All the maximum off-normal temperatures are below the short term condition design basis temperatures (HI-STORM SAR Table 2.2.3). Therefore, all components are within allowable temperatures for the 100° F ambient temperature condition.

The thermal analysis in the HI-STORM SAR discussed above includes the following global assumptions: a) the concrete pad is assumed to be an insulated surface, i.e., no heat transfer to or from the pad is assumed to occur; b) adjacent casks are assumed to be sufficiently separated from each other (i.e., cask pitch is sufficiently large) so that their ventilation actions are autonomous from each other; c) the cask is assumed to be subject to full solar insolation on its top surface as well as view-factor adjusted solar insolation on its lateral surface. Second order effects such as insolation heating of the concrete pad, heating of feed air traveling downward between casks and entering the inlet ducts of the reference cask, and radiative heat transfer from adjacent spent fuel casks were not explicitly modeled in the HI-STORM SAR analysis (nor in the comparable analysis in the SAR for the TranStor storage system).

In order to address these second order effects, PFS had the HI-STORM storage cask vendor, Holtec, perform a revised analysis (Reference 60). (The HI-STORM system was selected since its generic thermal evaluation resulted in higher temperatures than for TranStor). The revised analysis specifically applies to HI-STORM storage casks at the PFSF site and assumes the following: a) exposed areas of the storage pad and the storage casks are heated by the sun, with the intensity of radiation derived from 10 CFR 71.71(c); b) conductive heat transfer takes place between both the pad and the

cask and the pad and the soil beneath it, assumed to be at 77°F; c) convective heat transfer takes place between both the pad and the ambient air and the cask and ambient air; and d) radiative heat exchange takes place between the pad and the cask and the pad and ambient air. In order to conservatively assess the heating effects of adjacent casks, the revised model assumes a reflecting and insulated hypothetical cylindrical boundary around the cask which reflects all heat radiated from the cask surface in the lateral direction back onto the cask. This heat reflection mirrors the heat produced by and radiated from adjacent casks (emitting design basis maximum heat) from all sides towards the cask being analyzed. The hypothetical boundary is insulated so that radiative cooling of the reference cask in the lateral direction is conservatively neglected. Further, the revised analysis models heating of the cooler ambient air descending between casks by both the surface of the concrete cask itself and the concrete pad before the air enters the reference cask inlet ducts.

The revised analysis determined that the relatively cooler ambient air that descends downward between the concrete storage cask-to-hypothetical cylindrical tank annulus and sweeps across the hot, sunlight-exposed concrete surfaces of the cask and pad is heated by approximately 3°K (5.5°F) prior to entering the cask inlet ducts. Peak component temperatures of the HI-STORM 100 system were evaluated for steady state conditions with ambient air temperatures of 100°F and 125°F, and were determined to be below the applicable short-term temperature limits. This is a conservative approach, since it would take several days for cask and canister temperatures to equilibrate to steady state conditions, but day-night average temperatures would not be maintained at the high ambient air temperatures assumed. This analysis shows that the secondary effects of heating the reference cask by adjacent casks and pre-heating the inlet air by cask and pad concrete surfaces are insignificant, with minimal impact on temperatures.

Holtec performed the above thermal analysis based on the minimum design spacing between the casks as allowed by the HI-STORM SAR for both its square and

rectangular HI-STORM cask arrays. The PFSF uses a N x 2 rectangular cask array in which the spacing between casks is larger than the minimum design spacing specified by the HI-STORM SAR for a N x 2 rectangular array. Therefore, the Holtec thermal analysis, based on the minimum cask spacing allowed for the HI-STORM cask storage system, is applicable to and bounds the PFSF cask array.

Specifically, the PFS casks are arranged for placement on a regular array of concrete pads (see Figure 1.2-1). The concrete pads are arranged to provide a lateral (edge to edge) spacing of 30 feet between adjacent pads. Each concrete pad is sized to accommodate a 2 x 4 array of casks with a 15 feet pitch in each direction. The resulting cask geometry is defined by two parameters: (i) cask pitch parameter (A) on a concrete pad, and (ii) lateral cask spacing parameter (B) (between rows of pads in the east-west direction). For the PFSF cask array, the A and B parameters are 15 feet and 45 feet (cask centerline to cask centerline) respectively.

The HI-STORM rectangular cask array geometry (shown in Figure 1.4.1 of Reference 1) is defined by parameters p_1, p_2 and p_3. The p_1 and p_2 parameters (both equal) correspond to the PFSF cask parameter A, and p_3 corresponds to PFSF parameter B. The minimum p_1 and p_2 spacing specified in the SAR for the HI-STORM storage system is 13.5 ft. and the minimum p_3 spacing is 38 feet. Consequently, the PFSF cask spacing parameters are larger than the minimum HI-STORM cask design basis spacing parameters for which the thermal analysis was performed. Therefore, the revised thermal calculation based on the minimum design spacing identified in the HI-STORM SAR bounds the PFSF cask array, and the temperatures calculated by the analysis are conservative for the PFSF cask array.

The HI-STORM storage system was also analyzed for a -40° F extreme low ambient temperature condition, as discussed in HI-STORM SAR Chapter 4. Holtec conservatively assumed zero decay heat generation from spent fuel, and no solar

radiation, resulting in all storage system components reaching the -40° F temperature. As stated in the HI-STORM SAR, all HI-STORM materials of construction will satisfactorily perform their intended function in the storage mode at this minimum temperature condition.

The PFSF site low ambient temperature of -35° F, maximum annual average temperature of 51° F (normal), and average daily maximum temperature of 95° F (off-normal) are bounded by the corresponding temperatures used for the HI-STORM storage system of -40° F, 80° F, and 100° F, respectively. Therefore, the thermal design of the HI-STORM storage system bounds the site specific design requirements.

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generated missile loads are the same as the PFSF design criteria described in Section 3.2.8, the TranStor design meets the PFSF design criteria.

G. Flood

Flood loads are addressed in TranStor SAR Section 11.2.4. The TranStor system is designed to withstand a flood up to a depth of 20-ft and a stream velocity of 24.6 fps without overturning the cask. The PFSF is above probable maximum flood conditions, therefore, the TranStor design meets the PFSF design criteria in Section 3.2.9 for flood conditions.

H. Earthquake

Earthquake loads are addressed in the TranStor SAR Sections 2.2.5 and 11.2.5. The TranStor SAR shows that the storage system will withstand the imposed loads and not begin to rock when subjected to a generic seismic event. A generic seismic event was defined for the TranStor system using response spectra curves from Regulatory Guide 1.60 (Reference 6) with a zero period acceleration of 0.38g horizontal (both directions) and 0.25g vertical. In addition, the cask vendor initially performed a site specific analysis and determined the HI-STORM storage casks will withstand the imposed loads and not tip over when subjected to the PFSF deterministic design earthquake (0.67g horizontal, 0.69g vertical – See Section 8.2.1). Also, a site specific analysis was performed for TranStor storage casks subjected to the design basis ground motion associated with the probabilistic seismic hazard analysis with the 2,000-yr return period (0.53g horizontal, 0.53g vertical). This analysis determined maximum rocking of the top of the cask of less than 1 inch and maximum sliding of less than 2.25 inches (Reference 64). The analyses concluded that the casks do not tip over, collide, nor slide off the storage pad for these earthquakes. Soil-structure interaction was considered in the site specific analyses. The seismic cask stability analyses are fully described in Section 8.2.1.

Since the cask is demonstrated to remain standing during an earthquake, the stresses in the basket can be evaluated by comparison to the off-normal handling analysis. The seismic accelerations are well bounded by the 17.5 g load used for the basket design during the off-normal handling event. Therefore, no additional evaluation of basket stresses is required.

Even though the storage cask will not tip over during an earthquake, the storage cask is conservatively analyzed for a hypothetical cask tip over event in TranStor SAR Section 11.2.10. The analysis shows that tip over results in deceleration that would not cause any critical damage to the storage cask or fuel basket.

Therefore, the TranStor storage system design meets the PFSF design criteria requirements in Section 3.2.10 for seismic design.

I. Explosion Overpressure

Explosion overpressure loads are addressed in TranStor SAR Section 11.2.8. Regulatory Guide 1.91 (Reference 9) requires a detailed review of the system for overpressures that exceed 1 psi. The TranStor storage system is analyzed and designed to withstand an explosion that could result in overturning or sliding the storage cask. The minimum pressure on the cask to produce this force was an overpressure of 5.4 psig. As shown in Section 8.2.4, the PFSF is not subject to explosions that are in excess of 1 psig. Since the PFSF will not see explosion pressures that exceed 1 psig, the TranStor design meets the PFSF design criteria in Section 3.2.7 for explosion accident loads as required per 10 CFR 72.122(c).

J. Fire

Fire is addressed in TranStor SAR Section 2.3.6. The TranStor storage system materials and location at the PFSF safely protects the spent fuel from fires in accordance with 10 CFR 72.122(c). The storage cask is highly resistant to the effects

of fire. The thick concrete walls of the storage cask are capable of protecting thebasket. Although the exposed layer of concrete may lose a portion of its strength, it would not disintegrate from an exposure to flame temperatures on the order of 1,500° F as specified in 10 CFR 71. In addition, any fire would be required to burn for a long time (days) before much of the wall thickness would be affected. The cask materials and limited use of combustibles at the site minimizes the effects of fire on the storage system. As discussed in Section 8.2.5, a storage cask is postulated to be involved in a diesel fuel fire, involving up to 50 gallons of diesel fuel spilled from the fuel tank of the cask transporter, which is calculated to burn for 3.6 minutes. This fire would not damage the storage cask concrete, and would have a negligible effect on canister and fuel temperatures. Therefore, the TranStor design meets the PFSF design criteria in Section 3.2.6 for accident-level thermal loads as required per 10 CFR 72.122(c).

K. Lightning

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).

4.2.2.5.2 Thermal Design

Thermal performance for the TranStor storage system is addressed in TranStor SAR Chapter 4. The TranStor system is designed to transfer decay heat from the spent fuel assemblies to the environment. Heat generated in the fuel assemblies is transferred to the surrounding inert atmosphere and the basket sleeves by free convection and radiation. It is further conducted through the sleeves towards the exterior of the basket assembly where it conducts, convects, and radiates through the cover gas to the canister shell wall. Heat is then convected to the air in the annulus between the canister shell and the storage cask internal liner, and radiated from the canister shell to the cask liner. Cooling air enters the inlet ducts at the bottom of the TranStor storage cask, flows up the annulus by passive convection, and exits at the top of the storage cask. A small amount of heat is conducted through the concrete to the outer surfaces of the storage cask, then convected to the air.

As discussed in TranStor SAR Chapter 4, several basic models were utilized for the thermal evaluation of the TranStor storage system.

These include:

- Air flow and temperature
- Storage cask body and canister exterior heat transfer
- PWR canister interior heat transfer
- BWR canister interior heat transfer

The ANSYS finite element code was used for calculating storage cask and canister temperatures. The design basis canister heat load of 26 kW was assumed in all the thermal analyses, corresponding to 1.083 kW per PWR fuel assembly and 0.426 kW per BWR fuel assembly. Results of the thermal analyses determined that the TranStor

system operates well within thermal design limits. Therefore, no degradation due to temperature effects on materials or components is expected. The analyses results represent maximum temperatures, since the heat source from the fuel assemblies decays with time. While allowable temperatures for the TranStor construction materials do not change, the fuel temperature limits decrease with time. However, SNC notes in TranStor SAR Chapter 4 that the heat load decays faster than the corresponding maximum allowable cladding temperatures, and margins between actual and allowable fuel cladding temperatures increase with time.

Off-normal and accident cases are described in TranStor SAR Chapter 11. The following steady state conditions have been analyzed:

1. Normal condition, average ambient temperature = 75° F, no solar radiation.
2. Off-normal condition, ambient temperature = 100° F, solar radiation.
3. Off-normal condition, ambient temperature = -40° F, no solar radiation.
4. Off-normal condition, ambient temperature = 75° F, no solar radiation, 1/2 of air inlets blocked.
5. Accident condition, ambient temperature = 125° F, solar radiation.
6. Accident condition, ambient temperature = 75° F, no solar radiation, all air inlets blocked.

The 75° F average ambient temperature represents an annual average temperature, which takes into account both day and night, summer and winter temperatures throughout the year. The annual average temperature is the principal design parameter in the storage system design analysis because it establishes the basis for demonstration of long-term spent nuclear fuel integrity. The long-term integrity of the spent fuel cladding is a function of the averaged ambient temperature over the entire storage period, which is assumed to be at the maximum average yearly temperature in every year of storage for conservatism in the cladding service life components.

The 100° F off-normal condition temperature is based on a 24 hour average solar load in accordance with 10 CFR 71, which represents extreme environmental conditions or off-normal conditions. Performance of the storage cask under this temperature condition is also addressed in Section 8.1.2 of the PFSF SAR.

The -40° F off-normal condition temperature represents a steady state abnormally cold temperature. The TranStor analysis presented in TranStor SAR Chapter 11 conservatively assumes no solar radiation.

The 125° F accident condition temperature represents an extreme hot ambient temperature, conservatively assuming full solar radiation.

The TranStor thermal analysis performed for the concrete storage cask verifies that material temperature limits are not exceeded for normal, off-normal, and accident conditions. The TranStor thermal analysis verifies that fuel cladding allowable temperature limits are not exceeded. The minimum temperatures for the TranStor system correspond to the coldest environmental conditions of -40° F and no heat load in the cask. However, even at these extreme conditions the components are above their minimum material temperature limits. The TranStor cask does not employ any temperature-sensitive features such as gaskets, packing, or O-rings.

The results of the thermal analysis for normal, off-normal, and accident conditions is shown in TranStor SAR Table 4.1-1. These results are summarized in Table 4.2-6.

The PFSF site low ambient temperature of -35° F, maximum annual average temperature of 51° F (normal), and average daily maximum temperature of 95° F (off-normal) are bounded by the corresponding temperatures used for the TranStor storage system of -40° F, 75° F, and 100° F, respectively. The heat generation of the fuel to be stored at the PFSF is bounded by the heat generation of the TranStor design basis fuel.

Therefore, the thermal design of the TranStor storage system bounds the site specific design requirements.

4.2.2.5.3 Shielding Design

Shielding for the TranStor storage system is addressed in TranStor SAR Chapter 5. The TranStor storage system is designed to maintain ALARA radiation exposure in accordance with 10 CFR 72.126(a). The concrete storage cask is designed to limit the average external dose rate (gamma and neutron) one meter from the cask to less than 15 mrem/hr on the sides (30 mrem/hr for stainless steel clad fuel) and 200 mrem/hr on top at the cover lid centerline based on TranStor design basis fuel. The design dose rates allow limited personnel access during canister closure operations.

Radiation shielding of the TranStor storage system is provided by the 0.75 inch thick steel canister shell, the 2 inch thick steel storage cask liner, and the 29 inch thick reinforced concrete cask wall. Axial shielding at the top is provided primarily by the steel canister shield and structural lids, which have a combined thickness of 11 inches. The 0.75 inch thick steel storage cask lid also provides axial shielding. The inlet and outlet ducts are configured to prevent direct radiation streaming from the spent fuel assemblies to the outside of the cask.

TranStor SAR Section 5.1 provides calculated dose rates on contact and at 1 meter for the top and side surfaces of the TranStor storage cask for design PWR and BWR fuel, which show that the above criteria are met by the TranStor Storage System. Maximum dose rates for TranStor design basis fuels are shown to be approximately 19 mrem/hr on contact with the side and 10 mrem/hr at 1 meter from the side of the TranStor storage cask; 157 mrem/hr on contact with the center of the lid and 135 mrem/hr at 1 meter from the top of the cask; and 7.5 mrem/hr on contact with the top vent and 14 mrem/hr on contact with the bottom vent.

Section 3.3.5 presents the radiological requirements for the PFSF. The requirements originate from 10 CFR 72.104, which requires that the annual dose equivalent to any real individual located beyond the OCA boundary not exceed 25 mrem to the whole body, and from 10 CFR 20.1301, which requires that the hourly dose to any member of the public in any unrestricted area not exceed 2 mrem as a result of exposure to radiation from the PFSF. As discussed in Chapter 7, the TranStor storage system shielding design achieves compliance with this requirement for the PFSF array, assumed to consist of 4,000 TranStor storage casks, configured as shown in the detail on Figure 1.2-1.

4.2.3 Cask Storage Pads

The design criteria for the cask storage pads are described in Chapter 3. The analysis methods and resulting design of the pads are described below.

4.2.3.1 Design Specifications

The design of the cask storage pads is in accordance with ANSI/ANS-57.9 (Reference 14) and ACI 349 (Reference 15) as identified in Chapter 3.

The cask storage pads are independent structural units constructed of reinforced concrete. Each pad is 30 ft wide by 64 ft long and 3 ft thick. The size of the pad is based on a center to center spacing of 15 ft for the storage casks. The ends of the storage pad are provided with an additional 2 ft in length to support both tracks of the cask transporter on the pad. The pads are nearly flush with grade for direct access by the cask transporter. Each cask storage pad is capable of supporting 8 loaded HI-STORM or TranStor storage casks.

An independent modular pad design was chosen to simplify the pad analysis (i.e. minimize the number of cask placement combinations) and to minimize the effects of thermal expansion. The modular pad design also provides for ease of construction by limiting the number of concrete pad construction and/or expansion joints required and allows for staged construction of the facility.

The cask storage pad design is based on a maximum loaded storage cask weight of 356,521 lbs. This maximum weight was associated with the HI-STORM storage cask loaded with an MPC-32 (32 fuel assembly capacity PWR canister) and envelopes the maximum loaded weight of both the TranStor and HI-STORM concrete storage casks proposed for use at the PFSF. The TranStor storage cask has a maximum loaded

weight of 307,600 lb. as shown on TranStor SAR Table 3.2-1. The HI-STORM storage cask has a maximum loaded weight of 356,521 lb. (MPC-32) as shown on HI-STORM SAR Table 3.2.1, Revision 1. The Holtec MPC-32 has the maximum weight of all of the HI-STORM series canisters and is conservatively used in the design, even though it is not proposed for use at the PFSF. The HI-STORM canisters proposed for use at the PFSF are the MPC-24 and the MPC-68 with maximum weights of 348,321 lb. and 355,575 lb., respectively, when in the HI-STORM storage cask, both of which are bounded by the weight of the MPC-32, when in the HI-STORM storage cask.

The cask storage pad design also considers the weight of the loaded concrete storage casks in combination with the seismic loads due to the PFSF deterministic design earthquake (0.67g horizontal, 0.69g vertical – See Section 8.2.1.1).

4.2.3.2 Plans and Sections

The site plan, which shows the locations of the concrete storage pads, is shown in Figure 1.2-1. A typical concrete storage pad plan, cross section, and details are shown in Figure 4.2-7.

4.2.3.3 Function

The function of the cask storage pads is to provide a level and stable surface for placement and storage of the TranStor and HI-STORM concrete storage casks containing the spent fuel canisters.

53. Appendix C Supplement to Generic Licensing Topical Report EDR-1, Summary of Regulatory Positions to be Addressed by Applicant for PFSF, 200/25 Ton Bridge Crane, Revision 0, November 1998.

54. Appendix B Supplement to Generic Licensing Topical Report EDR-1, Summary of Facility Specific Crane Data Supplied by Ederer Incorporated for PFSF, 150/25 Ton Semi-gantry Crane, Revision 0, November 1998.

55. Appendix C Supplement to Generic Licensing Topical Report EDR-1, Summary of Regulatory Positions to be Addressed by Applicant for PFSF, 150/25 Ton Semi-gantry Crane, Revision 0, November 1998.

56. Seismic Qualification Analysis 200 Ton Bridge Crane, PFSF, No. ANA-QA-147, Anatech Corporation, Revision 0, November 1998.

57. Seismic Qualification Analysis 150 Ton Semi-gantry Crane, PFSF, No. ANA-QA-148, Anatech Corporation, Revision 0, November 1998.

58. Regulatory Guide 1.92, Combining Modal Responses and Spatial Components in Seismic Response Analysis, Revision 1, February 1976.

59. ABAQUS/Standard, Version 5.7, User Manual, Example Problem Manual, and Theory Manual, Hibbitt, Karlsson, & Sorensen, Inc., Pawtucket, RI, 1997.

60. Holtec Report HI-992134, HI-STORM Thermal Analysis for PFS RAI, Rev. 0, dated February 9, 1999.

61. Holtec Report No. HI-992277, Multi-Cask Response at the PFS ISFSI, From 2000 Year Seismic Event, Revision 0, dated August 20, 1999.

62. PFSF Calculation No. 05996.02 SC-10, Seismic Restraints for Spent Fuel Handling Casks, Revision 0, Stone & Webster.

63. Ederer Incorporated letter from S. Anderson to J. Cooper / S. Macie of Stone & Webster, Review of New Higher Seismic Accelerations on the Cranes for the Skull Valley Project, dated September 1, 1999.

64. Holtec Report No. HI-992295, TranStor Dynamic Response to 2000 Year Return Seismic Event, Revision 0, dated September 17, 1999.

TABLE 4.1-1
(Sheet 7 of 7)

PFSF COMPLIANCE WITH GENERAL DESIGN CRITERIA (10 CFR 72, SUBPART F)

10 CFR 72 REQUIREMENT	REQUIREMENT SUMMARY	SAR SECTION WHERE COMPLIANCE IS DEMONSTRATED
72.128 (a) Spent fuel storage and handling systems	Spent fuel storage and other systems that might contain or handle radioactive materials associated with spent fuel must be designed to ensure adequate safety under normal and accident conditions.	• Section 3.3.7 provides the requirements for ensuring the safe design of the spent fuel storage and handling systems. • Sections 4.2 and 4.7 describe the design features of the storage and handling systems to provide adequate shielding, confinement, and heat removal capability. • Section 10.2.3 addresses the surveillance specifications for testing and monitoring some components Important to Safely.
72.128 (b) Waste treatment	Radioactive waste treatment facilities must be provided.	• Section 3.3.7 addresses radioactive waste provisions. • Chapter 6 addresses the generation of radioactive wastes.
72.130 Criteria for decommissioning	The ISFSI must be designed for decommissioning.	• Section 3.5 provides the requirements for decommissioning the site. • Section 9.6.3 describes the design considerations to facilitate decommissioning. • Decommissioning Plan (License Application, Appendix B) presents an overall description of the decommissioning requirements.

TABLE 4.2-1

PHYSICAL CHARACTERISTICS OF THE HI-STORM CANISTER

PARAMETER	VALUE
Outside Diameter	68.38 inches
Length, maximum	190.5 inches
Capacity	24 PWR assemblies 68 BWR assemblies
Maximum Heat Load	20.88 kW for PWR canister (MPC-24) 21.52 kW for BWR canister (MPC-68)
Material of Construction	Stainless steel
Weight, maximum (loaded with spent fuel)	79,987 lb (MPC-24) 87,241 lb (MPC-68)
Internal Atmosphere	Helium

TABLE 4.2-2

PHYSICAL CHARACTERISTICS OF THE HI-STORM STORAGE CASK

PARAMETER	VALUE
Height	231.25 inches
Outside Diameter	132.5 inches
Capacity	1 loaded canister
Max. Radiation Dose[1] 1 meter from surface: Side Top On contact with surface: Side Top Top vents Bottom vents	 14 mrem/hr 2 mrem/hr 29 mrem/hr 7 mrem/hr 32 mrem/hr 50 mrem/hr
Material of Construction	Concrete (core and lid) Steel (liner and shell)
Weight, maximum	268,334 lb (empty) 348,321 lb (with loaded MPC-24) 355,575 lb (with loaded MPC-68)
Service Life	>100 years

1. Dose rate is based on HI-STORM design basis fuel.

TABLE 4.2-3

HI-STORM STORAGE SYSTEM STEADY STATE TEMPERATURE EVALUATION UNDER NORMAL CONDITIONS OF STORAGE

COMPONENT	MPC-24 TEMPERATURE (°F)	MPC-68 TEMPERATURE (°F)	NORMAL CONDITION TEMPERATURE LIMITS (°F)
Ambient Air	80	80	N.A.
Storage Cask Outer Shell	131	131	350
Air Outlet	179	185	N.A.
Storage Cask Inner Liner	166	171	200 *
Canister Shell	295	301	450
Basket	657	722	725
Fuel Cladding	692	742	**

* 200°F is Holtec's normal condition temperature limit on the concrete. The storage cask steel structure has a normal condition limit of 350°F (HI-STORM SAR Table 2.2.3).

** The temperature limits in accordance with DCCG (gross rupture) criteria are 787°F (PWR) and 824°F (BWR). Permissible cladding temperatures for the HI-STORM system are in accordance with PNL criteria (i.e. 692°F PWR and 742°F BWR).

TABLE 4.2-4

PHYSICAL CHARACTERISTICS OF THE TRANSTOR CANISTER

PARAMETER	VALUE
Outside Diameter	66 inches
Length	192.25 inches maximum
Capacity	24 PWR assemblies 61 BWR assemblies
Maximum Heat Load	26 kW
Material of Construction	Stainless steel (shell), Carbon steel (internals)
Weight, maximum (loaded with spent fuel)	77,760 lb (PWR) 84,020 lb (BWR)
Internal Atmosphere	Helium

TABLE 4.2-5

PHYSICAL CHARACTERISTICS OF THE TRANSTOR STORAGE CASK

PARAMETER	VALUE
Height	222.5 inches maximum (depending on fuel length)
Outside Diameter	136 inches
Capacity	1 loaded canister
Maximum Radiation Dose[1]	
1 Meter from surface:	
Side	10 mrem/hr
Top	135 mrem/hr
On contact with surface:	
Side	19 mrem/hr
Top	157 mrem/hr
Top vent	7.5 mrem/hr
Bottom vent	14 mrem/hr
Material of Construction	Reinforced concrete Steel (inner liner)
Weight, maximum	222,200 lb (empty) 297,200 lb (loaded with PWR canister) 307,600 lb (loaded with BWR canister)
Service Life	>50 years

1. Dose rate is based on TranStor design basis fuel

TABLE 4.2-8

DYNAMIC PAD ANALYSIS MAXIMUM RESPONSE VALUES

(based on PFSF deterministic design earthquake – See Section 8.2.1.1)

PFSF DETERMINISTIC DESIGN EARTHQUAKE LOADING	MAXIMUM MOMENT (k-ft/ft)	MAX. SHEAR FORCE (k/ft)	MAXIMUM SOIL PRESSURE (k/ft^2)	MAX. HORIZONTAL TOTAL SOIL REACTION (kips)	
				Longitudinal	Transverse
2 Casks	344.3	76.3	3.34	670	730
4 Casks	132.6	25.2	2.10	1,195	910
8 Casks	114.0	27.7	2.80	1,330	2,030

TABLE 4.7-1

PHYSICAL CHARACTERISTICS OF THE HI-TRAC TRANSFER CASK

PARAMETER	VALUE
Inside Diameter	68.75 inches
Outside Diameter	94.625 inches
Height	203.50 inches
Materials of Construction	Steel (inner and outer shell) Lead (gamma shield) Water (neutron absorber)
Weight (empty)	152,636 lb
Maximum Working Dose Rate[1] (1 meter from surface) Side	 49 mrem/hr

[1] Dose rates are based on HI-STORM design basis fuel.

Table 4.7-2

HI-TRAC TRANSFER CASK STEADY STATE TEMPERATURE EVALUATION

COMPONENT	TEMPERATURE (°F)	SHORT-TERM TEMPERATURE LIMITS (°F)
Ambient Air	100	N/A
Transfer Cask Outer Shell	223	700
Top Neutron Shield	175	300
Bulk Average Water Jacket	269	307
Transfer Cask Inner Surface	323	600
Canister Shell	459	775
Basket	884	950
Fuel Cladding	902	1058

TABLE 4.7-3

PHYSICAL CHARACTERISTICS OF THE TRANSTOR TRANSFER CASK

PARAMETER	VALUE
Inside Diameter	67 inches
Outside Diameter	86 inches
Height	204 inches maximum (depends on fuel length)
Materials of Construction	Steel (inner and outer shell) Lead (gamma shield) Polymer (neutron absorber)
Weight (empty)	126,230 lb max.
Maximum Working Dose Rate[1] (1 meter from surface) Side	 79 mrem/hr

1. Dose rates are based on TranStor design basis fuel.

CHAPTER 5

OPERATION SYSTEMS

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5.2 SPENT FUEL CANISTER HANDLING SYSTEMS

5.2.1 Spent Fuel Canister Receipt, Handling, and Transfer

An operational description for the systems used for the receipt and transfer of spent fuel canisters is provided in the following paragraphs. Special features of these systems to ensure safe handling of the spent fuel canisters are also described.

5.2.1.1 Spent Fuel Canister Receipt

5.2.1.1.1 Functional Description

The shipping casks and impact limiters comprise the system in which the spent nuclear fuel canisters are contained when they arrive at the PFSF. The shipping cask system protects the enclosed spent fuel canister from physical damage, provides shielding, and allows sufficient cooling of the canister while enroute to the PFSF.

5.2.1.1.2 Safety Features

Safety features of the system include the impact limiters, which help protect the spent fuel shipping cask during transportation, and the design, materials, and construction of the shipping casks, which provide gamma and neutron shielding, conductive and radiant cooling, criticality control, and structural strength to protect the spent fuel canister. A tamperproof device on the cask provides indication of an unauthorized attempt to obtain access to the cask. These safety features are fully described in the HI-STAR and TranStor shipping SARs.

5.2.1.2 Spent Fuel Canister Handling

5.2.1.2.1 Functional Description

The overhead bridge and semi-gantry cranes perform handling functions inside the Canister Transfer Building for the shipping cask, the transfer cask, and the TranStor canister. The canister downloader, bolted on top of the HI-TRAC transfer cask is used to raise and lower the HI-STORM canister.

Shipping and transfer cask handling components include the shipping cask and transfer cask lifting yokes, trunnions, and seismic support struts.

The storage cask handling component consists of the storage cask lifting attachments, cask transporter, and the overhead bridge crane, if needed.

The canister handling components consist of the lifting slings, HI-STORM canister lifting cleats, and TranStor canister hoist rings.

5.2.1.2.2 Safety Features

Safety features of the overhead bridge and semi-gantry cranes include single-failure-proof designs for sustaining the load upon failure of any single component, limit switches for prevention of hook travel beyond safe operating positions, and provisions for lowering a load in the event of an overload trip. The cranes are classified as ASME NOG-1 Type I cranes. 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. Design requirements for the cranes require testing, inspection, and maintenance

activities on the cranes in accordance with 10 CFR 72.122(f) which, are performed 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 ensure adequate safety margins are provided to prevent damage to the shipping cask, canister, or storage cask during normal, off-normal, and accident conditions. The crane designs 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.

Safety features of the HI-TRAC downloader, used to raise and lower the HI-STORM canister during canister transfer operations, include a single-failure-proof design for sustaining the load upon failure of any single component and/or loss of hydraulic pressure as described in the HI-STORM SAR.

Safety features of the shipping and transfer cask handling components include single-failure-proof lift capacity or equivalent safety factor as described in the HI-STORM and TranStor SARs.

Use of seismic support struts ensure the shipping and transfer cask do not topple over during an earthquake. Safety features of the seismic support struts include using standard rigid support assemblies that conform to ASME III, NF requirements for Class 2 nuclear grade supports. As such, the struts are 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. Each cask utilizes 2 struts, which provide restraint in both orthogonal horizontal directions.

There are no safety features associated with the cask transporter since the storage cask is designed to withstand drops that could result from a failure associated with the transporter lift components. The transporter is designed such that the lift mechanism can only lift the storage cask within lift heights specified by the Technical Specifications. The hydraulic lift cylinders are 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 inform the operator 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.

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

5.2.1.3 Spent Fuel Canister Transfer

5.2.1.3.1 Functional Description

The transfer cask is used for transfer of the spent fuel canister between the shipping cask and the storage cask. The transfer cask protects the spent fuel canister from physical damage and provides radiation shielding.

5.2.1.3.2 Safety Features

The transfer casks provide radiation shielding and act as special lifting devices when carrying a canister loaded with spent fuel. The transfer cask lifting trunnions are designed and tested to the single-failure-proof requirements of NUREG-0612 (Reference 6) and ANSI N14.6 (Reference 7) so that canisters can be lifted by the transfer cask without the requirement to analyze a transfer cask drop. However, annual

testing requirements per ANSI N14.6 of the transfer cask trunnion welds is not performed since the welds cannot be accessed for testing and NDE.

The transfer casks consist of cylindrical steel liners with a lead gamma shield and a neutron shield. Two trunnions are provided for transfer cask handling. The transfer cask has movable shield doors at the bottom to allow raising the canister into the transfer cask, lowering of the canister into the storage or shipping cask, or to support the canister weight and provide shielding while in the transfer cask. The doors slide in steel guides along each side of the transfer cask. Steel pins or bolts are used to prevent inadvertent opening of the doors. Roller bearings on the HI-TRAC transfer cask enable the cask doors to be manually operated. Hydraulic cylinders are used to open the TranStor transfer cask doors.

The transfer casks are designed to prevent the canister from being lifted beyond the top of the cask, which would expose the canister and cause high radiation doses. On the HI-TRAC transfer cask, the canister downloader, which raises the canister, is bolted on top of the cask. The canister can only be lifted up to the downloader hoist mechanical stops and is prevented from being raised beyond the top of the HI-TRAC cask. On the TranStor transfer cask, the top cover of the transfer cask is designed to stop the canister and prevent the crane from inadvertently lifting the canister up and out of the transfer cask while being raised.

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

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

5.2.2 Spent Fuel Canister Storage

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

5.2.2.1 Safety Features

Safety features include a passive dry cask design and administrative controls. The canister is enclosed in the cavity of the concrete storage cask, which protects the canister from severe natural phenomena (such as tornado-driven missiles), provides required shielding of the canister, and flow paths for natural convection cooling. The results of analyses of hypothetical storage cask tipover events are described in Section 8.2.6, where it is concluded that the canister will remain intact inside the storage cask and canister internals will not be damaged. Safety features are discussed in greater detail in Chapter 4, Chapter 8, and the HI-STORM and TranStor SARs.

6.4 SOLID WASTES

All spent fuel stored at the PFSF is contained in sealed canisters. Under all normal, off-normal, and credible accident conditions of transport, handling, and storage, the potential does not exist for breach of the canister and release of radioactive material associated with spent fuel from inside the canister.

There is a potential for the presence of some contamination on the external surfaces of canisters as a result of submergence in spent fuel pools during spent fuel loading operations at the originating nuclear power plants, even though measures are taken to prevent contamination (see Chapter 7 of the HI-STAR and TranStor shipping SARs). Following fuel loading operations at the originating nuclear power plants, a smear survey is performed to determine removable contamination levels on accessible outer canister surfaces near the top of the canister (canister lid and approximately 3 to 6 inches on canister sides down from the lid). In addition, smears are taken on internal surfaces of the transfer cask, following transfer of the canister from the transfer cask into the shipping cask, and removable contamination levels on the transfer cask internal surfaces are considered to be representative of removable contamination levels on the outer surfaces of the canister. In the event canister removable contamination levels (measured on accessible canister surfaces or inferred from levels measured inside the transfer cask) exceed the criteria specified in Chapter 10, the canister will not be released for shipment to the PFSF. Canisters with levels of removable contamination above the specified limit must be decontaminated prior to release for transport to the PFSF.

Once the shipping cask arrives at the PFSF and its closure is removed, a smear survey of accessible portions of the canister is again performed. If removable surface contamination levels exceed the limits specified in Section 10.2.2.1, the canister is returned to the originating nuclear power plant for decontamination.

Even with these measures to assure canister external surfaces are relatively free of removable contamination, contamination surveys are performed on outer surfaces of storage casks, following loading of canisters into the storage casks in the Canister Transfer Building. Under off-normal conditions, such as a canister mishandling event, it is considered possible for removable contamination to be released from the external surfaces of a canister, possibly depositing contamination upon surfaces of the shipping, transfer, or storage casks. Any necessary decontamination of these casks will be performed using dry methods. If such decontamination is necessary, a small quantity of solid LLW may be generated, consisting of smears, disposable clothing, tape, blotter paper, rags, and related health physics material. This material will be collected, identified, packaged in suitable LLW containers (such as standard 55-gallon steel drums that comply with transportation and disposal requirements), marked in accordance with 10 CFR 20 requirements, and temporarily stored in the LLW holding cell of the Canister Transfer Building while awaiting removal to a LLW disposal facility. The LLW holding cell is regularly surveyed and inventoried, including inspection of the materials stored, to evaluate the status of materials and controls (e.g., physical condition of containers, access control, posting).

The volume of solid waste is expected to be minimal since the occurrence of contamination would be due to an off-normal event.

Any wastes that are generated are controlled, stored, and disposed in compliance with the requirements of 10 CFR 20. All solid wastes are packaged for removal to a LLW disposal facility. Packaging complies with requirements specified by 49 CFR 171-177, 10 CFR 71, and the disposal facility criteria, as applicable.

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.

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

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

7.2.1.3 Neutron Source

Neutrons are produced in the active fuel region by spontaneous fission sources from various actinides and alpha/neutron reactions. The primary neutron source is the spontaneous fission of Cm-244. HI-STORM neutron sources for the PWR and BWR fuels, determined using the SAS2H and ORIGEN-S codes, are shown in HI-STORM SAR Tables 5.2.16 through 5.2.20. These tables present the neutron sources for HI-STORM reference fuels, including intact Zircaloy and stainless steel clad fuels and damaged BWR fuel. The neutron source strengths for the Zircaloy clad fuels are greater than the source strengths for the stainless steel clad fuels, for all neutron energy groups. The neutron sources for PWR and BWR fuels for TranStor, taken from the OCRWM LWR Database, are shown in the TranStor SAR Tables 5.2-5 and 5.2.6.

Unlike the gamma source spectrum, the neutron source spectrum does not vary significantly with fuel burnup level or cooling time. As noted above, SNC used the low initial enrichments permitted by the OCRWM LWR Database with each assumed

burnup. Holtec assumed enrichments of 3.7 percent for the PWR fuel and 3.4 percent for the BWR fuel, which are below the average enrichments normally used to obtain the burnups analyzed, as indicated in the OCRWM LWR Database. Low initial enrichments are assumed since the neutron source strength increases substantially as initial enrichment decreases for LWR fuel of a given burnup.

7.2.2 Airborne Radioactive Material Sources

Loading of spent fuel into the canisters takes place at the originating nuclear power plants where procedures are in place to prevent the spread of contamination. The canisters are dried and seal welded within the controlled environment of the originating nuclear power plant. Once the canister is dried and seal welded, there are no credible off-normal events or accidents that will cause breach of the canister and thus no credible releases of airborne radioactivity from the spent fuel assemblies.

During normal operation of the PFSF, the only potential source of airborne radioactivity is from loose surface contamination on the canister exterior, which could potentially be deposited there during fuel loading operations. However, measures are implemented at the originating nuclear power plants to prevent contaminating the canisters. For wet transfers in spent fuel pools utilizing the HI-STORM system, an inflatable seal is placed in the annulus between the canister and the HI-TRAC transfer cask and the annulus is filled with demineralized water (borated for PWR fuel pools) prior to submerging the empty transfer cask/canister in the pool. The seal prevents contaminated spent fuel pool water from entering the annulus and contaminating the outer surface of the canister. For the TranStor fuel loading operation, a shield ring is placed in the annulus between the canister and transfer cask, which reduces the area of the annulus, and demineralized water or filtered fuel pool water (borated for PWR fuel pools) is continuously injected into the transfer cask/canister annulus. This water flows out of the area at the top of the annulus where the shield ring is installed, preventing the

relative to the transfer casks at which dose rates were calculated. The dose locations for point 4 differ slightly for the two transfer casks, with the HI-TRAC point located at the side of the cask above the neutron shield and the TranStor transfer cask point located at the top of the cask directly above the annulus between the canister and the inside of the transfer cask.

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

Each of the storage cask vendors calculated gamma and neutron dose rates at various distances from a single storage cask, assuming fuel with conservative burnup and cooling time representative of high radiation source fuel expected to be stored at the PFSF, instead of reference fuel. The results of these single storage cask calculations were then used in support of the dose rate vs. distance analyses for the fully loaded PFSF array of 4,000 casks.

The basis for these calculations is that all 4,000 casks contain 40 GWd/MTU burnup and 10-year cooled PWR spent fuel, with a low initial enrichment assumed for this burnup. 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).

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 12 years for the Holtec HI-STAR shipping cask system (Revision 8 of the HI-STAR Shipping Cask SAR), and 8 years for SNC's TranStor shipping cask system (Revision A of the TranStor Shipping Cask SAR); 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.

DOE's Energy Information Administration's Service Report entitled "Spent Nuclear Fuel Discharges from U.S. Reactors - 1994" (Reference 19), 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 PWRs was approximately 19,000 MTU, and the inventory from BWRs approximately 11,000 MTU, for a total inventory of approximately 30,000 MTU (Table 5 of Reference 19). 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 Reference 19 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 Reference 19 to be approximately 32.4 GWd/MTU (Reference 20).

Weighted average cooling times were also calculated from the data presented in Tables

6 and 7 of Reference 19, 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 (Reference 20).

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.

Holtec computed dose rates at the surface of a HI-STORM storage cask and at various distances from the cask, assuming fuel with 40 GWd/MTU burnup and 10-year cooling time, using the MCNP code. The HI-STORM SAR shows that a HI-STORM storage cask containing a PWR canister (MPC-24) has higher contact dose rates on the top and at the duct openings than a HI-STORM storage cask containing a BWR canister (MPC-68), for fuel of identical burnup and cooling times. The dose rate at the midplane for an overpack containing a PWR canister is essentially the same as that for a storage cask containing a BWR canister. Therefore, it was determined that the dose rates from a HI-STORM storage cask containing a PWR canister will bound dose rates from a storage cask containing a BWR canister, and dose rates at distances from the PFSF array were assessed conservatively assuming all storage casks are loaded with PWR canisters. The primary radiation source terms accounted for in Holtec's analysis were: gamma and neutron sources from the decay of fission products and the gamma source from the decay of Co-60 in the fuel assembly end-fittings. Secondary radiation source terms accounted for were secondary neutrons from fast fission in the fuel and secondary

gammas from prompt neutron interaction in the canister and overpack. The canister and overpack were modeled in full three-dimensional detail using the MCNP code, in the same manner that the storage cask was modeled with reference fuel, as described in the HI-STORM SAR. A surface source file was generated containing information regarding neutron and gamma tracks of the radiation leaving the surface of a single storage cask. This file was then used in the computation of dose rates at various distances from the cask, and in modeling the cask array.

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

Information from the OCRWM LWR Database shows that the total neutron source strength for 40 GWd/MTU, 10-year cooled PWR fuel is 0.83 times the total neutron source strength for 40 GWd/MTU, 5-year cooled fuel. Since the neutron source spectrum does not vary significantly with fuel cooling time, the single neutron dose rate

vs. distance values previously calculated by SNC (Reference 12) for the 40 GWd/MTU, 5-year cooled PWR reference fuel were multiplied by 0.83 to yield the single cask neutron dose rate vs. distance data for 40 GWd/MTU, 10-year cooled fuel.

The single storage cask dose rate versus distance data for HI-STORM and TranStor casks containing 40 GWd/MTU, 10-year cooled fuel are shown in Tables 7.3-5 and 7.3-6 for the following four components: gammas and neutrons from the cask side and top. This data was used, along with the layout of the cask array at the PFSF (see PFSF Site Plan, Figure 1.1-2), to determine dose rates at various distances, including the RA fence and the OCA boundary from the PFSF array of 4,000 casks. The following paragraphs summarize the methodology used by the vendors and results of dose rate projections from the PFSF array, assuming the PFSF is filled with either HI-STORM or TranStor storage casks containing 40 GWd/MTU, 10-year cooled fuel.

HI-STORM

Holtec used the dose rate vs. distance data from a single HI-STORM storage cask, shown in Table 7.3-5, to project dose rates at various distances from the PFSF array, assumed to be filled with 4,000 HI-STORM storage casks containing 40 GWd/MTU, 10-year cooled fuel (Reference 13). The dose rate contributions from the tops and sides of the casks were separately analyzed using the MCNP code. The total dose rate from the tops of casks is a summation of the gamma and neutron top doses from all 4,000 casks, where the actual distance from each cask to the dose receptor is accounted for.

The total dose from the sides of the casks is a summation of side doses from all 4,000 casks where the distances within the facility and self-shielding of one row of casks by another row are accounted for. The fraction of radiation blocked by a cask directly in front of another cask was calculated by MCNP and used in the determination of total side dose rates. Self-shielding effects are different along the north/south faces than along the east/west faces because of the different geometries, as seen in Figure 1.2-1.

It was impractical to model the entire facility in MCNP, therefore, numerous smaller calculations were performed for configurations of several casks and combined in a conservative fashion to accurately estimate dose rates from the sides of the casks at various distances from the PFSF array. Modeling of configurations of casks determined: the number of casks in a single row along the east/west and north/south faces that effectively constitute an infinite line at various distances from the dose receptor; the fractional increases in dose rates when a second row of casks is added directly behind the first row along the east/west and north/south faces at various distances; and the fractional increases when two more rows of casks are added behind the first two rows (adding a second column of storage pads, with two rows of casks per pad) along the east/west faces at various distances.

The results of the dose rate vs. distance analysis for the PFSF array full of HI-STORM storage casks are given in Table 7.3-7. Total dose rates at the RA fence (150 ft from the nearest storage pads) at the north side of the array are 1.19 mrem/hr. The RA fence south of the array is 265 ft from the nearest storage pads, so will have lower dose rates. Total dose rates at the RA fence on the east and west sides of the array (also 150 ft from the nearest storage pads) are 0.98 mrem/hr. These dose rates are less than the 2 mrem/hour criteria for unrestricted areas specified in 10 CFR 20.1301 and are therefore acceptable. The total dose rates at the OCA boundary were calculated to be 1.94 E-3 mrem/hr at a point on the boundary 1,969 ft (600 meters) north of the RA fence, and 1.17 E-3 mrem/hr at a point on the boundary 600 meters west of the RA fence. Dose rates will be lower at points along the south and east sides of the OCA boundary, since these points are further from the storage casks than the north and west OCA boundaries. Conservatively assuming a hypothetical individual spends 2,000-hours per year at the north OCA boundary results in a maximum annual dose of 3.88 mrem. This is less than the 25 mrem criteria specified in 10 CFR 72.104 for maximum permissible annual whole body dose to any real individual located beyond the controlled area boundary and is therefore acceptable.

TranStor

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

The casks are spaced at a 15-ft pitch, resulting in about 3.7 ft between adjacent TranStor casks. Because of the close proximity of the casks and the fact that casks are positioned in evenly spaced rows and columns and not staggered, the analysis assumed that cask side dose rate contributions from all casks except those on the edge of the ISFSI are completely blocked by other casks. Streaming could be significant where substantial space exists between storage casks, such as the 150-ft spaces between storage pad quadrants, and the 30-ft spacing between columns of storage pads. To conservatively account for streaming in these spaces between casks, SNC assumed that the spaces between columns and the space between quadrants in the front row of casks are filled in with additional casks spaced at 15-ft pitch. Thus, there

are assumed to be two additional casks in each 30-ft space between columns and 10 additional casks in the 150-ft space between quadrants. This "hole plugging" assumption results in an additional 56 casks assumed to be in the front row along the north edge of the PFSF, for a total of 106 casks assumed in the front row for the dose rate vs. distance analysis. The dose rates at various distances from the side of a single TranStor storage cask containing 40 GWd/MTU, 10-year cooled fuel are given in Table 7.3-6. For a dose receptor point assumed to be centered in front of the north side of the PFSF storage area, SNC determined the distance to the nearest (center) cask, and the dose rate at this distance. This dose rate was multiplied by 106 to estimate the total side dose from the assumed 106 casks along the north edge. This approach is conservative since it assumes that all the casks in the front row are at the same distance from the dose receptor point, whereas in actuality casks near the east and west ends of the front row are much further from the dose receptor point than the nearest casks in the center of the row.

The results of the dose rate vs. distance analysis for the PFSF array full of TranStor storage casks are given in Table 7.3-8. Total dose rates at the RA fence at the north side of the array (highest doses) are calculated to be 0.45 mrem/hr. This is less than the 2 mrem/hour criteria for unrestricted areas specified in 10 CFR 20.1301 and is therefore acceptable. The total dose rate at the OCA boundary 600 meters north of the RA fence was calculated to be 1.21 E-3 mrem/hour. It was determined that the dose rates at the north OCA boundary were higher than those along the other sides, even though the west OCA boundary is the same distance from the storage pads (646 meters) as the north boundary. Conservatively assuming a hypothetical individual spends 2,000 hours per year at the portion of the OCA boundary fence with the highest calculated dose rate from the storage cask array results in an annual dose of 2.42 mrem. This is less than the 25 mrem criteria specified in 10 CFR 72.104 for maximum permissible annual whole body dose to any real individual located beyond the controlled area boundary and is therefore acceptable.

Dose at Nearest Residence

The approximate distance to the nearest residence is 2 miles east-southeast of the PFSF. At distances greater than several thousand feet, the accuracy of computer code calculational techniques becomes questionable. The error bands in statistical codes like MCNP become large and for deterministic codes like Skyshine, the conditions may be beyond the range of the codes data. However, both Holtec and SNC estimated dose rates that could occur at long distances from the PFSF, assuming the PFSF array of 4,000 HI-STORM storage casks loaded with 40 GWd/MTU, 10-year cooled PWR fuel, and conservatively taking no credit for any intervening shielding from berms, natural terrain or buildings at the PFSF. Holtec estimated the dose rate at 2.0 miles from the PFSF by extrapolating the maximum dose rate at the OCA boundary (1.94 E-3 mrem/hr) out to a distance of 2.0 miles using a power curve. The result was 2.7 E-6 mrem/hr, which would result in an annual dose of 0.024 mrem at a distance of 2.0 miles from the OCA boundary, assuming a person continually present (8,760 hrs/yr) at this location.

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

7.3.4 Ventilation

10 CFR 72.122(h)(3) requires that ventilation systems and off-gas systems be provided where necessary to ensure the confinement of airborne radioactive particulate materials during normal or off-normal conditions. However, there are no special ventilation systems installed in the PFSF facilities. There are no credible scenarios that would require installation of ventilation systems to protect against off-gas or particulate filtration.

7.3.5 Area Radiation and Airborne Radioactivity Monitoring Instrumentation

10 CFR 72.122(h)(4) requires the capability for continuous monitoring of the storage system to enable the licensee to determine when corrective action needs to be taken to maintain safe storage conditions. This is not applicable to the PFSF because the canisters are sealed by welding and with the canisters in storage casks and the casks on the storage pads, there are no credible events that could result in releases of radioactive material from within the canisters or unacceptable increases in direct radiation levels. Area radiation and airborne radioactivity monitors are therefore not needed at the storage pads. However, TLDs will be used to record dose rates in the RA and along the OCA boundary fence. TLDs provide a passive means for continuous monitoring of radiation levels and provide a basis for assessing the potential impact on the environment.

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.

Local radiation monitors with audible alarms will be installed in the Canister Transfer Building. These will provide warning to personnel involved in the canister transfer operation of abnormal radiation levels that could possibly occur during transfer operations. Because of the measures taken at the originating nuclear power plants to minimize loose surface contamination levels on the exterior of the canisters during fuel loading operations, as discussed in Section 7.2.2, and limits on surface contamination concentrations, as discussed in Chapter 10, it is unlikely that canister transfer operations would generate significant levels of airborne contaminants. Airborne radioactivity concentrations will be detected by continuous air monitors located in the exhaust of each canister transfer cell. The continuous air monitors will include local alarms to warn operating personnel in the unlikely event of an airborne release, remote alarm in the Security and Health Physics Building alarm station to ensure coverage at all times, and charting capability to provide data necessary to quantify any release. The radiological alarm systems will be designed with provisions for calibration and operability testing. There are no liquid or gaseous effluent releases from the PFSF. This satisfies the requirements of 10 CFR 72.126(b) and (c).

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7.4 ESTIMATED ONSITE COLLECTIVE DOSE ASSESSMENT

The shipping, transfer and storage casks are designed to limit dose rates to ALARA levels for operators, inspectors, maintenance, and radiation protection personnel when the canisters are being transferred from the shipping to the storage casks, when the storage casks are being moved to the storage pads, and while the storage casks are being stored on the pads.

Table 7.4-1 shows the estimated occupational exposures to PFSF personnel during receipt of the HI-STAR shipping cask, transfer of the canister from the shipping cask to the HI-STORM storage cask using the HI-TRAC transfer cask, movement of the storage cask to the pad, and emplacement on the pad. Table 7.4-2 shows the estimated occupational exposures to PFSF personnel for these operations involving the TranStor shipping, transfer, and storage systems. The estimated occupational exposures were calculated in Reference 20. The operational sequence for these operations is also described in Chapter 5.

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 and reflect expected personnel exposures. 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, and shown to be conservative in that section, were not applied to estimate worker integrated doses. Evaluation of weighted average burnups and cooling times of the nations' PWR and BWR spent fuel inventory in existence at the end of 1994, as discussed in Section 7.3.3.5, 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 characteristics for spent fuel assumed in the onsite dose assessment are considered to be representative of typical fuel expected to be received at the PFSF.

From Table 7.4-1, the total dose from receipt of a loaded shipping cask, transfer of the canister into a storage cask, movement of the storage cask to the pad, and performance of initial surveillances is estimated to be about 205 person-mrem for both HI-STORM and TranStor systems. Assuming a storage cask loading rate of 200 casks per year, the total annual dose to operations and Radiation Protection personnel involved in these operations is estimated to be approximately 41 person-rem. Occupational doses to individuals will be administratively controlled to ensure that they are maintained below 10 CFR 20.1201 limits and ALARA.

Temporarily positioned shielding will be used during transfer operations to reduce dose rates from streaming paths or relatively high radiation areas where its use will result in a net reduction in worker exposures. The effects of temporarily positioned shielding, calculated in Reference 20, are considered in the Table 7.4-1 and 7.4-2 dose estimates for canister transfer operations.

Occupational exposures are also estimated to security personnel and PFSF personnel that conduct inspections, surveillances, and maintain the storage systems. These estimates are based on the assumption that the PFSF is at its 4,000 storage cask capacity. It is estimated that security personnel that conduct security inspections will accrue approximately 1.3 person-rem annually, based on one inspection per shift (3 shifts per day, 365 days per year) along the RA fence, using the highest dose rate at the fence discussed in Section 7.3.3.5. It is considered that dose rates inside the Security and Health Physics Building are negligible due to shielding provided by the

building structure. One visual inspection per quarter is required to be performed for each storage cask to check for the buildup of debris at the inlet ducts and to inspect the cask exterior. Assuming one person spends 1.0 minute inspecting each cask, in an average dose field of 15 mrem/hr during the inspection, this surveillance will result in approximately 1.0 person-rem per quarter to PFSF personnel conducting the inspections, for a total of 4.0 person-rem annually. The 15 mrem/hr average dose field estimate near a cask inside the cask array is based on the Reference 21 calculation, which assumes that storage casks contain "typical" PFSF fuel, represented by PWR fuel with 35 GWd/MTU burnup and 20 year cooling time. 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 (Reference 21), an additional annual dose of 0.5 person-rem is estimated. Monitoring of temperatures representative of the thermal performance of the casks will be performed remotely with a data acquisition system and will not result in significant exposure. Based on the above, the total dose to personnel involved in security inspections, surveillance, and storage cask maintenance operations is estimated to be 5.8 person-rem annually.

A combination of building location and shielding will minimize the dose to staff personnel working in the PFSF facilities. The west sides of the Canister Transfer Building and Security and Health Physics Building are approximately 425 ft (130 meters) and 948 ft (289 meters), respectively, from the nearest storage pad (see Figure 1.2-1). The building structures will provide shielding to reduce doses to workers in the buildings from the cask storage area to levels that are ALARA. The Operations and Maintenance Building and Administration Building will be located near the entrance gate to the OCA (see Figure 1.1-2). The Administration Building is further from the storage pads (2,580 ft) than the nearest distances to the OCA boundary (2,119 ft), and the Operations and Maintenance Building is nearly as far away (1,960 ft). Dose rates at these buildings will be less than 25 mrem/yr (at a 2,000 hr/yr occupancy rate) without consideration for shielding provided by the building structures.

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16. NUREG-0041, Manual of Respiratory Protection Against Airborne Radioactivity Materials, October 1976.

17. Regulatory Guide 8.26, Application of Bioassay for Fission and Activation Products, U.S. NRC, September 1980.

18. Regulatory Guide 8.9, Acceptable Concepts, Models, Equations and Assumptions for a Bioassay Program, U.S. NRC, September 1973.

19. U.S. Department of Energy, Energy Information Administration's Service Report entitled "Spent Nuclear Fuel Discharges from U.S. Reactors - 1994", published in February 1996.

20. PFSF 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, Revision 1, Stone & Webster.

21. PFSF Calculation No. 05996.02-UR-5, Dose Rate Estimates from Storage Cask Inlet Duct Clearing Operations, Revision 0, Stone & Webster.

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CHAPTER 8

ACCIDENT ANALYSES

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8.1.2 Off-Normal Ambient Temperatures

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

8.1.2.1 Postulated Cause of the Event

In order to bound expected steady-state temperatures of the storage system during periods of abnormally high temperatures, analyses were performed by the storage system vendors to calculate the steady-state temperatures for the storage cask, canister, and fuel for a continuous 100°F ambient condition with solar insolation. The design basis spent fuel decay heat generation rates were used for these analyses. The postulated 100°F ambient condition bounds the design basis average daily maximum temperature of 95°F for the PFSF. Since it would take 4 to 5 days for the storage systems to achieve steady-state thermal conditions, component temperatures resulting from the constant 100°F off-normal event with solar insolation bound those associated with the 95°F average daily maximum temperature condition.

8.1.2.2 Detection of Event

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

8.1.2.3 Analysis of Effects and Consequences

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

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

8.1.2.4 Corrective Actions

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

8.1.3 Partial Blockage of Storage Cask Air Inlet Ducts

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

8.1.3.1 Postulated Cause of the Event

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

8.1.3.2 Detection of Event

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

8.1.3.3 Analysis of Effects and Consequences

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

8.1.3.4 Corrective Actions

Upon receiving indication of high storage cask(s) temperatures, PFSF personnel will inspect the affected cask(s) ducts for blockage. Once an obstruction has been identified, PFSF personnel will remove the debris or other foreign material blocking the ducts. Since screening is provided for all air inlets, material blocking inlet ducts is expected to be on the outside and may be removed by hand or hand-held tools. Dose rates at the air inlets are higher than the nominal dose rates at the storage cask walls, so a worker clearing the vents will be subject to above-normal dose rates. As a worst case estimate, it is assumed that a worker kneeling with hands on the vent inlets requires up to 30 minutes to clear the vents. Assuming the affected cask has the highest dose rates associated with a storage cask containing design fuel (Tables 7.3-1 and 7.3-2), and assuming nearby casks contain PWR spent fuel having the 40 GWd/MTU burnup and 10 years cooling time characteristics discussed in Section 7.3.3.5, 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 nearby casks in the array (Reference 44).

8.1.5.2 Detection of Event

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

8.1.5.3 Analysis of Effects and Consequences

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

For the dose assessment, it is assumed that all of the contamination on the external surfaces of a canister is Co-60. This assumption is justified based on the following: 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 nuclear power plants 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 (Chapter 7, Table 7.1 of NUREG-1536, Reference 24) as being present in the form of crud on fuel rods and is listed as the only nuclide which contributes significantly to doses from the postulated radioactivity 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.

Doses resulting from this postulated release of contamination from the external surfaces of a canister were calculated in Reference 45. Assuming the contamination is Co-60 particulate activity evenly distributed at a concentration of 1 E-4 $\mu Ci/cm^2$ over the entire external surface of a HI-STORM canister (TranStor canister has a smaller area), with a surface area of approximately 312,000 cm^2, there would be a total activity inventory of approximately 31.2 μCi. The nearest distance from a storage pad to the OCA fence (site area boundary) is 646 meters, and the nearest distance from the Canister Transfer Building to the OCA fence is 500 meters. A χ/Q of 1.94 E-3 sec/cubic meter was calculated in accordance with Regulatory Guide 1.145 (Reference 6), assuming a distance of 500 meters to the dosereceptor, a wind speed of 1 meter/sec, atmospheric stability class F, with no consideration for plume meander.

The dose conversion factor for intake of Co-60 is specified in EPA Federal Guidance Report No. 11 (Reference 7) as a committed effective dose equivalent (CEDE) of 5.91 E-8 Sv/Bq, equal to 219 mrem/μCi. The highest dose conversion factor for committed dose equivalent (CDE) to any organ from Co-60 is that for the lungs, 3.45 E-7 Sv/Bq, equal to 1,277 mrem/μCi. An adult breathing rate of 3.3 E-4 cubic meters per second is assumed in accordance with Reference 7. A respirable fraction of 1.0 is assumed. Assuming an individual is located within the plume 500 meters from the release point for the duration of the release, the individual would receive a CEDE of 4.37 E-3 mrem and a CDE to the lungs of 2.55 E-2 mrem. The dose to an individual at the OCA boundary from external exposure to radiation emitted by the plume (submersion dose) was also calculated, using the effective dose conversion factor for Co-60 specified in EPA Federal Guidance Report No. 12 (Reference 30). This dose conversion factor, representative of exposure to a semi-infinite cloud of radioactive material, is 1.26 E-13 Sv/sec per Bq/m^3, equal to 4.66 E-4 mrem/sec per μCi/m^3. The submersion dose from external exposure to Co-60 in the plume was calculated to be 2.82 E-5 mrem. Adding the external dose from submersion to the internal CEDE and CDE to the lungs results in a total effective dose equivalent (TEDE) of 4.40 E-3 mrem, and a total lung dose of 2.55 E-2 mrem. These doses are well below the 10 CFR 72.106(b) criteria of 5 rem TEDE and 50 rem lung dose that apply to accidents. Assuming an off-normal condition resulting in release of contamination to the atmosphere occurs on the order of once per year, total annual dose consequences at the OCA boundary from this event and radiation emanating from storage casks (Section 7.6) will not exceed 25 mrem, in accordance with 10 CFR 72.104.

The dose was also calculated to onsite personnel assumed to be located 150 meters from the release point using the same methodology and assumptions discussed above, with a calculated χ/Q of 1.40 E-2 sec/cubic meter. Onsite personnel 150 meters from the release point would receive a CEDE of 3.15 E-2 mrem, a CDE to the lungs of 1.84

E-1 mrem, and an external dose due to submersion of 2.04 E-4 mrem. Adding the external dose from submersion to the internal CEDE and CDE to the lungs results in a TEDE of 3.17 E-2 mrem and a total lung dose of 1.84 E-1 mrem.

8.1.5.4 Corrective Actions

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

seismic hazard at a nuclear power plant site. In response to the regulatory changes in seismic analysis methodology for siting nuclear power plants, and anticipated changes to Part 72 (SECY-98-126), a PSHA has been performed for the PFSF for vibratory ground motions and surface fault displacement. The seismic design basis for the PFSF has been revised (References 29 and 41), with the current design basis ground motions based on the PSHA, as discussed in Sections 2.6 and 3.2.10. The design basis ground motions are characterized by site specific response spectrum curves having peak ground accelerations of 0.53 g horizontal (two directions) and 0.53 g vertical, as identified in Sections 2.6.4.9 and 3.2.10.1.1.

The site specific cask stability analyses were initially performed based on the PFSF original site specific deterministic design earthquake, which has been superseded by the current design basis ground motion established by the PSHA. These analyses determined that while the casks do rock slightly, they do not tip over, nor does rocking result in collision of storage casks with adjacent casks. In addition, the analyses determined that while the casks could slide, they could not slide off the storage pad, nor would sliding result in collision of storage casks with adjacent casks. The response spectrum of the PFSF original site specific deterministic design earthquake (referred to as the PFSF deterministic design earthquake) bounds that associated with the current design basis ground motion. The initial site specific cask stability analyses are therefore conservative, being based on significantly higher ground accelerations than those of the design basis ground motion. Since the initial cask stability analyses were performed, Holtec (the HI-STORM storage cask vendor) has performed cask stability analyses for both the HI-STORM and TranStor storage casks using the same methodology to analyze the two different storage casks, based on the PSHA design basis ground motion (0.53 g horizontal and 0.53 g vertical). The results of these analyses have been included in the following section.

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

8.2.1.2 Accident Analysis

The HI-STORM and TranStor storage casks are analyzed for a generic design earthquake as selected by each cask vendor and as described in their respective SARs (References 2 and 3). The HI-STORM and TranStor storage casks were also analyzed for the PFSF site specific deterministic design earthquake, represented by response spectrum curves 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 by the respective cask vendors to assure structural strength of the cask and cask stability. Since the response spectra of the PFSF deterministic design earthquake bounds that associated with the current design basis ground motion determined by a PSHA, the initial cask stability analyses for the PFSF site specific deterministic design earthquake provide assurance that the casks will not tip over or slide excessively in an earthquake. More recently, both the HI-STORM and TranStor storage casks were analyzed for the PSHA design basis ground motion (0.53 g horizontal and 0.53 g vertical), as discussed below.

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

excessively in an earthquake and confirms the conclusions of the vendors' analyses of the capability of their storage casks to withstand the PFSF deterministic design earthquake.

A summary of the vendor's cask stability analyses and the independent cask stability analyses performed by J. D. Stevenson, Consulting Engineer, follow. Holtec has completed the analyses of the HI-STORM and TranStor storage casks for the PSHA design basis ground motion (0.53 g horizontal and 0.53 g vertical), applying the same methodology to analyze the two different storage casks. The results of these more recent analysis, which supercede the analyses for the PFSF deterministic design earthquake, are presented below.

HI-STORM Cask Stability Analysis

The HI-STORM generic seismic cask stability analysis is described in Section 3.4.7 of the HI-STORM SAR (Revision 9). The analysis basis is a conservative two-dimensional quasi-static evaluation of incipient tipping or sliding. The seismic input is: (1) a horizontal force, applied at the cask centroid, equal to the loaded cask weight multiplied by the Zero Period Acceleration (ZPA) associated with the resultant of two horizontal seismic events; and (2), a vertical force, applied at the cask centroid, equal to the loaded cask weight multiplied with a ZPA for the vertical earthquake.

The generic analysis determined that inertia loads produced by the seismic event are less than the 45 g loads for which the storage system is designed. Stresses in the canister due to the seismic event are bounded by stresses resulting from the hypothetical end drop and side drop events described in Section 3.4.10 and Appendix 3A of the HI-STORM SAR (Revision 9). Further, as discussed in Appendix 3.B of the HI-STORM SAR, ready retrievability of the MPC is assured under the most severe postulated accident event, hypothetical cask tipover.

The generic cask stability analysis in the HI-STORM SAR for incipient tipping or sliding does not bound the PFSF design basis ground motion. In order to demonstrate the cask stability under site specific conditions, site specific cask stability analyses have been performed by the cask vendor. Results of the initial HI-STORM cask stability analysis for the PFSF deterministic design earthquake are documented in Reference 8. Holtec has also performed a cask stability analysis for the PSHA design basis ground motion (Reference 42), described below.

The HI-STORM storage cask was analyzed using proprietary qualified software for the PFSF design basis ground motion characterized by response curves with a zero period acceleration of 0.53g in both horizontal directions and 0.53g in the vertical direction. The analysis considered soil-structure interaction, actual storage pad size, and a variety of cask placements on the pad.

The site specific cask stability analysis was performed by developing three statistically independent acceleration time histories from the site specific response spectra, generated from the PSHA. This seismic input was applied three-dimensionally to the structural system model, which included the storage pad, soil springs, and various cask placements to determine the worst case response. The site specific seismic analysis employs a mass-spring representation of the cask behavior and boundary conditions, and a numerical integration of the dynamic equations.

Each cask is modeled as a two body system with each overpack described by six degrees of freedom to capture the inertial rigid body motion of the overpack. Within each overpack the internal MPC is modeled by an additional five degrees of freedom which are sufficient to define all but the rotational motion of the MPC about its own longitudinal axis, a motion which is of no significance in this analysis. Compression-only spring constants are developed to simulate the contact stiffness between the MPC

and the overpack cavity. Interface spring constants are developed for the overpack-to-concrete pad linear compression only contact springs and for the associated friction springs at each of the 36 contact locations for each overpack on the pad.

Soil-structure interaction is incorporated into the model by the development of soil springs to reflect the characteristics of the underlying soil mass beneath the pad. Horizontal, vertical, rocking and torsional spring rates were calculated along with appropriate soil mass and damping values and applied at the pad-soil interface. The sensitivity of the cask response to upper and lower bounds of soil-spring interaction was studied and determined not to have a significant effect on cask displacements.
The cask stability analysis was performed by computer methods using cask-to-pad coefficients of friction equal to 0.2 (which emphasizes sliding potential) and 0.8 (which emphasizes tipping potential) to bound the maximum sliding and tipping behavior of the cask. The results of the site-specific analysis show that the storage casks will not tip over or slide to the extent of impacting adjacent casks during the PFSF design basis ground motion.

For the limiting case with a 0.8 coefficient of friction (maximum tip), there is minimal rotation of the cask vertical centerline. The maximum excursion of the top of the cask during rocking, identified as the lateral motion of the cask top center point from its initial position, is less than 4 inches for any of the configurations. For the limiting case bounded by a 0.2 coefficient of friction (maximum slide), the maximum distance in which a cask will slide is shown to be less than 3 inches. For both coefficients of friction considered, cask motions are generally in-phase with each other. The casks are spaced on the storage pad at 15 ft center-to-center which provides 47.5 inches clear between casks (cask diameter is 132.5 inches) and provides a considerable margin of safety against impacts between casks during a seismic event.

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

TranStor Cask Stability Analysis

The TranStor generic cask stability analysis is described in Section 11.2.5 of the TranStor SAR (Revision C). The analysis demonstrates the storage cask is stable and will not begin to tip when subjected to a seismic event characterized by Regulatory Guide 1.60 response spectra curves with a zero period acceleration of 0.38 g in two horizontal directions and 0.25 g in the vertical direction. The three components of the earthquake motion are combined using the 100-40-40 rule defined in NUREG/CR-0098 (Reference 9). The analysis also concludes that with a maximum ground displacement of 0.45 inches (substantially less than 44 inches of clear space between the casks), sliding would not cause impact between adjacent casks.

The generic cask stability analysis in the TranStor SAR assumes a single cask resting on a rigid surface. The analysis does not consider soil-structure interaction, which will affect the dynamic properties and seismic response of the structural system. In order to verify the cask stability under actual conditions, a site specific cask stability analysis was performed by Sierra Nuclear Corporation, the TranStor cask vendor, for the PFSF deterministic design earthquake characterized by site specific response curves with zero period accelerations of 0.67 g in both horizontal directions and 0.69 g in the vertical direction. The analysis considered soil-structure interaction, actual storage pad size, and a variety of cask placements on the pad. Results of the TranStor cask

stability analysis for the PFSF deterministic design earthquake are documented in References 53 and 54.

In addition, Holtec (vendor of the HI-STORM storage cask) has performed a cask stability analysis of the TranStor storage cask for the PSHA design basis ground motion (Reference 55), applying the same methodology used in the HI-STORM storage cask stability analysis and accounting for TranStor storage cask weights and geometry. The TranStor storage cask was analyzed using proprietary qualified software for the PFSF design basis ground motion characterized by response curves with a zero period acceleration of 0.53g in both horizontal directions and 0.53g in the vertical direction. The analysis considered soil-structure interaction, actual storage pad size, and a variety of cask placements on the pad.

Soil-structure interaction was incorporated into the model by the development of soil springs to reflect the characteristics of the underlying soil mass beneath the pad. Horizontal, vertical, rocking and torsional spring rates were calculated along with appropriate soil mass and damping values and applied at the pad-soil interface. The sensitivity of the cask response to upper and lower bounds of soil-spring interaction was studied and determined not to have a significant effect on cask displacements.

For the limiting case with a 0.8 coefficient of friction (maximum tip), there is minimal rotation of the cask vertical centerline. The maximum excursion of the top of the TranStor cask during rocking, identified as the lateral motion of the cask top center point from its initial position, is less than 1.0 inch for any of the configurations. For the limiting case bounded by a 0.2 coefficient of friction (maximum slide), the maximum distance in which a TranStor cask will slide is shown to be less than 2.25 inches. For both coefficients of friction considered, cask motions are generally in-phase with each other. The casks are spaced on the storage pad at 15 ft center-to-center which

provides 44 inches clear between casks (cask diameter is 136 inches) and provides a considerable margin of safety against impacts between casks during a seismic event.

The site specific cask stability analysis demonstrates that the TranStor storage cask will not tip over in a seismic event. The calculated cask movements are much less than the cask spacing on the storage pad and as such, the storage casks are shown not to impact one another or move off of the storage pad in a seismic event.

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

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

The seismic analyses show that the storage cask will not tip over and no damage will be sustained by either the canisters, their internals, or the storage casks in the event of an earthquake. Therefore, no radioactive material would be released from the storage system when subjected to the PFSF deterministic design earthquake. The TranStor

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storage system thus meets the general design criteria of 10CFR 72.122(b), as it relates to earthquakes.

Independent Cask Stability Analysis

An independent cask stability analysis was performed by J. D. Stevenson, Consulting Engineer, for the purpose of confirming the conclusions of the vendors' site specific cask stability analyses. The analysis considered both the HI-STORM and TranStor storage casks to determine a controlling and bounding storage cask configuration. Although both storage casks are similar in overall dimensions and weight, the cask which was selected as bounding for evaluation of seismic stability was the HI-STORM MPC-32 canister. The HI-STORM MPC-32 canister was the heaviest and the tallest loaded canister and storage cask combination with a weight of 356,521 lb. and a center of gravity of 118 in. above the base of the cask (HI-STORM SAR, Revision 1, Tables 3.2.1 and 3.2.3).

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

The independent cask stability analysis utilized the PFSF deterministic design earthquake response spectra curves, having a zero period acceleration of 0.67 g horizontal (two directions) and 0.69 g vertical. The free field ground surface response spectra were used to develop 3 independent synthetic time histories using the SPECTRA (Reference 14) computer program. These time histories were used as input to the SUPER SASSI/PC computer analysis to evaluate the soil-structure interaction.

conservative estimate of the dose rate one meter from the damaged area is 150 mrem/hr, and the total dose to repair the cask is estimated to be 150 person-mrem.

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 3/4 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 transfer of the canister from one storage cask to another would be similar to that presented in Table 7.4-1, Estimated Personnel Exposures For HI-STORM Canister Transfer Operations, 198.7 person-mrem.

8.2.3 Flood

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

8.2.3.1 Cause of Accident

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

8.2.3.2 Accident Analysis

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

8.2.3.3 Accident Dose Calculations

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

8.2.4 Explosion

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

8.2.4.1 Cause of Accident

Potential for Offsite Explosions

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

The northern perimeter of the Dugway Proving Grounds is approximately 9 miles from the PFSF and the Tooele Army Depot (south area) is approximately 21 miles from the PFSF. There is no interstate highway, railroad (other than the rail which may be installed specifically for shipments of spent fuel shipping casks to and from the PFSF), or river traffic within the vicinity of the PFSF. The nearest interstate highway and commercial rail line are about 24 miles to the north of the facility. The Skull Valley Road, which runs north and south through the Skull Valley Indian Reservation to the

east of the PFSF and provides entrance to the site access road, is 1.9 miles from the Canister Transfer Building and 2.0 miles from the nearest storage pad. The worst-case explosion potential at the PFSF is considered to be from an accident associated with the transportation of explosives along the Skull Valley Road (elevation approximately 4,580 ft, with no obstacles intervening between PFSF).

Potential for Onsite Explosions

A diesel fuel oil storage tank will be located inside the RA, and will supply diesel fuel oil for onsite vehicles, including the cask transporter. This tank will be located near the RA fence, approximately 200 ft northeast of the northeast corner of the Canister Transfer Building and approximately 700 ft from the nearest storage casks. A double-wall subbase diesel fuel oil tank will be mounted on the backup diesel generator skid in the Security and Health Physics Building to provide fuel for operation of the backup diesel generator. This area will be protected with a fire suppression system designed to NFPA 13 requirements for water sprinklers. A fire involving the indoor tank will not affect structures, systems or components outside of the Security and Health Physics Building. The outdoor tank will be above-ground, and will be designed in accordance with the requirements of NFPA 30, with dikes around the tank to contain fuel in the event of leaks or spillage.

While unlikely, it is considered possible that collision or tornado-driven missile impact with the outdoor tank could result in tank rupture and spillage of diesel fuel oil. If there were an ignition source at the location of the spilled diesel fuel, it would be possible to initiate a fire, though diesel fuel is difficult to ignite due to its low volatility. Rupture of a storage tank and spillage of diesel fuel does not create the potential for an explosion. It is planned to use Grade Low Sulfur No. 2-D diesel fuel oil in both applications (onsite vehicles and backup diesel generator), which has a flash point of 126°F (52°C) per Reference 48. Diesel fuel is not a flammable liquid (defined as a liquid having a flash point below 100°F), but falls into the classification of a Class II combustible liquid which

has a flash point above 100°F and below 140°F (Reference 49). The flash point is defined as the lowest temperature at which the vapor pressure of the liquid is just sufficient to produce a flammable mixture at the lower limit of flammability above the surface of the liquid. In recognition of the relatively high flash point of diesel fuel oil (at above-ambient temperatures), NFPA 30 does not require use of explosion proof electrical equipment in the vicinity of diesel fuel oil. While spilled diesel fuel could burn it could not detonate, and therefore an explosion associated with diesel fuel oil is not considered to be a credible event. The outdoor diesel fuel oil storage tank is sufficiently removed from the Canister Transfer Building and the storage casks (nearest important-to-safety structures, systems, and components) that radiant heat energy from a diesel fuel oil fire at the storage tank would not result in damage.

Propane for heating the Canister Transfer Building and the Security and Health Physics Building is stored in two 1,000 gallon propane fuel storage tanks, located outside of the RA, approximately 400 ft east of the Canister Transfer Building and 1,030 ft from the nearest storage casks. The storage tanks will be above-ground, designed in accordance with the requirements of NFPA 58. Propane is stored as a liquefied petroleum gas with the tank pressurized to the vapor pressure of the propane liquid, whose temperature will be close to the average ambient daily temperature. The vapor pressure of commercial propane is 132 psig at 70°F and 216 psig at 105°F (Table 5-5E of Reference 49). Relief valves on the tank will be set at approximately 275 psig. Propane is classified as a flammable liquid, and at standard atmospheric pressure (14.7 psia) commercial propane has a boiling point of minus 51°F (Table 5-5E of Reference 49). It is heavier than air, with propane vapor having a specific gravity of 1.52 at 60°F (Table 5-5E of Reference 49, with specific gravity air = 1). NFPA 58 requires that propane tanks between 50 and 2,000 gallon capacity be located at least 25 ft away from any building, adjacent container, or adjacent property.

8.2.4.2 Accident Analysis

Offsite Explosions

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

The effects of explosions on the storage systems are discussed in the HI-STORM and TranStor SARs, and it is determined that the canisters are protected from the effects of explosions. Overpressures of substantially greater than 1 psi would be required to cause damage to the cask storage systems. The Canister Transfer Building is designed to withstand extreme winds, pressure drops of 1.5 psi, and missiles associated with the design tornado. The effects of credible explosions occurring on the Skull Valley Road, with resultant overpressures less than 1 psi at the PFSF, would not challenge the Canister Transfer Building's structural integrity. Therefore, the canister storage and transfer systems meet the general design criteria of 10 CFR 72.122(c), as it applies to

explosion, which states that structures, systems, and components Important to Safety must be designed and located so that they can continue to perform their safety functions effectively under credible fire and explosion exposure conditions.

Onsite Explosions

It is conservatively assumed that one of the propane tanks contains 1,000 gallons of liquefied propane and that it ruptures. At 60F, one gallon of propane liquid weighs 4.24 lbs (Table 5-5E of Reference 49). The total weight of propane is (1,000 gal) (4.24 lb/gal) = 4,240 lbs. It is also conservatively assumed that a large fraction of this propane mixes with air so that it is in an explosive concentration (in range of 2.15% to 9.60%, per Table 5-5E of Reference 49), ignites, and is involved in an explosion. The magnitude of the postulated explosion was assessed in Reference 50 using the TNT energy equivalent methodology. The TNT energy equivalence of 4,240 lbs of propane is estimated as follows:

Based on Table 5-5E of Reference 49, the total heating value of commercial propane after vaporization is 21,591 Btu/lb. 4,240 lbs of propane has a total heating value of 9.155 E7 Btu, equal to 2.31 E10 calories. Regulatory Guide 1.91 indicates that investigations led to estimates that less than one percent of the calorific energy of hydrocarbon gas/air vapor clouds that exploded was released in blast effects. It is conservatively assumed that 25% of the vapor is in a flammable gas-air mixture having concentrations ranging from the lower flammable limit of 2.15% to the upper flammable limit of 9.60% (Table 5-5E of Reference 49) and that 10% of the total heat of combustion of this flammable mixture is released in blast effects.

Energy Released in Blast = (2.31 E10 cal) (0.25) (0.10) = 5.78 E8 cal.

Trinitrotoluene (TNT) has a "heat of explosion" of 1,050 cal/g (Reference 51). The equivalent weight of TNT that would release 5.78 E8 calories of heat energy is:

(5.78 E8 calories) / (1.05 E3 cal/g) = 5.505 E5 g = 1,214 lbs

The overpressure effects of postulated detonation of this weight of TNT can be assessed using Figure 4-12 of Reference 52, "Shock-Wave Parameters for Hemispherical TNT Surface Explosion at Sea Level". This Reference 52 Army Technical Manual on Explosion Effects is Ref. 1 of Reg. Guide 1.91, and provides the basis for Figure 1 of the Reg Guide. Figure 4-12 of Reference 52 presents overpressures at various scaled ground distances from TNT detonations, with varying weights of TNT. While the storage casks can withstand a much higher overpressure before they begin to slide or tip, the Canister Storage Building is designed to withstand a pressure differential of 1.5 psi due to a tornado (Sections 3.2.8.1 and 3.2.8.3) and an even higher load due to a seismic event. Therefore, the limiting overpressure for important-to-safety structures that could be impacted by a propane explosion is considered to be 1.5 psi. Reference 50 calculated that for an explosion involving 1,214 lbs of TNT, an overpressure of 1.5 psi will occur at a distance of 341 ft from the explosion.

Thus, based on the TNT energy equivalence approach and Reference 5, the resulting overpressure from a propane explosion will not exceed 1.5 psi at important-to-safety structures as long as the propane tank is located a distance of at least 341 ft from the Canister Transfer Building and storage casks. The propane tanks will be sited at a distance of approximately 400 ft east of the Canister Transfer Building, which locates them approximately 1,030 ft from the nearest storage casks. This assures that postulated explosion of propane leaked from a tank will not produce overpressures greater than 1.5 psi and will not challenge the integrity of the storage casks or the Canister Transfer Building.

8.2.4.3 Accident Dose Calculations

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

8.2.5 Fire

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

8.2.5.1 Cause of Accident

The only combustible material at the PFSF storage pads during storage operations is insulation on the temperature monitoring instrumentation wiring, which is present in insignificant quantities at each storage cask. No combustible or explosive materials are allowed to be stored on or near the storage pads. The PFSF Restricted Area (RA) is cleared of vegetation and the entire RA surfaced with compacted gravel. The concrete pads and storage casks are located a minimum distance of 150 ft from the outer edge of the RA (i.e., the inner fence surrounding the RA); the Canister Transfer Building is located by a minimum distance of 112 ft from the outer edge of the RA. The area between the outer edge of the RA and the outer edge of the perimeter road (50 ft distance, see Figure 1.2-1) is also covered with crushed rock. The only significant sources of combustibles that would be present inside the RA would be: 1) the diesel fuel in the tanks of any heavy haul trucks transporting shipping casks to/from the PFSF site; 2) the diesel fuel in the tanks of any train locomotive transporting shipping casks to/from the PFSF site; 3) the diesel fuel in the cask transporter vehicle that would move casks from the Canister Transfer Building to the storage pads; 4) the diesel generator fuel tank inside the Security and Health Physics Building; and 5) the diesel fuel storage tank, which would be located at least 50 ft inside the inner fence surrounding the RA, approximately 200 ft northeast of the Canister Transfer Building and 700 ft east of the nearest storage casks. The effects of wildfires in the vicinity of the PFSF and the effects of fires involving combustibles and transient combustibles located in the RA are evaluated below.

Wildfires

A discussion of the annual probability of wildfires in Skull Valley, as well as range fire magnitudes, duration, propagation and heat generation, is included in Reference 40. The crushed rock surface of the RA and of the contiguous area out to the outer edge of the perimeter road provides a fire break of at least 200 ft to the concrete pads, where the storage casks are located, and a fire break of 162 ft to the Canister Transfer Building. In addition, the spent fuel, equipment, and the PFSF personnel inside the RA will be protected from wildfires by a barrier of crested wheat grass that PFS will plant around the RA. The barrier will be 300 ft wide and will run outward from the outer edge of the perimeter road around the RA. A barrier of crested wheat grass would remain in place with little maintenance after it is planted. Crested wheat grass is fire resistant and thus would eliminate or greatly reduce the effect of any wildfire approaching the PFSF. Because of the distance that would separate a wildfire from the Canister Transfer Building and the and the casks containing spent fuel at the PFSF, a wildfire would pose no direct threat to the spent fuel casks or the SSCs important to safety in the Canister Transfer Building. The magnitude and duration of temperatures resulting from a wildfire at both the storage pads, and the storage casks located there, and at and within the Canister Transfer Building would be far less than those of the design basis fire, discussed below, for which the casks are designed to withstand (Reference 40).

Furthermore, a wildfire could not cause a fire or explosion on site that would threaten the spent fuel casks or SSCs important to safety. The location of the diesel fuel storage tank, at least 50 ft inside the inner fence around the RA, provides a 100 ft firebreak between the outer edge of the perimeter road and the tank, with the crested wheat grass barrier providing an additional 300 ft between a wildfire and the storage tank. At that distance a wildfire would not ignite or explode the diesel fuel in the tank. The diesel emergency generator tank will be a double-walled tank located inside the Security and Health Physics Building, which has reinforced concrete masonry construction, located 50 ft inside the crested wheat barrier, or 350 ft from a wildfire. A

wildfire would not ignite or explode the fuel in the diesel emergency generator tank. All other diesel fuel sources would be farther than 100 ft inside the edge of the crested wheat grass barrier, and would similarly not be threatened by a wildfire due to their distance from a fire, even if it were assumed the wildfire somehow penetrated this grass barrier.

A wildfire in the vicinity of the PFSF would not cause the evacuation of PFSF security personnel. By virtue of the 300 ft crested wheat grass barrier surrounding the PFSF RA and the distance between the outer edge of the perimeter road around the RA, the heat from a wildfire would not pose a threat to any personnel inside the RA. PFSF security personnel will have appropriate emergency breathing apparatus available such that the smoke from a wildfire near the PFSF will not force them to evacuate.

Combustion Sources Inside the Restricted Area

Movement of a storage cask from the Canister Transfer Building to a storage pad involves the use of a diesel-powered cask transporter, whose fuel tank has a capacity of 50 gallons of diesel fuel. The worst-case fire at the storage pads involves a postulated spill and ignition of this diesel fuel in the vicinity of a storage cask. The accident scenario involving a storage cask in the following section assumes that the fuel tank of the transporter vehicle ruptures, resulting in 50 gallons of diesel fuel spilled, which is postulated to ignite and burn.

The combustibles of key concern in the Canister Transfer Building are the transient combustibles associated with the diesel fuel tanks of the cask transporter and the heavy haul vehicle tractor. For rail delivery/retrieval of shipping casks, the train locomotives are required by administrative procedure to stay out of the Canister Transfer Building. The design of the building and its surroundings will assure that any diesel fuel spilled outside the building will not flow into the building, which could create a fire hazard. The heavy haul vehicle tractors have saddle tanks with a total capacity of up to 300 gallons of diesel fuel. Spillage of diesel fuel does not create the potential for

explosions in the Canister Transfer Building, due to this fuel's low volatility. It should be noted that diesel fuel is difficult to ignite, and it is highly unlikely that spillage of diesel fuel would result in a fire. The following assumes that spillage of diesel fuel is somehow ignited, and considers fires in the Canister Transfer Building associated with postulated rupture of the cask transporter's fuel tanks, with up to 50 gallons of fuel spilled in a transfer cell, and postulated rupture of a heavy haul tractor's fuel tanks, with up to 300 gallons of diesel fuel spilled in the cask load/unload bay.

8.2.5.2 Accident Analysis

Storage System

A fire is assumed to occur when the fuel tank of the cask transporter ruptures spilling diesel fuel in the vicinity of a storage cask that is at its location on a storage pad, or enroute from the Canister Transfer Building to its storage location, and the diesel fuel is postulated to ignite and burn. This scenario is analyzed in Section 11.2.4 of the HI-STORM SAR. From IAEA requirements (Reference 18), the "pool" of fuel is assumed to completely encircle a storage cask and extend 1 meter beyond the cask surface. Based on the minimum outer cask diameter of 132.5 inches (HI-STORM), this spill would result in a ring of fuel with a pool surface of about 147.6 sq ft around the storage cask. A fuel consumption rate of 0.15 in/min was assumed (Reference 19) based on gasoline/tractor kerosene experimental burning rates. This translates into a fuel consumption rate of approximately 14 gal/min. Therefore, the 50 gallons of fuel would sustain a fire for about 3.6 minutes.

The storage system designs are highly resistant to the effects of fires. The thick concrete walls are not significantly affected by short-term exposure to fire induced temperatures, and the thermal diffusivity is such that any fire would be required to burn for many hours before much of the wall thickness would be affected. HI-STORM SAR Section 11.2.4 describes the results of a transient analysis of the effects of a diesel fuel fire encircling a storage cask assumed to burn for 3.6 minutes. The analysis concludes

that because of the comparatively short fire duration and the thermal inertia of the cask, the effect of a fire accident on the canister temperature is negligible. The ability of the HI-STORM system to cool the spent fuel within design temperature limits during post-fire equilibrium is not compromised. Intense heat from the fire only partially penetrates the storage cask wall, and the majority of concrete experiences a relatively minor temperature increase. Less than an inch of the concrete (less than 4% of the total overpack radial concrete section) exceeds the short term temperature limit. Concrete exposed to extreme temperatures would experience some reduction of its neutron shielding capability, but because of the small amount of concrete exposed to high temperature this would not significantly increase the dose rate from the cask. Based on this analysis, the effects of a fire of approximately 3.6 minutes duration on the storage system will have a negligible effect on canister and fuel temperatures and cause no reduction in nuclear safety. In fact, the HI-STORM storage cask has been analyzed to show that it can withstand a diesel fire, caused by a 200 gallon diesel spill, of 1475° F for 15 minutes duration without threatening the integrity of the canister or fuel rod cladding.

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

A locomotive fuel spill and associated fire at the PFSF is extremely unlikely given the low speeds at which the locomotives will operate at the PFSF and the difficulty of igniting spilled diesel fuel. Nevertheless, it could be postulated that the fuel tank(s) of a

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

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

8.2.7 Canister Leakage Under Hypothetical Accident Conditions

The leakage of a canister under hypothetical accident conditions wherein cladding of 100% of the fuel rods is postulated to have ruptured is classified as Design Event IV as defined by ANSI/ANS-57.9. This is not a credible accident at the PFSF.

8.2.7.1 Cause of Accident

The HI-STORM and TranStor canisters are totally sealed, integrally welded pressure vessels, designed to Section III of the ASME BPVC. There are no gaskets, mechanical seals, or packing that could provide a potential leakage path for the radioactive fission products contained within the fuel cladding. The canisters are provided with multiple closures to confine the radioactive fuel. Following welding of the closures, the canisters are tested to verify their leaktight integrity. No components are required to penetrate the sealed canisters after helium backfilling is completed and the outer closure is welded in place. The postulated failure of the cladding of all fuel rods in a canister and release of gases normally contained in the fuel rod cladding under pressure would not challenge the integrity of the canisters (Section 8.2.10). Maximum canister leakage under conditions wherein cladding of 100% of the fuel rods is postulated to have ruptured is considered to be a non-credible event, which will not occur over the life of the PFSF. Nevertheless, this accident is hypothesized and analyzed below. Doses resulting from the canister leakage under hypothetical accident conditions were calculated in accordance with Interim Staff Guidance-5 (ISG-5, Reference 31).

8.2.7.2 Accident Analysis

In this accident analysis, it is postulated that a canister leaks at the maximum rate permitted by the closure helium leakage test acceptance criteria. Such a leak would require a significant defect in each of two redundant closure welds. In this hypothetical

Nuclides	Release Fractions
Gases (Includes H-3, Kr-85, I-129)	0.30
Crud (Includes Co-60)	1.0
Volatiles (Includes Sr-90, Ru-106, Cs-134, Cs-137)	2.0 E-4
Fuel Fines (Includes Y-90, Sb-125, Te-125m, Ce-144, Pr-144, Pm-147, Eu-154, Eu-155, Pu-238, Pu-239, Pu-240, Pu-241, Am-241, Cm-244)	3.0 E-5

The activity inventory of each radionuclide released from the fuel into the canister was calculated by multiplying the total activity of the radionuclide associated with 61 BWR fuel assemblies times the above release fraction. For conservatism no credit was taken for holdup of particulates and volatiles released from the fuel inside the canister.[1] Therefore, 100 percent of these radionuclides and 100 percent of the H-3, Kr-85 and I-129 are assumed to be available for release from the canister. The release rate for each radionuclide was calculated by taking the TranStor canister gaseous leak rate under the hypothetical accident conditions (1.0E-4 cc/sec) divided by the minimum free volume of a TranStor canister loaded with BWR fuel assemblies (accounting for the fuel rod plenum volumes), multiplied by the activity of each radionuclide available for release.

[1] Based on Table XIX of Reference 25, 90 percent of particulate and volatile fission products would be subject to plateout or deposition within the leaking canister following release from the fuel rods, and would not be available for release to the atmosphere. However, for conservatism no credit has been taken for holdup of particulates and volatiles released from the fuel inside the canister.

8.2.7.3 Accident Dose Calculations

Doses resulting from the postulated leaking canister were calculated in Reference 46. The nearest distance from a PFSF storage pad to the OCA fence (site area boundary) is 646 meters, and the nearest distance from the Canister Transfer Building to the OCA fence is 500 meters. A χ/Q of 1.94 E-3 sec/cubic meter was calculated in accordance with Regulatory Guide 1.145 (Reference 6), assuming a distance of 500 meters from the release source to the dose receptor, a wind speed of 1 meter/sec, and atmospheric stability class F, with no consideration for plume meander.

The dose conversion factors for internal doses due to inhalation, the Committed Effective Dose Equivalent (CEDE) and Committed Dose Equivalent (CDE) to organs, were obtained from the EPA Federal Guidance Report No. 11 (Reference 7). An adult breathing rate of 3.3 E-4 cubic meters per second was assumed (Reference 7). For conservatism no credit was taken for a respirable fraction, and internal doses were calculated assuming that 100% of radionuclides released from the leaking canister are of respirable size.[2] In addition to internal doses, doses due to external radiation from submersion in the plume (deep dose equivalent) were also evaluated in Reference 46. Dose conversion factors for submersion were obtained from EPA Federal Guidance Report No. 12 (Reference 30). In accordance with ISG-5, a canister leakage duration of 30 days was assumed for this hypothetical accident condition. Dose calculations were conservatively based on the assumption that an individual is continuously present at the location nearest the canister transfer building on the OCA boundary for the 30 day leakage duration, and the wind constantly blows in this direction for 30 days.

[2] Based on Table XX of Reference 25, 95 percent of particulates released from inside the fuel rod due to cladding breach are greater than 10 microns aerodynamic diameter and are non-respirable. However, for conservatism no such credit has been taken and the respirable fraction is assumed to be 1.

Calculation determined a CEDE of 75.7 mrem due to inhalation and an external dose due to submersion in the plume of 0.155 mrem, for a Total Effective Dose Equivalent (TEDE) of 75.7 + 0.155 = 75.9 mrem.[3] The maximum organ dose is the CDE to the bone surface plus the submersion dose, calculated to be 824 + 0.155 = 824 mrem. The skin dose was calculated to be 0.28 mrem, which serves as a reasonable approximation of the dose to the lens of the eye. 10 CFR 72.106(b) requires that any individual located on or beyond the nearest boundary of the controlled area may not receive from any design basis accident the more limiting of a total effective dose equivalent of 5 rem, or the sum of the deep-dose equivalent and the committed dose equivalent to any individual organ or tissue (other than the lens of the eye) of 50 rem. The lens dose equivalent shall not exceed 15 rem and the shallow dose equivalent to skin or to any extremity shall not exceed 50 rem. Based on the above TEDE and organ doses, the bounding leaking canister accident, involving maximum leakage of a TranStor canister containing failed BWR fuel, does not exceed the limits specified in 10 CFR 72.106(b). Note that although the consequences have been evaluated, this is not considered to be a credible event for the PFSF.

[3] Although no such credit has been taken, if credit were taken for a 10% canister release fraction of volatiles and particulates (with 90% of volatiles and particulates released from the fuel rods assumed to be held up in the canister by plateout/deposition and unavailable for release), based on Table XIX of Reference 25, and if credit were also taken for the 5% respirable fraction of particulate fission products released from inside the fuel rods in accordance with Table XX of Reference 25, then the TEDE would be 2.70 mrem to the individual at the owner controlled area fence. The CDE to the maximally exposed organ, the lungs in this case, would be 14.7 mrem.

As an evaluation of the potential doses from environmental pathways following deposition of material in the plume, a pathway analysis using the RESRAD computer code (Reference 36) was next conducted (Reference 47). The first step of this evaluation was to estimate the amount of material deposited on the ground from the plume. This estimate was made assuming that the effluent concentration in a given sector is uniform across the sector at a given distance, as described in Regulatory Guide 1.111 (Reference 37).

Using a straight-line trajectory model, this approach requires that the relative deposition rate should be divided by the arc length of the sector at the given downwind distance being considered to estimate deposition. The value of relative deposition (m^{-1}) was obtained from Figure 6 of Regulatory Guide 1.111, with the resulting value of 8.0 E-5 m^{-1} at 500 meters downwind. Deposition estimates were made for each of the radionuclides in the source term. These values, in units of pCi/m^2, were next modified to units of pCi/g to match the input requirements of the RESRAD code, by assuming a soil density of 1.5 E+6 g/m^3 and uniform contamination of the soil to a depth of 1 cm.

The exposure scenario considered in the RESRAD analysis includes direct exposure to contaminated ground, inhalation of resuspended radioactive material, ingestion of milk and beef following grazing, and ingestion of soil. This scenario is considered to be a conservative representation of the land use conditions and environment of the land surrounding the PFSF. 2,000 hours/year occupancy time was assumed at the 500 meter distance along the owner controlled area fence. Although natural vegetation is quite sparse, it is conservatively assumed that the RESRAD default values for fodder intake are met both for the dairy and beef cattle. Default values for human consumption provided in RESRAD for air, milk, beef, and soil were assumed (with the inhalation value reduced from the default value by a factor of 0.228 (2000 hrs / 8760 hrs) to account for partial occupancy). The default values include inhalation of 1,918 m^3 of air with a mass loading factor for air of 2.0 E-4 g/m^3, ingestion of 92 liters of milk,

ingestion of 63 kg of beef, and ingestion of 36.5 g soil. The resulting TEDE for the accident case was 2.70 mrem/yr at 500 meters downwind. This dose is a small fraction of the inhalation plus submersion doses identified above, and well below the 5 rem TEDE accident limit imposed by 10 CFR 72.106(b). The dominant exposure pathway was determined to be external exposure to contaminated land and the radionuclide with the largest contribution to the dose was Co-60.

8.2.7.4 Recovery Plan for a Hypothetical Canister Breach

This section has been removed in accordance with the NRC's Interim Staff Guidance-3 (ISG-3, Reference 38), which indicates that recovery from non-mechanistic failures of the confinement boundary by such means as over-packs or dry transfer systems would not be considered and evaluated in the licensing process.

8.2.8 100% Blockage of Air Inlet Ducts

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

8.2.8.1 Cause of Accident

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

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

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

8.2.9 Lightning

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

8.2.9.1 Cause of Accident

This event would be caused by meteorological conditions at the site. Lightning 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 belimited 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 storage 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 (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.

8.3 SITE CHARACTERISTICS AFFECTING SAFETY ANALYSIS

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

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

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

Information associated with aircraft flights in the vicinity of the PFSF, presented in Section 2.2 of this SAR, is based on data obtained from the U.S. Air Force, the Dugway Proving Ground, the Department of Energy (DOE), the Federal Aviation Administration (FAA), the National Transportation Safety Board, and the National Oceanic and Atmospheric Administration (NOAA). As discussed in Section 2.2, the probability of an aircraft impacting the PFSF is below applicable NRC regulatory standards and guidance and therefore is not considered to be a credible event. As also discussed in

Section 2.2, other activities associated with military and industrial facilities and military ranges in the vicinity of the PFSF pose no credible hazard to the facility.

8.4 REFERENCES

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