STONE & WEBSTER ENGINEERING CORPORATION

CALCULATION TITLE PAGE

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Private Fuel Storage Limited Liability Corporation / Private Fuel Storage Facility	
CALCULATION TITLE **Dose Rate Estimates from Storage Cask Inlet Duct Clearing Operations**	QA CATEGORY III

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Dose Rate Estimates from Storage Cask Inlet Duct Clearing Operations	III

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(Revisions, Additions, Deletions, Etc.)

CALCULATION OBJECTIVE

The objective of this calculation is to support the estimated dose rates and integrated doses documented in the PFSF SAR (Reference 1) to PFSF workers who are assumed to clear debris that is blocking or partially blocking storage cask inlet ducts.

BACKGROUND / HISTORICAL INFORMATION

Section 8.1.3 of the PFSF SAR, "Partial Blockage of Storage Cask Air Inlet Ducts", indicates that a worker postulated to spend one-half hour clearing inlet ducts for a cask with 50% of its inlet ducts blocked would accrue approximately 35 mrem to the hands and forearms and approximately 25 mrem to the chest and body from the cask with blockage and from adjacent casks. Section 8.2.8 of the PFSF SAR, "100% Blockage of Air Inlet Ducts", indicates that workers clearing inlet ducts would accrue double the doses estimated in Section 8.1.3, characterizing this as "approximately 70 person-mrem".

PFSF SAR Section 7.4, "Estimated Onsite Collective Dose Assessment", states "Conservatively assuming that 5 percent of the 4,000 casks require clearing of debris from the inlet ducts once a year at 10 minutes each, in a dose field of 15 mrem/hr, an additional annual dose of 0.5 person-rem is estimated." This Section 7.4 estimate was performed for different assumed characteristics (lower burnup and longer cooling time) of the spent fuel than the Chapter 8 accident analysis estimates.

For the purpose of estimating worst case dose rates and integrated doses for the PFSF accident analysis, the Chapter 8 cases assume that the storage cask whose inlet ducts are being cleared contains design basis fuel and nearby casks contain spent fuel with above average source strength. The Chapter 7 case assumes that the storage cask whose inlet ducts are being cleared and surrounding casks all contain "typical" spent fuel, whose characteristics are discussed below, for the purpose of estimating realistic or expected dose rates and personnel doses for use in the PFSF annual exposure estimate.

CALCULATION METHOD / ASSUMPTIONS

Section 5 of the HI-STORM and TranStor Storage Cask SARs (References 2 and 3) identify conservative dose rates calculated on the surfaces and at 1 meter from single storage casks, assuming the casks contain canisters loaded with each vendor's design basis fuel, but do not identify dose rates at other relatively short distances representative of dose rates to workers located on a storage pad supporting eight casks in the PFSF cask array. For this reason, simplistic conservative methods were used to estimate dose rates at distances of interest from the storage casks, assuming casks at the PFSF are loaded with spent fuel having different characteristics to obtain a conservative dose estimate for use in the accident analysis, and a realistic dose estimate for use in the integrated personnel exposure estimate.

The following assumptions are common to all inlet duct blockage cases analyzed, both the accident analysis cases in Chapter 8 and the case for the annual onsite worker dose estimate in Chapter 7.

- The storage cask with blockage and nearby storage casks are assumed to be HI-STORM storage casks containing PWR spent fuel. This assumption is based on Tables 7.3-1 and 7.3-2 of the PFSF SAR, which indicate that calculated maximum dose rates from HI-STORM storage casks containing design basis fuel are higher than calculated maximum dose rates from TranStor storage casks containing design basis fuel. Based on Table 7.3-1, dose rates on contact and at 1 meter from the side of a HI-STORM storage cask containing PWR spent fuel with 45 GWd/MTU burnup and 5 year cooling time do not differ significantly from those associated with a HI-STORM storage cask containing BWR spent fuel with the same burnup and cooling time; but dose rates on contact with the bottom vent are significantly higher for PWR fuel than a cask containing BWR fuel. For these reasons it is assumed that the storage cask with blockage, as well as nearby storage casks, are HI-STORM casks containing PWR fuel.

- For purposes of estimating distances to the worker from casks near the cask whose inlet ducts are being cleared, it is assumed that the worker clearing the blocked inlet ducts spends 100% of the time at a point located approximately 9 inches away from the affected cask, toward the pad centerline, and 39 inches (1 meter) from the nearest cask in the opposite column on the same storage pad. For purposes of estimating doses from the cask whose inlet ducts have blockage, it is conservatively assumed that the worker's body is in contact with the cask and the workers hands and forearms are in contact with the inlet ducts.

- Dose rates on contact with the affected storage cask and at 1 meter from the nearest cask in the opposite column are taken from Table 7.3-1, "Maximum Dose Rates on Contact and at One Meter From a HI-STORM Storage Cask", or dose rates are scaled from dose rates given in this table to obtain dose rates from HI-STORM casks with less radioactive fuel than the design basis fuel.

- Dose rates at distances beyond 1 meter from the casks are not presented in the PFSF SAR, nor in the HI-STORM or TranStor Storage Cask SARs. Therefore, simplistic conservative methods are used to estimate dose rates at distances of interest. It is assumed that dose rate falls off with distance as 1/r (Reference 4, thumb rule for dose rate vs. distance from a line source), where r is distance from the outer surface of the storage cask. Reference 4 (Chapter 11) states that for a disk source or cylindrical source, "The dose rate falls off a little faster than 1/r but not as fast as $1/r^2$." Since the fuel in the canister contained in a storage cask represents a cylindrical source, use of a line source to approximate dose rates vs. distance may be somewhat conservative but is not unreasonable. Reference 4 also discusses dose rates vs. distance from a point source, and states (Chapter 5):

  "As long as the distance away from the source is at least three times the longest dimension of the source, then inverse square law calculations will give the correct answer to within a percent."

  PWR fuel assemblies are approximately 15 ft high. Therefore, it is considered that it is reasonable to treat the source as a line source and not as a point source out to a distance of 45 ft (3 times source height).

Following are key assumptions for the accident analysis scenarios evaluated in Sections 8.1.3 and 8.2.8 of the PFSF SAR:

- As stated in PFSF SAR Section 8.1.3, it is assumed that it takes a worker 30 minutes to clear inlet ducts from a cask with complete blockage of two of the four air inlet ducts.

- The storage cask with duct blockage is assumed to contain design basis PWR fuel having 45 GWd/MTU burnup and 5 year cooling time.

- Nearby storage casks are assumed to contain PWR fuel having 40 GWd/MTU burnup and 10 year cooling time, conservative average PFSF fuel. This assumption is based on the following statements from PFSF SAR Section 7.3.3.5:

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

Following are key assumptions for the annual onsite worker dose assessment evaluated in Section 7.4 of the PFSF SAR:

- As stated in PFSF SAR Section 7.4, it is assumed that it takes a worker 10 minutes to clear inlet ducts of a single cask having some degree of blockage.
- The storage cask with duct blockage and nearby casks are assumed to contain PWR fuel having 35 GWd/MTU burnup and 20 year cooling time. This is based on PFSF SAR Section 7.4, which states the following:

  "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."

DOE's Energy Information Administration's Service Report entitled "Spent Nuclear Fuel Discharges from U.S. Reactors - 1994", published in February 1996 (Reference 8), provides information regarding characteristics of spent fuel in the U.S. This report was reviewed to evaluate average burnups and cooling time associated with the spent fuel inventory at the end of 1994. At this time, the spent fuel inventory from pressurized water reactors (PWRs) was approximately 19,000 metric tons of uranium (MTU), and the inventory from boiling water reactors (BWRs) approximately 11,000 MTU, for a total inventory of approximately 30,000 MTU. This spent fuel inventory represents 75% of the capacity of the PFSF. While it is recognized that provisions already exist for storage of some of this spent fuel and the PFSF will not furnish storage for this entire inventory, data associated with this spent fuel is considered representative of fuel that the PFSF could be expected to receive. The weighted average burnup (weighted by MTU) for the BWR spent fuel inventory in the U.S. was calculated from Table 6 of the above referenced DOE Report to be approximately 23.8 GWd/MTU, and the weighted average burnup for the PWR spent fuel inventory in the U.S. was calculated from Table 7 of this report to be approximately 32.4 GWd/MTU.

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

Because of the large inventory of spent fuel taken into account (approximately 30,000 MTU), this is considered to be a reasonable representation of typical fuel that will be received at the PFSF. Based on this evaluation, the 35 GWd/MTU burnup and 20-year cooling time spent fuel assumed in the onsite dose assessment is considered to be representative of typical fuel expected to be received at the PFSF whose use will result in reasonably accurate occupational exposure estimates.

REFERENCES

1. Private Fuel Storage Facility Safety Analysis Report, Rev. 0, Docket No. 72-22.

2. Topical Safety Analysis Report for the Holtec International Storage and Transfer Operation Reinforced Module Cask System (HI-STORM 100 Cask System), Holtec Report HI-951312, Docket 72-1014, Revision 1, January 1997.

3. Safety Analysis Report for the TranStor Storage Cask System, SNC-96-72SAR, Sierra Nuclear Corporation, Docket 72-1023, Revision B, March 1997.

4. Gollnick, Daniel A., "Basic Radiation Protection Technology", 3rd Edition, Published by Pacific Radiation Corporation, July 1994.

5. Sierra Nuclear Corporation Design Calculation No. PFS01.10.02.03, Revision 0, Private Fuel Storage SKYSHINE-II ISFSI Dose Rate Calculation (4000 Casks), April 10,1997.

6. DOE/RW-0184-R1, Characteristics of Potential Repository Wastes, Office of Civilian Radioactive Waste Management, U.S. Department of Energy, July 1992. Light Water Reactor Radiological Computer Database for Generic PWR Spent Fuel.

7. NRC NUREG-1567, Standard Review Plan for Spent Fuel Dry Storage Facilities (Draft Report for Comment), October 1996.

8. SR/CNEAF/96-01, Report by the Energy Information Administration of the Department of Energy, "Spent Nuclear Fuel Discharges from U.S. Reactors - 1994", published February 1996.

CONCLUSION

Following is a summary of the results of this calculation:

Accident Analysis

For the accident analysis case involving partial blockage of inlet ducts, it is assumed that an operator spends 30 minutes in the vicinity of a storage cask containing design basis fuel having dose rates of approximately 30 mrem/hr on contact with the side and 50 mrem/hr on contact with the inlet ducts (PFSF SAR Table 7.3-1). The accident analysis assumed an additional 20 mrem/hr general area dose rate from nearby storage casks, resulting in total dose rates of 50 mrem/hr whole body and 70 mrem/hr to the hands and forearms cleaning debris near the inlet ducts. The one-half hour assumed exposure time produces integrated dose to a worker of 25 mrem to the whole body and 35 mrem to the extremities, as presented in Section 8.1.3 of the PFSF SAR. An objective of this calculation document is to check the validity of the assumed 20 mrem/hr general area dose rate.

This calculation determined that the dose rate from nearby casks having a direct radiation path to the assumed worker location would be 33.9 mrem/hr if nearby casks are assumed to contain design basis fuel, which was considered to be overly conservative. While it is assumed that the affected cask contains design basis fuel, it is assumed that nearby casks contain the conservative average PFSF fuel, with 40 GWd/MTU burnup and 10 year cooling time. Scaling down direct radiation dose rates from nearby casks to account for this "cooler" fuel results in a calculated direct dose rate of 16.54 mrem/hr from nearby casks. This was multiplied by a factor of 1.25% to account for scattered radiation at the assumed worker location from canisters which do not have a line-of-sight radiation path to the assumed worker location, resulting in a total estimated dose rate from nearby casks of 20.7 mrem/hr. Based on this calculation, the value of 20 mrem/hr from contribution of nearby casks assumed in the PFSF accident analysis is reasonable.

Integrated Personnel Dose Estimate

For the integrated personnel dose estimate in Section 7.4 of the PFSF SAR, it is assumed that for routine clearing of debris from inlet ducts of 5% of the casks stored at the PFSF annually takes an operator about 10 minutes per cask in a radiation field of approximately 15 mrem/hr. Based on the same assumed operator location used in the accident analysis, the dose rate at this point is calculated to be 64 mrem/hr from direct radiation (canisters having line-of-sight radiation path to the assumed worker location), assuming the affected cask and nearby casks contain design basis fuel. In order to obtain a realistic dose estimate to workers performing routine tasks, it was assumed that the affected cask and nearby casks contain typical PFSF fuel, with 35 GWd/MTU burnup and 20 year cooling time. Scaling down direct radiation dose rates from the affected cask whose inlet ducts are being cleared and nearby casks to account for this "cooler" fuel results in a calculated direct dose rate of 8.6 mrem/hr. This was multiplied by a factor of 1.25% to account for scattered radiation at the assumed worker location from canister which do not have a line-of-sight radiation path to the assumed worker location, resulting in a total estimated dose rate from the affected cask and nearby casks of 10.8 mrem/hr. Based on this calculation, the value of 15 mrem/hr assumed in Section 7.4 of the PFSF SAR for the annual onsite worker dose assessment is reasonable, and somewhat conservative, for typical fuel.

CALCULATION

Accident Analysis

PFSF SAR Section 8.1.3.4 states the following:

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

Based on Section 11.1.2.3 of the TranStor SAR (Reference 3), it is assumed that a person spends 30 minutes in the radiation field clearing blocked inlet ducts. The TranStor SAR characterizes this as a "worst case estimate" for the time to clean the vents, assuming the person is kneeling next to the cask with his hands on the vent inlets the entire time. This is considered to be a conservative estimate for the time required to remove blockage from one-half the inlet ducts.

From PFSF SAR Table 7.3-1, which assumes design basis fuel in a HI-STORM storage cask, the dose rate on contact with the bottom air inlet duct is 50 mrem/hr, and the dose rate on contact with the side of the cask is nearly 30 mrem/hr. These are maximum dose rates associated with design basis PWR fuel with 45 GWd/MTU burnup and 5-year cooled. It was estimated in Chapter 8 of the PFSF SAR that surrounding casks contribute an additional 20 mrem/hr, which is an estimated average dose rate to a person located in the cask array, with surrounding casks loaded with the conservative "average" fuel (assumed to have 40 GWd/MTU burnup and 10 years cooling time). Thus, the total dose rate at the extremities involved in cleaning the blocked inlets is 50 + 20 = 70 mrem/hr, and the total dose rate at the whole body is 30 +20 = 50 mrem/hr. One-half hour exposure time produces integrated doses of 35 mrem to the hands and forearms and 25 mrem to the whole body

The estimated 20 mrem/hr value from casks adjacent to the affected cask can be checked by first calculating dose rates from adjacent casks assuming that they contain design basis fuel, then scaling these dose rates to account for the assumption that the casks are loaded with "cooler" fuel having 40 GWd/MTU burnup and 10 years cooling time. Dose rates from nearby casks within line-of-sight of the location where the operator is assumed to work are conservatively estimated assuming that dose rates fall off with distance from the cylindrical storage cask the same as for a line source, 1/r, where r is the distance away from the cask outer surface. HI-STORM storage casks (which have the highest calculated side dose rates) are approximately 11 ft in diameter and center-to-center spacing on the storage pads is 15 ft (PFSF SAR Section 10.2.1.6), so the surfaces of adjacent casks are approximately 4 ft apart.

It is assumed that the body of the operator kneeling next to the cask with blockage is near the center of a storage pad, due east of the center of the affected cask, as shown in Figure 1. The operator's body is assumed to be less than one ft from the cask with blockage (contact dose rates are conservatively assumed), which locates the body approximately 1.0 meter from the nearest cask in the opposite column on the storage pad. Dose rates at 1.0 meter are taken to be 14.21 mrem/hr (13.66 gamma and 0.55 neutron) for a HI-STORM storage cask containing design basis fuel, based on PFSF SAR Table 7.3.1.

Figure 1 of this calculation, which is based on Figure 4.2-7 of the PFSF SAR, assigns identification numbers to casks near the affected cask. Distances from the assumed operator location to the surfaces of nearby storage casks whose canisters have a direct radiation path are given in Table 1, along with calculated dose rates from each of these casks. Direct radiation from the canisters in casks numbered 1 and 8 in Figure 1 are substantially shielded (>75%) from the assumed worker location by intervening casks (cask no. 3 shields cask no. 1, and cask no. 6 shields cask no. 8), while radiation from the canisters in casks 2 and 9 are completely shielded with no direct path for radiation to travel from these canisters to the assumed worker location without passing through intervening casks (cask no. 4 blocks the canister in cask no. 2, and cask no. 7 blocks the canister in cask no. 9). Even for casks no.s 4 and 7 which are adjacent to the affected cask, radiation from a portion of the canister would have to pass through the concrete walls of the affected cask to reach the assumed worker location (or scatter off other casks). For conservatism, this self shielding of the nearby casks was neglected for all casks whose canisters have some direct radiation path (cask no.s 1, 3, 4, 5, 6, 7, and 8) , and it is assumed that all of the radiation emitted from the canisters in these seven casks has a direct path to the assumed worker location.

Radiation from canisters in casks no.s 2 and 9 and other casks on nearby storage pads that do not have a direct path to the assumed worker location would be substantially shielded by the intervening casks and direct radiation would not contribute significantly to the dose rate (scattered radiation is considered later). This is the result of the massive design of the storage casks, with the HI-STORM cask having walls 26-3/4 inch thick concrete and 2-3/4 inch thick steel (2 inch inner shell and 3/4 inch outer shell), and the TranStor cask having walls 29 inch thick concrete and 2 inch thick steel (inner shell). For comparison purposes, 8.1 inch concrete and 2.7 inch iron would each attenuate the gamma radiation intensity from a Co-60 source (average energy gamma is 1.25 MeV) by a factor of 10 (these are 1/10 thicknesses of material per Reference 4), without consideration for shielding buildup factors. Gamma radiation (which comprises over 95% of the dose rate from a storage cask) would be decreased by at least a factor of 100 and the neutron dose rate would also be substantially reduced (since concrete is an excellent neutron shield) with radiation passing through only one wall thickness of either vendor's storage cask system. For the most part, direct radiation from nearby canisters shielded from the assumed worker location would have to pass through two wall thicknesses, and possibly the intervening canister (depending on the geometry from source location to dose receptor). Therefore, it is justified to consider that the contribution of direct radiation from canisters blocked by intervening casks is negligible in comparison to casks whose canister have a line-of-sight path to the assumed worker location. Complete blockage of direct radiation by intervening cask/canisters is also an assumption in Sierra Nuclear Corporation's (SNC) calculation to assess dose rates from the array of 4,000 storage casks at the PFSF (Reference 5), which states:

> "Due to the close proximity of the casks, the cask side dose rate contributions from all casks except those on the edge of the ISFSI are assumed to be completely blocked by other casks. Thus, for a detector at some distance from a given side of the ISFSI, only the casks in the 'front row' of the ISFSFI will contribute to the cask side dose rate component."

The following Table 1 identifies the distances and calculated dose rates at the assumed worker location from nearby casks whose canisters have some degree of direct radiation path to this location, based on the assumption that there is no attenuation of radiation by intervening casks which would shield the radiation from a portion of the canisters. As noted previously, dose rates are calculated assuming that dose rates decrease linearly with distance (1/r rule for a line source) from the surface of the cask given that the dose rate at 1.0 meter (3.28 ft) from a cask containing design basis fuel is 14.21 mrem/hr (PFSF SAR Table 7.3-1).

Table 1 Calculated Dose Rates to an Operator Clearing Blocked Inlet Ducts, from Casks Near the Affected Cask

Cask Number, from Figure 1	Distance to Assumed Worker Location from Center of Cask (feet)	Distance to Outer Surface of Cask from Assumed Worker Location (feet)	Dose Rate from Cask at Assumed Worker Location (mrem/hour)
1	39.97	34.47	1.35
3	17.37	11.87	3.93
4	16.25	10.75	4.34
5	8.75	3.28	14.21
6	17.37	11.87	3.93
7	16.25	10.75	4.34
8	31.25	25.75	1.81
		Total =	33.91

The 33.9 mrem/hr value is calculated based on the assumption that the casks in the vicinity of the cask with blockage also contain design basis fuel, which is considered overly conservative for estimating dose resulting from the task of clearing debris from inlet ducts. Therefore, scaling factors are applied to assess dose rates assuming nearby casks contain conservative average PFSF fuel assumed to have 40 GWd/MTU burnup and 10 year cooling time rather than design basis fuel. These scaling factors are calculated using source data obtained from the OCRWM LWR Database (Reference 6) using the scaling method applied by Sierra Nuclear Corporation and discussed in Section 5.4.1 of the TranStor SAR (Reference 3). Source data obtained from the OCRWM LWR Database is included in Attachment A.

The following table compares a key portion of the gamma source energy spectrum associated with HI-STORM design basis PWR fuel having 45 GWd/MTU burnup and 5 year cooling time with that associated with PFSF fuel assumed to have 40 GWd/MTU burnup and 10 year cooling time. Gamma energy spectra are compared, and not simply the total gamma production rate, since the fraction of total energy contributed by each energy bin varies significantly with burnup and cooling time.

Table 2 Determination of Gamma Scaling Factor for 40 GWd/MTU Burnup and 10-Year Cooled Fuel

Average Energy (MeV)	45 GWd/MTU, 5-yr cooled, 3.7% enrichment (photons/sec per metric ton heavy metal)	40 GWd/MTU, 10-year cooled, 3.02% enrichment (photons/sec per metric ton heavy metal))	Ratio of photons/sec 40 GWd/45 GWd
0.575	7.221 E+15	4.019 E+15	5.566 E-1
0.850	1.825 E+15	3.859 E+14	2.115 E-1
1.250	7.531 E+14	3.574 E+14	4.746 E-1
1.750	1.079 E+13	4.151 E+12	3.847 E-1
2.250	4.551 E+12	7.575 E+10	1.66 E-2
2.750	1.638 E+11	5.893 E+9	3.60 E-2
3.50	2.101 E+10	7.530 E+8	3.58 E-2

The highest ratio of the gamma source strengths is 5.566 E-1 photons/sec, associated with the relatively low average energy of 0.575 MeV. Section 9.4.2.1 of NRC NUREG-1567 (Reference 7), states "In general, only gamma sources with energies from approximately 0.8 to 2.5 MeV will contribute significantly to the dose rate through typical types of shielding, however all energy ranges should be included in shielding calculations." Considering that the 0.575 MeV average energy bin will not contribute significantly to dose rates outside the storage cask, the highest ratio of the 40 GWd/45 GWd sources is associated with the 1.250 MeV energy bin, having a ratio of 4.75 E-1. This ratio (scaling factor) is conservatively applied to the total gamma dose rate to scale dose rates from all gamma energies from those associated with 45 GWd/MTU 5-year cooled fuel to those applying to 40 GWd/MTU 10-year cooled fuel.

In order to assess dose rates associated with neutrons, it is not necessary to compare neutron energy spectra, since the fraction of total energy contributed from each energy bin does not vary significantly with burnup and cooling time (Section 5.2.2 of Reference 3). For this reason, only the total neutron source strengths extracted from the OCRWM LWR Database for fuel having the two different characteristics are compared. The database indicates that PWR fuel having 45 GWd/MTU 5-year cooled fuel emits 8.340 E+8 neutrons/sec per metric ton heavy metal, while the 40 GWd/MTU 10-year cooled fuel emits 6.757 E+8 neutrons/sec per metric ton heavy metal, resulting in a 40 GWd/45 GWd neutron source ratio of 8.10 E-1. This factor is conservatively applied to the total neutron dose rate associated with storage casks containing design basis fuel to scale neutron dose rates from those associated with 45 GWd/MTU 5-year cooled fuel to those applying to 40 GWd/MTU 10-year cooled fuel.

From PFSF SAR Table 7.3-1, the 14.21 mrem/hr value at 1 meter from the sides of a storage cask consists of 13.66 mrem/hr gamma (96.1%) and 0.55 mrem/hr neutron (3.9%). The 33.9 mrem/hr total dose rate from casks adjacent to the affected cask has the same fraction of gammas and neutrons, with gamma comprising 32.6 mrem/hr and neutron 1.3 mrem/hr. Applying the scaling factors derived above results in dose rates of:

gamma dose rate = (32.6 mrem/hr) (4.75 E-1)	=	15.49 mrem/hr
neutron dose rate = (1.3 mrem/hr) (8.10 E-1)	=	1.05 mrem/hr
total dose rate from adjacent casks	=	16.54 mrem/hr

Besides the casks considered above that have a line-of-sight path from their canister to the worker, it is assumed that scattered radiation from casks whose canisters are shielded from the worker by intervening casks contribute an additional 25% to the overall dose rate, so the total dose rate (direct + scattered radiation) from casks other than that whose inlet ducts are being cleared would be approximately (16.54) (1.25) = 20.7 mrem/hr. Based on this calculation, the 20 mrem/hr estimated contribution from casks in the vicinity of the affected cask to the worker clearing inlet ducts is considered to be reasonable.

Integrated Personnel Dose Assessment

PFSF SAR Section 7.4 states:

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

This assessment is concerned with average dose rates from routine clearing of small amounts of debris from inlet ducts and not worst case conditions. It is assumed that quarterly visual surveillances of the storage casks and pads identify 200 casks each year that have accumulation of debris at the inlet ducts. It is considered reasonable that, on average, one worker with a hand tool (e.g. rake or shovel) can clean up debris near the inlet ducts of a single storage cask in 10 minutes. The 15 mrem/hr is an estimated average dose rate to a person located in the cask array from the affected cask as well as nearby casks, with the affected cask as well as nearby casks all assumed to be loaded with typical fuel. This estimate is checked in the following paragraphs using the dose rates at distance assumptions for the same assumed worker location described above for the accident analysis case. Dose rates are scaled down from those associated with the design basis fuel to approximate dose rates from casks assumed to be loaded with typical fuel having 35 GWd/MTU burnup and 20 years cooling time.

As discussed for the cask blockage case assessed in the accident analysis, it is assumed that the body of the operator is situated less than 1 ft from the cask having debris at the inlet duct near the center of a storage pad supporting 8 casks, which locates the body approximately 1.0 meter from the nearest cask in the opposite column on the storage pad, 10.8 ft from the two casks adjacent to the cask being worked, and 11.9 ft from the two casks adjacent to the nearest cask in the opposite column on the same storage pad (Table 1). Assuming all these casks contain design basis fuel, dose rates were previously calculated to be 30 mrem/hr from contact (conservatively assumed) with the side of the cask being worked, and a total of 33.9 mrem/hr from nearby casks having some line-of-sight radiation path from the canisters to the assumed worker location, for a total dose rate of 30 + 34 = 64 mrem/hr. Scaling factors are again calculated using source data obtained from the OCRWM LWR Database, to scale from PWR fuel having 45 GWd/MTU burnup and 5 year cooling time to that associated with typical PFSF fuel assumed to have 35 GWd/MTU burnup and 20 year cooling time. Source data obtained from the OCRWM LWR Database is included in Attachment A. Gamma scaling factors are calculated in the following table:

Table 3 Determination of Gamma Scaling Factor for 35 GWd/MTU Burnup and 20-Year Cooled Fuel

Average Energy (MeV)	45 GWd/MTU, 5-yr cooled, 3.7% enrichment (photons/sec per metric ton heavy metal)	35 GWd/MTU, 20-year cooled, 3.43% enrichment (photons/sec per metric ton heavy metal)	Ratio of photons/sec 35 GWd/45 GWd
0.575	7.221 E+15	2.524 E+15	3.495 E-1
0.850	1.825 E+15	5.149 E+13	2.821 E-2
1.250	7.531 E+14	9.075 E+13	1.205 E-1
1.750	1.079 E+13	1.412 E+12	1.309 E-1
2.250	4.551 E+12	3.513 E+8	7.719 E-5
2.750	1.638 E+11	4.001 E+8	2.443 E-3
3.50	2.101 E+10	1.828 E+7	8.701 E-4

Eliminating the 0.575 average energy bin due to its negligible contribution to dose rates outside the cask, the highest ratio of the gamma source strengths is associated with the 1.750 MeV energy bin, having a ratio of 1.309 E-1. This factor is conservatively applied to the total gamma dose rate to scale dose rates from all gamma energies from those associated with 45 GWd/MTU 5-year cooled design basis fuel to those applying to 35 GWd/MTU 20-year cooled fuel.

The OCRWM LWR database indicates that PWR fuel having 45 GWd/MTU 5-year cooled fuel emits 8.340 E+8 neutrons/sec per metric ton heavy metal, while the 35 GWd/MTU 20-year cooled fuel emits 1.786 E+8 neutrons/sec per metric ton heavy metal, resulting in a 35 GWd/45 GWd neutron source ratio of 2.14 E-1.

The dose rate from the affected cask and nearby casks assuming they all contain design basis fuel is 64 mrem/hr. Using the same fractions of gamma (96.1%) and neutron (3.9%) radiation as in the previous case gives 61.5 mrem/hr gamma and 2.5 mrem/hr neutron. Applying the scaling factors derived above to the cask whose inlet ducts are being cleared and the seven casks in the immediate vicinity for which at least a fraction of the canisters have a direct path to the assumed worker location results in the following:

gamma dose rate = (61.5 mrem/hr) (1.31 E-1)	= 8.06 mrem/hr
neutron dose rate = (2.5 mrem/hr) (2.14 E-1)	= 0.54 mrem/hr
total dose rate from adjacent casks	= 8.60 mrem/hr

Besides the casks considered above that have some degree of line-of-sight radiation path from their canister to the worker, it is assumed that scattered radiation from casks whose canisters are shielded from the worker by intervening casks would contribute an additional 25% to the overall dose rate, so the total dose rate is estimated to be approximately (8.60) (1.25) = 10.8 mrem/hr. This calculation indicates that the 15 mrem/hr dose rate used to estimate the annual integrated dose to personnel assumed to clear debris from the inlet ducts of 200 casks is reasonable and somewhat conservative, based on the casks containing typical fuel.

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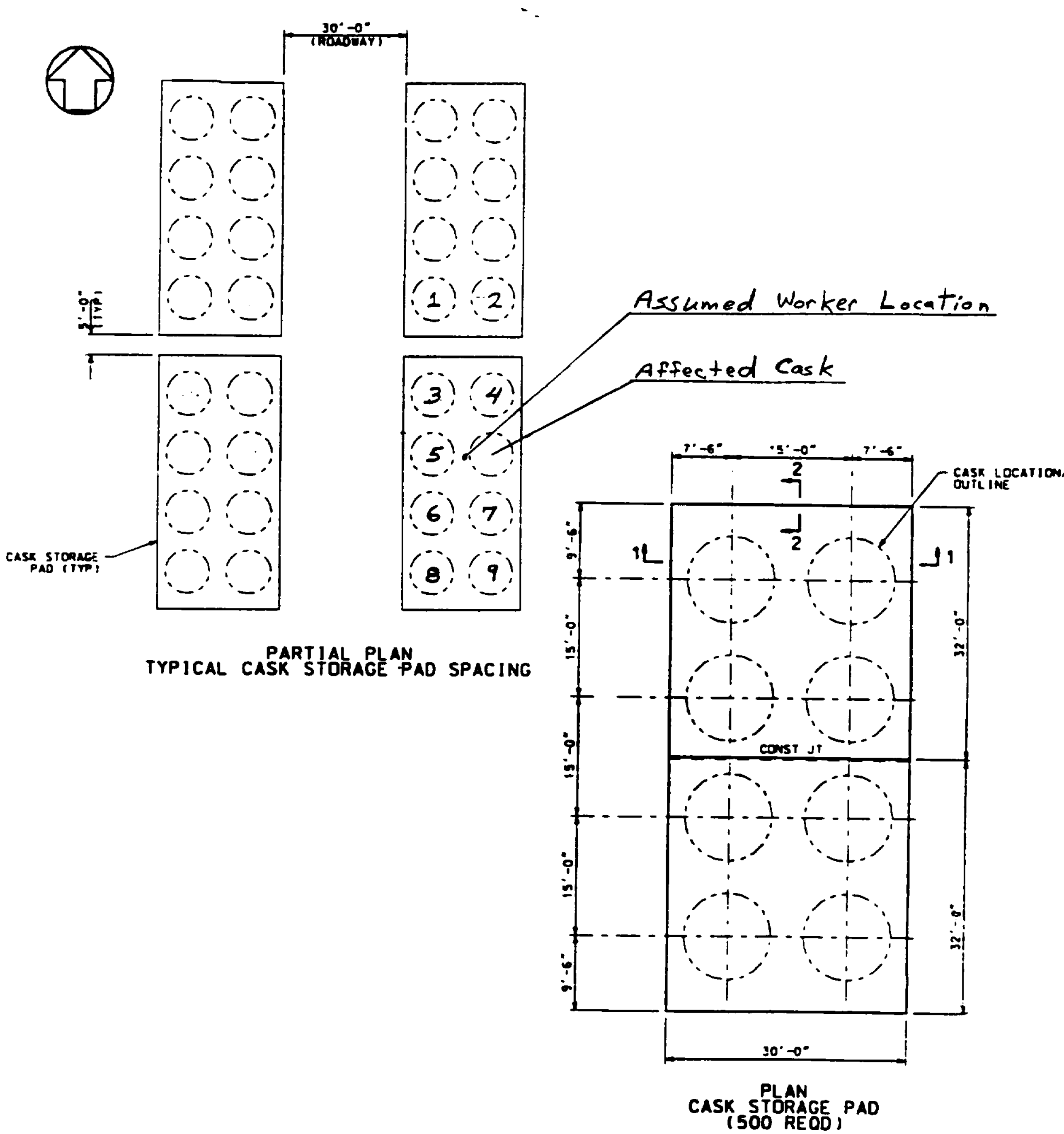
Figure 1 Layout of Storage Casks on the Pads

(Extracted from PFSF SAR Figure 4.2-7)



Attachment A

LWR Radiological DATABASE
PHOTONS REPORT
REACTOR TYPE & BURNUP: PWR 45000
ENRICHMENT: 3.70%
DECAY TIME: 5 YEARS
The data is shown in Photons per second/MTIHM

ENERGY (MeV)	PHO/SEC	% TOTAL
1.000E-02	3.973E+15	21.73%
2.500E-02	9.383E+14	5.13%
3.750E-02	1.026E-15	5.61%
5.750E-02	7.861E+14	4.30%
8.500E-02	5.151E+14	2.82%
1.250E-01	5.314E+14	2.91%
2.250E-01	4.266E+14	2.33%
3.750E-01	2.536E+14	1.39%
5.750E-01	7.221E+15	39.50%
8.500E-01	1.825E+15	9.98%
1.250E+00	7.531E+14	4.12%
1.750E+00	1.079E+13	0.06%
2.250E+00	4.551E+12	0.02%
2.750E+00	1.638E+11	0.00%
3.500E+00	2.101E+10	0.00%
5.000E+00	3.600E+07	0.00%
7.000E+00	4.152E+06	0.00%
9.500E+00	4.770E+05	0.00%
TOTAL	*1.826E+16	99.92%*

*This value was obtained by interpolating TOTALS values from ORIGEN2 run to the specific burnup/enrichment/decay time combination you specified. Percentages have been calculated from this interpolated value and may not add up to 100 percent in all cases.

LWR Radiological DATABASE
RADIOLOGICAL TOTALS REPORT
REACTOR TYPE & BURNUP: PWR 45000
ENRICHMENT: 3.70%
DECAY TIME: 5 YEARS

CURIES/MTIHM

ACTIVATION PRODUCTS	8.148E+03
ACTINIDES AND DAUGHTERS	1.441E+05
FISSION PRODUCTS	5.846E+05
TOTAL	7.373E+05

WATTS/MTIHM

ACTIVATION PRODUCTS	8.810E+01
ACTINIDES AND DAUGHTERS	4.900E+02
FISSION PRODUCTS	2.084E+03
TOTAL	2.667E+03

GRAMS/MTIHM

ACTIVATION PRODUCTS	4.403E+05
ACTINIDES AND DAUGHTERS	9.534E+05
FISSION PRODUCTS	4.625E+04
TOTAL	1.440E+06

NEUTRONS/MTIHM

ALPHA,N NEUTRONS	1.463E+07
SPONTANEOUS FISSION NEUTRONS	8.193E+08
TOTAL NEUTRONS	8.340E+08

PHOTONS per Second/MTIHM

TOTAL PHOTONS/SEC	1.828E+16

*Some of the above values were obtained by interpolating TOTALS values f ORIGEN2 runs to the specific burnup/enrichment/decay time combination yo specified. □

PHOTONS REPORT
REACTOR TYPE & BURNUP: PWR 40000
ENRICHMENT: 3.02%
DECAY TIME: 10 YEARS
The data is shown in Photons per second/MTIHM

ENERGY (MeV)	PHO/SEC	% TOTAL
1.000E-02	2.179E+15	23.53%
2.500E-02	4.669E+14	5.04%
3.750E-02	5.846E+14	6.31%
5.750E-02	4.362E+14	4.71%
8.500E-02	2.577E+14	2.78%
1.250E-01	2.568E+14	2.77%
2.250E-01	2.100E+14	2.27%
3.750E-01	1.032E+14	1.11%
5.750E-01	4.019E+15	43.40%
8.500E-01	3.859E+14	4.17%
1.250E+00	3.574E+14	3.86%
1.750E+00	4.151E+12	0.04%
2.250E+00	7.575E+10	0.00%
2.750E+00	5.893E+09	0.00%
3.500E+00	7.530E+08	0.00%
5.000E+00	2.913E+07	0.00%
7.000E+00	3.359E+06	0.00%
9.500E+00	3.859E+05	0.00%
TOTAL	9.261E+15	100.00%

RADIOLOGICAL TOTALS REPORT
REACTOR TYPE & BURNUP: PWR 40000
ENRICHMENT: 3.02%
DECAY TIME: 10 YEARS

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CURIES/MTIHM	
ACTIVATION PRODUCTS	3.999E+03
ACTINIDES AND DAUGHTERS	1.180E+05
FISSION PRODUCTS	3.515E+05
TOTAL	4.735E+05
WATTS/MTIHM	
ACTIVATION PRODUCTS	4.476E+01
ACTINIDES AND DAUGHTERS	4.360E+02
FISSION PRODUCTS	1.058E+03
TOTAL	1.539E+03
GRAMS/MTIHM	
ACTIVATION PRODUCTS	4.403E+05
ACTINIDES AND DAUGHTERS	9.589E+05
FISSION PRODUCTS	4.111E+04
TOTAL	1.440E+06
NEUTRONS/MTIHM	
ALPHA,N NEUTRONS	1.295E+07
SPONTANEOUS FISSION NEUTRONS	6.627E+08
TOTAL NEUTRONS	6.757E+08
PHOTONS per Second/MTIHM	
TOTAL PHOTONS/SEC	9.261E+15

PHOTONS REPORT
REACTOR TYPE & BURNUP: PWR 35000
ENRICHMENT: 3.43%
DECAY TIME: 20 YEARS
The data is shown in Photons per second/MTIHM

ENERGY (MeV)	PHO/SEC	% TOTAL
1.000E-02	1.541E+15	26.88%
2.500E-02	3.098E+14	5.40%
3.750E-02	3.752E+14	6.55%
5.750E-02	3.239E+14	5.65%
8.500E-02	1.709E+14	2.98%
1.250E-01	1.388E+14	2.42%
2.250E-01	1.435E+14	2.50%
3.750E-01	6.044E+13	1.05%
5.750E-01	2.524E+15	44.03%
8.500E-01	5.149E+13	0.90%
1.250E+00	9.075E+13	1.58%
1.750E+00	1.412E+12	0.02%
2.250E+00	3.513E+08	0.00%
2.750E+00	4.001E+08	0.00%
3.500E+00	1.828E+07	0.00%
5.000E+00	7.547E+06	0.00%
7.000E+00	8.698E+05	0.00%
9.500E+00	9.989E+04	0.00%
TOTAL	*5.731E+15	99.99%*

*This value was obtained by interpolating TOTALS values from ORIGEN2 runs to the specific burnup/enrichment/decay time combination you specified. Percentages have been calculated from this interpolated value and may not add up to 100 percent in all cases.

RADIOLOGICAL TOTALS REPORT
REACTOR TYPE & BURNUP: PWR 35000
ENRICHMENT: 3.43%
DECAY TIME: 20 YEARS

CURIES/MTIHM

ACTIVATION PRODUCTS	1.146E+03
ACTINIDES AND DAUGHTERS	6.022E+04
FISSION PRODUCTS	2.355E+05
TOTAL	2.968E+05

WATTS/MTIHM

ACTIVATION PRODUCTS	1.005E+01
ACTINIDES AND DAUGHTERS	2.632E+02
FISSION PRODUCTS	6.752E+02
TOTAL	9.505E+02

GRAMS/MTIHM

ACTIVATION PRODUCTS	4.403E+05
ACTINIDES AND DAUGHTERS	9.636E+05
FISSION PRODUCTS	3.604E+04
TOTAL	1.440E+06

NEUTRONS/MTIHM

ALPHA,N NEUTRONS	7.570E+06
SPONTANEOUS FISSION NEUTRONS	1.708E+08
TOTAL NEUTRONS	1.786E+08

PHOTONS per Second/MTIHM

TOTAL PHOTONS/SEC	5.732E+15

*Some of the above values were obtained by interpolating TOTALS values from ORIGEN2 runs to the specific burnup/enrichment/decay time combination you specified. □