CHAPTER 5†: SHIELDING EVALUATION

5.0 INTRODUCTION

The shielding analysis of the HI-STORM FW system is presented in this chapter. As described in Chapter 1, the HI-STORM FW system is designed to accommodate both PWR and BWR MPCs within HI-STORM FW overpacks (see Table 1.0.1).

In addition to storing intact PWR and BWR fuel assemblies, the HI-STORM FW system is designed to store BWR and PWR damaged fuel assemblies and fuel debris. Damaged fuel assemblies and fuel debris are defined in Subsection 2.1. Both damaged fuel assemblies and fuel debris are required to be loaded into Damaged Fuel Containers (DFCs).

PWR fuel assemblies may contain burnable poison rod assemblies (BPRAs), with any number of full-length rods and thimble plug rodlets in the locations without a full-length rod, thimble plug devices (TPDs), control rod assemblies (CRAs) or axial power shaping rod assemblies (APSRs), neutron source assemblies (NSAs), or similarly named devices. These non-fuel hardware devices are an integral yet removable part of PWR fuel assemblies and therefore the HI-STORM FW system has been designed to store PWR fuel assemblies with or without these devices. Since each device occupies the same location within a fuel assembly, a single PWR fuel assembly will not contain multiple devices, with the exception of instrument tube tie rods (ITTRs), which may be stored in the assembly along with other types of non-fuel hardware.

As described in Chapter 1(see Tables 1.2.3 and 1.2.4), the loading of fuel in all HI-STORM FW MPCs will follow specific heat load limitations.

In order to offer the user more flexibility in fuel storage, the HI-STORM FW System offers two heat load patterns, each with a three-region loading configuration, in the MPC-37. The MPC-89 has one heat load pattern with a three-region loading configuration. The regionalized storage patterns are guided by the considerations of minimizing occupational and site boundary dose to comply with ALARA principles.

The sections that follow will demonstrate that the design of the HI-STORM FW dry cask storage system fulfills the following acceptance criteria outlined in the Standard Review Plan, NUREG-1536 [5.2.1]:

† This chapter has been prepared in the format and section organization set forth in Regulatory Guide 3.61. However, the material content of this chapter also fulfills the requirements of NUREG-1536. Pagination and numbering of sections, figures, and tables are consistent with the convention set down in Chapter 1, Section 1.0, herein. Finally, all terms-of-art used in this chapter are consistent with the terminology of the glossary and component nomenclature of the Bill-of-Materials (Section 1.5).

Acceptance Criteria

1. The minimum distance from each spent fuel handling and storage facility to the controlled area boundary must be at least 100 meters. The "controlled area" is defined in 10CFR72.3 as the area immediately surrounding an ISFSI or monitored retrievable storage (MRS) facility, for which the licensee exercises authority regarding its use and within which ISFSI operations are performed.

2. The system designer must show that, during both normal operations and anticipated occurrences, the radiation shielding features of the proposed dry cask storage system are sufficient to meet the radiation dose requirements in Sections 72.104(a). Specifically, the vendor must demonstrate this capability for a typical array of casks in the most bounding site configuration. For example, the most bounding configuration might be located at the minimum distance (100 meters) to the controlled area boundary, without any shielding from other structures or topography.

3. Dose rates from the cask must be consistent with a well established "as low as reasonably achievable" (ALARA) program for activities in and around the storage site.

4. After a design-basis accident, an individual at the boundary or outside the controlled area shall not receive a dose greater than the limits specified in 10CFR72.106.

5. The proposed shielding features must ensure that the dry cask storage system meets the regulatory requirements for occupational and radiation dose limits for individual members of the public, as prescribed in 10CFR Part 20, Subparts C and D.

Consistent with the Standard Review Plan, NUREG-1536, this chapter contains the following information:

- A description of the shielding features of the HI-STORM FW system, including the HI-TRAC transfer cask.
- A description of the source terms.
- A general description of the shielding analysis methodology.
- A description of the analysis assumptions and results for the HI-STORM FW system, including the HI-TRAC transfer cask.
- Analyses are presented for each MPC showing that the radiation dose rates follow As-Low-As-Reasonably-Achievable (ALARA) practices.
- Analyses to show that the 10CFR72.106 controlled area boundary radiation dose limits can be met during accident conditions of storage for non-effluent radiation from illustrative ISFSI configurations at a minimum distance of 100 meters. Since only representative dose rate values for normal conditions are presented for this chapter, compliance with the radiation and

exposure objectives of 10CFR72.104 is not being evaluated herein but will be performed as part of the site specific evaluations.

Chapter 2 contains a detailed description of structures, systems, and components important to safety.

Chapter 7 contains a discussion on the release of radioactive materials from the HI-STORM FW system. Therefore, this chapter only calculates the dose from direct neutron and gamma radiation emanating from the HI-STORM FW system.

Chapter 11, Radiation Protection, contains the following information:

- A discussion of the estimated occupational exposures for the HI-STORM FW system, including the HI-TRAC transfer cask.
- A summary of the estimated radiation exposure to the public.

The safety analyses summarized in this chapter demonstrate that under accident conditions, acceptable margins to allowable limits exist under all design basis loading conditions. For normal and off-normal conditions, the analyses in this chapter simply provide a generic evaluation that demonstrates that the dose requirements as specified in 10CFR72.104 can be met under site specific conditions. Minor changes to the design parameters that inevitably occur during the product's life cycle which are treated within the purview of 10CFR72.48 and are ascertained to have an insignificant effect on the computed dose rates in this chapter may not prompt a formal reanalysis and revision of the results and associated data in the tables of this chapter unless the cumulative effect of all such unquantified changes cannot be deemed any more to be insignificant. For accident conditions, the dose limit as specified in 10CFR72.106 is 5 rem. The only accident which impacts dose rates is the loss of water in the water jacket for the HI-TRAC VW. For the purposes of determining if the changes to the HI-TRAC VW are insignificant, an insignificant loss of margin with reference to the 5 rem acceptance criteria is defined as the estimated reduction that is no more than one order of magnitude less than the available margin reported in the FSAR. For normal and off-normal conditions, site specific dose evaluations are required to demonstrate compliance with 10CFR72.104. Incorporating any minor changes into those site specific evaluations is only warranted if it would be expected, on a site specific basis, that those changes could result in a situation where the limits are no longer met and where therefore other compensatory measures are required, such as a change in the loading plan or the concrete density. Incorporating changes into the analyses in this chapter for normal and off-normal conditions will only be performed under extenuating circumstances, e.g. major changes to the shielding design, in order to provide an updated template for the site specific dose analyses.

To ensure rigorous configuration control, the information in the Licensing drawings in Section 1.5 should be treated as the authoritative source for numerical analysis at all times. Reliance on the input data and associated results in this chapter for additional mathematical computations may not be appropriate as they serve the sole purpose of establishing safety compliance in accordance with the acceptance criteria set down in Chapter 2 and in this chapter.

5.1 DISCUSSION AND RESULTS

The principal sources of radiation in the HI-STORM FW system are:

- Gamma radiation originating from the following sources:

  1. Decay of radioactive fission products
  2. Secondary photons from neutron capture in fissile and non-fissile nuclides
  3. Hardware activation products generated during core operations

- Neutron radiation originating from the following sources

  1. Spontaneous fission
  2. α,n reactions in fuel materials
  3. Secondary neutrons produced by fission from subcritical multiplication
  4. γ,n reactions (this source is negligible)

During loading, unloading, and transfer operations, shielding from gamma radiation is provided by the stainless steel structure and the basket of the MPC and the steel, lead, and water in the HI-TRAC transfer cask. For storage, the gamma shielding is provided by the MPC, and the steel and concrete ("Metcon" structure) of the overpack. Shielding from neutron radiation is provided by the concrete of the overpack during storage and by the water of the HI-TRAC transfer cask during loading, unloading, and transfer operations. It is worth noting that the models, used to evaluate the dose calculations in this chapter, are constructed with minimum concrete densities and minimum lead thicknesses.

The shielding analyses were performed with MCNP5 [5.1.1] developed by Los Alamos National Laboratory (LANL). The source terms for the design basis fuels were calculated with the SAS2H and ORIGEN-S sequences from the SCALE 5 system [5.1.2, 5.1.3]. A detailed description of the MCNP models and the source term calculations are presented in Sections 5.3 and 5.2, respectively.

The design basis zircaloy clad fuel assemblies used for calculating the dose rates presented in this chapter are Westinghouse (W) 17x17 and the General Electric (GE) 10x10, for PWR and BWR fuel types, respectively. Required site specific shielding evaluations will verify whether those assemblies and assembly parameters are appropriate for the site-specific analyses. Subsection 2.1 specifies the acceptable fuel characteristics, including the acceptable maximum burnup levels and minimum cooling times for storage of fuel in the HI-STORM FW MPCs.

The following presents a discussion that explains the rationale behind the burnup and cooling time combinations that are evaluated in this chapter for normal and accident conditions.

10CFR72 contains two sections that set down main dose rate requirements: §104 for normal and off-normal conditions, and §106 for accident conditions. The relationship of these requirements to the analyses in this Chapter 5, and the burnup and cooling times selected for the various analyses, are as follows:

- 10CFR72.104 specifies the dose limits from an ISFSI (and other operations) at a site boundary under normal and off-normal conditions. Compliance with §104 can therefore only be demonstrated on a site-specific basis, since it depends not only on the design of the cask system and the loaded fuel, but also on the ISFSI layout, the distance to the site boundary, and possibly other factors such as use of higher density concrete or the terrain around the ISFSI. The purpose of this chapter is therefore to present a general overview over the expected dose rates, next to the casks and at various distances, to aid the user in applying ALARA considerations and planning of the ISFSI. To that extent, it is sufficient to present reasonably conservative dose rate values, based on a reasonable conservative choice of burnups and cooling times of the assemblies.
- For the accident dose limit in 10CFR72.106 it is desirable to show compliance in this Chapter 5 on a generic basis, so that calculations on a site-by-site basis are not required. To that extend, a burnup and cooling time calculation that maximizes the dose rate under accident conditions needs to be selected.

The HI-STORM FW System offers three-region loading configurations as shown in Table 1.2.3 and Table 1.2.4 in Chapter 1.

- For the MPC-37, there are two heat load patterns, each with a three-region loading configuration – Loading Pattern A and Loading Pattern B. An important difference between Pattern A and Pattern B loading is the loading is the maximum allowed heat load of the cells on the periphery of the MPC-37. Pattern A contains the cells with the lowest decay heat on the periphery, while Pattern B contains the cells with the highest decay heat on the periphery. In Pattern A, fuel assemblies with higher heat loads are loaded in the inner region allowing the user to take advantage of self-shielding from the fuel assemblies with lower heat loads in the outer regions. However, for Pattern B, the fuel assemblies with the higher heat loads could be loaded in the outer region (Region 3). Based on this difference it is expected that Pattern B will have higher dose rates than Pattern A. Therefore, for dose calculations Pattern B is selected, as it is the more limiting of the two loading patterns. Furthermore, uniform loading of MPC-37 cells is assumed for dose calculations. The burnup and cooling time combination is selected as representative of the cells on the periphery. This is a conservative approach, as it assumes that all thirty seven cells have a decay heat per cell equal to or slightly exceeding the decay heat of the periphery cells.

- For the MPC-89, there is only one heat load pattern with a three-region loading configuration. Based on the configuration for the MPC-89, fuel assemblies with higher heat loads would be loaded in the inner region allowing the user to take advantage of self-shielding from fuel assemblies with lower heat loads in the outer regions (see Table 1.2.4). However, for simplification, the shielding analyses are performed for a single region, i.e. assuming all assemblies in the basket have the same burnup and cooling time.

In the case of the MPC-89 the burnup and cooling time combination is selected as a representative average for the entire basket.

While Loading Pattern B for the MPC-37 allows assemblies with higher heat loads and therefore higher source terms in the outer region (Region 3) of the MPC, the guiding principle in selecting fuel loading should still be to preferentially place assemblies with higher source terms in the inner regions of the basket as far as reasonably possible.

It is recognized that for a given heat load, an infinite number of burnup and cooling time combination could be selected, which would result in slightly different dose rate distributions around the cask. For a high burnup with a corresponding longer cooling time, dose locations with a high neutron contribution would show higher dose values, due to the non-linear relationship between burnup and neutron source term. At other locations dose rates are more dominated by contribution from the gamma sources. In these cases, short cooling time and lower burnup combinations with heat load comparable to the higher burnup and corresponding longer cooling time combinations would result in higher dose rates. However, in those cases, there would always be a compensatory effect, since for each dose location, higher neutron dose rates would be partly offset by lower gamma dose rates and vice versa.

Based on these considerations, average burnup and cooling time values are selected for all calculations for normal conditions, i.e values that are away from the extreme values. The selected values are shown in Table 5.0.1, and are based on a total heat loads presented in Table 1.2.3. For the accident conditions however, it is recognized that the bounding accident condition is the loss of water in the HI-TRAC VW, a condition that is neutron dominated due to the removal of the principal neutron absorber in the HI-TRAC VW (water). For this case, the upper bound burnup is selected, in order to maximize the neutron source strength of all assemblies in the basket, and a corresponding higher cooling time is selected in order to meet the overall heat load limit in the cask. The resulting burnup and cooling times values for accidents are therefore different from those for normal conditions and are listed in Table 5.0.2. In all cases, low initial enrichments are selected, which further increases the neutron source terms from the assemblies

With the burnup and cooling times selected based on above considerations, dose rates calculated for normal conditions will be reasonably conservative, while for accident conditions those will represent reasonable upper bound limits.

Table 5.0.1

DESIGN BASIS FUEL BURNUP, COOLING TIME AND ENRICHMENT FOR NORMAL CONDITIONS

Design Basis Burnup and Cooling Times Zircaloy Clad Fuel	
MPC-37	**MPC-89**
45,000 MWD/MTU 4.5 Year Cooling 3.6 wt% U-235 Enrichment	45,000 MWD/MTU 5 Year Cooling 3.2 wt% U-235 Enrichment

Table 5.0.2

DESIGN BASIS FUEL BURNUP, COOLING TIME AND ENRICHMENT FOR ACCIDENT CONDITIONS

Design Basis Burnup and Cooling Times Zircaloy Clad Fuel	
MPC-37	**MPC-89**
65,000 MWD/MTU 8 Year Cooling 4.8 wt% U-235 Enrichment	65,000 MWD/MTU 10 Year Cooling 4.8 wt% U-235 Enrichment

5.1.1 Normal and Off-Normal Operations

Chapter 12 discusses the potential off-normal conditions and their effect on the HI-STORM FW system. None of the off-normal conditions have any impact on the shielding analysis. Therefore, off-normal and normal conditions are identical for the purpose of the shielding evaluation.

The 10CFR72.104 criteria for radioactive materials in effluents and direct radiation during normal operations are:

1. During normal operations and anticipated occurrences, the annual dose equivalent to any real individual who is located beyond the controlled area, must not exceed 25 mrem to the whole body, 75 mrem to the thyroid and 25 mrem to any other critical organ.

2. Operational restrictions must be established to meet as low as reasonably achievable (ALARA) objectives for radioactive materials in effluents and direct radiation.

10CFR20 Subparts C and D specify additional requirements for occupational dose limits and radiation dose limits for individual members of the public. Chapter 11 specifically addresses these regulations.

In accordance with ALARA practices, design objective dose rates are established for the HI-STORM FW system and presented in Table 2.3.2.

Figure 5.1.1 identifies the locations of the dose points referenced in the dose rate summary tables for the HI-STORM FW overpack. Dose Point #2 is located on the side of the cask at the axial mid-height. Dose Points #1 and #3 are the locations of the inlet and outlet air ducts, respectively. The dose values reported for these locations (adjacent and 1 meter) were averaged over the duct opening. Dose Point #4 is the dose location on the overpack lid. The dose values reported at the locations shown on Figure 5.1.1 are averaged over a region that is approximately 1 foot in width.

Figure 5.1.2 identifies the location of the dose points for the HI-TRAC VW transfer cask. Dose Point Locations #1 and #3 are situated below and above the water jacket, respectively. Dose Point #4 is the dose location on the HI-TRAC VW lid and dose rates below the HI-TRAC VW are estimated with Dose Point #5. Dose Point Location #2 is situated on the side of the cask at the axial mid-height.

The total dose rates presented are presented for two cases: with and without BPRAs. The dose from the BPRAs was conservatively assumed to be the maximum calculated in Subsection 5.2.4.

Tables 5.1.1 and 5.1.2 provides dose rates adjacent to and one meter from the HI-TRAC VW during normal conditions for the MPC-37 and MPC-89. The dose rates listed in Table 5.1.1 correspond to the normal condition in which the MPC is dry and the HI-TRAC water jacket is filled with water.

Tables 5.1.5 and 5.1.6 provide the design basis dose rates adjacent to the HI-STORM FW overpack during normal conditions for the MPC-37 and MPC-89. Tables 5.1.7 and 5.1.8 provide the design basis dose rates at one meter from the HI-STORM FW overpack containing the MPC-37 and MPC-89, respectively.

The dose to any real individual at or beyond the controlled area boundary is required to be below 25 mrem per year. The minimum distance to the controlled area boundary is 100 meters from the ISFSI. Table 5.1.2 presents the annual dose to an individual from a single HI-STORM FW cask and various storage cask arrays, assuming an 8760 hour annual occupancy at the dose point location. The minimum distance required for the corresponding dose is also listed. It is noted that these data are provided for illustrative purposes only. A detailed site-specific evaluation of dose at the controlled area boundary must be performed for each ISFSI in accordance with 10CFR72.212. The site-specific evaluation will consider dose from other portions of the facility and will consider the actual conditions of the fuel being stored (burnup and cooling time).

Figure 5.1.3 is an annual dose versus distance graph for the HI-STORM FW cask array configurations provided in Table 5.1.3. This curve, which is based on an 8760 hour occupancy, is provided for illustrative purposes only and will be re-evaluated on a site-specific basis.

Subsection 5.2.3 discusses the BPRAs, TPDs, CRAs and APSRs that are permitted for storage in the HI-STORM FW system. Subsection 5.4.4 discusses the increase in dose rate as a result of adding non-fuel hardware in the MPCs.

The analyses summarized in this section demonstrate that the HI-STORM FW system is in compliance with the radiation and exposure objectives of 10CFR72.106. Since only representative dose rate values for normal conditions are presented in this chapter, compliance with 10CFR72.104 is not being evaluated. This will be performed as part of the site specific evaluations.

5.1.2 Accident Conditions

The 10CFR72.106 radiation dose limits at the controlled area boundary for design basis accidents are:

> 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. The minimum distance from the spent fuel or high-level radioactive waste handling and storage facilities to the nearest boundary of the controlled area shall be at least 100 meters.

Structural evaluations, presented in Chapter 3, shows that a freestanding HI-STORM FW storage overpack containing a loaded MPC remains standing during events that could potentially lead to a tip-over event. Therefore, the tip-over accident is not considered as part of the shielding evaluation.

Design basis accidents which may affect the HI-STORM FW overpack can result in limited and localized damage to the outer shell and radial concrete shield. As the damage is localized and the vast majority of the shielding material remains intact, the effect on the dose at the site boundary is negligible. Therefore, the site boundary doses for the loaded HI-STORM FW overpack for accident conditions are equivalent to the normal condition doses, which meet the 10CFR72.106 radiation dose limits. However the adjacent and one meter dose rates may be increased, which should be considered in any post-accident activities near the affected cask.

The design basis accidents analyzed in Chapter 11 have one bounding consequence that affects the shielding materials of the HI-TRAC transfer cask. It is the potential for damage to the water jacket shell and the loss of the neutron shield (water). In the accident consequence analysis, it is conservatively assumed that the neutron shield (water) is completely lost and replaced by a void.

Throughout all design basis accident conditions the axial location of the fuel will remain fixed within the MPC because of the MPC's design features (see Chapter 1). Further, the structural evaluation of the HI-TRAC VW in Chapter 3 shows that the inner shell, lead, and outer shell remain intact throughout all design basis accident conditions. Localized damage of the HI-TRAC outer shell is possible; however, localized deformations will have only a negligible impact on the dose rate at the boundary of the controlled area.

The complete loss of the HI-TRAC neutron shield significantly affects the dose at mid-height (Dose Point #2) adjacent to the HI-TRAC. Loss of the neutron shield has a small effect on the dose at the other dose points. To illustrate the impact of the design basis accident, the dose rates at Dose Point #2 (see Figure 5.1.2) are provided in Table 5.1.4 (MPC-37) for the HI-TRAC VW at a distance of 1 meter and at a distance of 100 meters. The normal condition dose rates are provided for reference. The dose for a period of 30 days is shown in Table 5.1.9, where 30 days is used to illustrate the radiological impact for a design basis accident. Based on this dose rate and the short duration of use for the loaded HI-TRAC transfer cask, it is evident that the dose as a result of the design basis accident cannot exceed 5 rem at the controlled area boundary for the short duration of the accident.

Analyses summarized in this section demonstrate that the HI-STORM FW system, including the HI-TRAC VW transfer cask, is in compliance with the 10CFR72.106 limits.

Table 5.1.1

DOSE RATES FROM THE HI-TRAC VW FOR NORMAL CONDITIONS
MPC-37 DESIGN BASIS FUEL
45,000 MWD/MTU AND 4.5-YEAR COOLING

Dose Point Location	Fuel Gammas (mrem/hr)	(n,γ) Gammas (mrem/hr)	^{60}Co Gammas (mrem/hr)	Neutrons (mrem/hr)	Totals (mrem/hr)	Totals with BPRAs (mrem/hr)
ADJACENT TO THE HI-TRAC VW						
1	975	25	808	67	1874	1874
2	2939	75	<1	154	3169	3169
3	20	5	339	6	371	561
4	98	1	530	225	854	1147
5	940	3	2074	1022	4038	4038
ONE METER FROM THE HI-TRAC VW						
1	695	12	99	30	835	835
2	1382	22	10	58	1472	1474
3	268	6	142	9	425	501
4	80	<1	295	73	449	613
5	470	1	1129	297	1897	1897

Notes:

- Refer to Figure 5.1.2 for dose locations.
- Values are rounded to nearest integer.
- Dose rates are based on no water within the MPC, an empty annulus, and a water jacket full of water. For the majority of the duration that the HI-TRAC bottom lid is installed, the MPC cavity will be flooded with water. The water within the MPC greatly reduces the dose rate.
- Streaming may occur through the annulus. However, during handling/operations the annulus is filled with water and lead snakes are typically present to reduce the streaming effects. Further, operators are not present on top of the transfer cask.
- The "Fuel Gammas" category includes gammas from the spent fuel, ^{60}Co from the spacer grids, and ^{60}Co from the BPRAs in the active fuel region.

Table 5.1.2 DOSE RATES FROM THE HI-TRAC VW FOR NORMAL CONDITIONS MPC-89 DESIGN BASIS FUEL 45,000 MWD/MTU AND 5-YEAR COOLING					
Dose Point Location	**Fuel Gammas (mrem/hr)**	**(n,γ) Gammas (mrem/hr)**	**^{60}Co Gammas (mrem/hr)**	**Neutrons (mrem/hr)**	**Totals (mrem/hr)**
ADJACENT TO THE HI-TRAC VW					
1	244	18	2247	40	2549
2	2466	107	<1	219	2793
3	3	3	581	4	591
4	25	<1	505	138	669
5	132	2	2135	720	2989
ONE METER FROM THE HI-TRAC VW					
1	411	13	291	29	744
2	1142	30	21	74	1267
3	119	5	280	8	412
4	16	<1	300	43	360
5	79	<1	1202	202	1484

Notes:

- Refer to Figure 5.1.2 for dose locations.
- Values are rounded to nearest integer.
- Dose rates are based on no water within the MPC, an empty annulus, and a water jacket full of water. For the majority of the duration that the HI-TRAC bottom lid is installed, the MPC cavity will be flooded with water. The water within the MPC greatly reduces the dose rate.
- Streaming may occur through the annulus. However, during handling/operations the annulus is filled with water and lead snakes are typically present to reduce the streaming effects. Further, operators are not present on top of the transfer cask.
- The “Fuel Gammas” category includes gammas from the spent fuel and ^{60}Co from the spacer grids.

Table 5.1.3

DOSE RATES FOR ARRAYS OF HI-STORM FWs with MPC-37

Array Configuration	1 cask	2x2	2x3	2x4	2x5
HI-STORM FW Overpack					
45,000 MWD/MTU AND 4.5-YEAR COOLING					
Annual Dose (mrem/year)	18	15	23	11	14
Distance to Controlled Area Boundary (meters)	300	400	400	500	500

Notes:

- Values are rounded to nearest integer.
- 8760 hour annual occupancy is assumed.
- Dose location is at the center of the long side of the array.

Table 5.1.4

DOSE RATES FROM HI-TRAC VW WITH MPC-37
FOR ACCIDENT CONDITIONS
AT BOUNDING BURNUP AND COOLING TIMES

Dose Point Location	Fuel Gammas (mrem/hr)	(n,γ) Gammas (mrem/hr)	^{60}Co Gammas (mrem/hr)	Neutrons (mrem/hr)	Totals (mrem/hr)	Totals with BPRAs (mrem/hr)
ONE METER FROM HI-TRAC VW						
65,000 MWD/MTU AND 8-YEAR COOLING						
2 (Accident Condition)	1735	3	13	2651	4403	4407
2 (Normal Condition)	893	50	7	122	1071	1074
100 METERS FROM HI-TRAC VW						
65,000 MWD/MTU AND 8-YEAR COOLING						
2 (Accident Condition)	0.7	<0.1	0.1	1.4	2.3	2.4

Notes:
- Refer to Figure 5.1.2 for dose locations.
- Values are rounded to nearest integer where appropriate.
- The "Fuel Gammas" category includes gammas from the spent fuel, ^{60}Co from the spacer grids, and ^{60}Co from the BPRAs in the active fuel region.

Table 5.1.5

DOSE RATES ADJACENT TO HI-STORM FW OVERPACK
FOR NORMAL CONDITIONS
MPC-37
BURNUP AND COOLING TIME
45,000 MWD/MTU AND 4.5-YEAR COOLING

Dose Point Location	Fuel Gammas (mrem/hr)	(n,γ) Gammas (mrem/hr)	^{60}Co Gammas (mrem/hr)	Neutrons (mrem/hr)	Totals (mrem/hr)	Totals with BPRAs (mrem/hr)
1	273	2	14	4	292	292
2	135	1	<1	1	141	141
3 (surface)	11	1	25	2	39	53
3 (overpack edge)	13	<1	63	1	78	113
4 (center)	<1	1	<1	<1	<4	<4
4 (mid)	1	1	4	1	7	10
4 (outer)	10	<1	30	<1	42	59

Notes:

- Refer to Figure 5.1.1 for dose locations.
- Values are rounded to nearest integer where appropriate.
- Dose location 3 (surface) is at the surface of the outlet vent. Dose location 3 (overpack edge) is in front of the outlet vent, but located radially above the overpack outer diameter.
- Dose location 4 (center) is at the center of the top surface of the top lid. Dose location 4 (mid) is situated directly above the vertical section of the outlet vent. Dose location 4 (outer) is extended along the top plane of the top lid, located radially above the overpack outer diameter.
- The "Fuel Gammas" category includes gammas from the spent fuel, ^{60}Co from the spacer grids, and ^{60}Co from the BPRAs in the active fuel region.

Table 5.1.6

DOSE RATES ADJACENT TO HI-STORM FW OVERPACK
FOR NORMAL CONDITIONS
MPC-89
BURNUP AND COOLING TIME
45,000 MWD/MTU AND 5-YEAR COOLING

Dose Point Location	**Fuel Gammas (mrem/hr)**	**(n,γ) Gammas (mrem/hr)**	**^{60}Co Gammas (mrem/hr)**	**Neutrons (mrem/hr)**	**Totals (mrem/hr)**
1	172	2	31	3	208
2	92	2	<1	1	96
3 (surface)	3	<1	29	2	35
3 (overpack edge)	5	<1	69	<1	76
4 (center)	0.1	0.4	0.4	0.1	1
4 (mid)	0.2	0.5	4.3	0.5	6
4 (outer)	2	<1	33	<1	37

Notes:
- Refer to Figure 5.1.1 for dose locations.
- Values are rounded to nearest integer where appropriate.
- Dose location 3 (surface) is at the surface of the outlet vent. Dose location 3 (overpack edge) is in front of the outlet vent, but located radially above the overpack outer diameter.
- Dose location 4 (center) is at the center of the top surface of the top lid. Dose location 4 (mid) is situated directly above the vertical section of the outlet vent. Dose location 4 (outer) is extended along the top plane of the top lid, located radially above the overpack outer diameter.
- The "Fuel Gammas" category includes gammas from the spent fuel and ^{60}Co from the spacer grids.

Table 5.1.7

DOSE RATES AT ONE METER FROM HI-STORM FW OVERPACK
FOR NORMAL CONDITIONS
MPC-37
BURNUP AND COOLING TIME
45,000 MWD/MTU AND 4.5-YEAR COOLING

Dose Point Location	**Fuel Gammas (mrem/hr)**	**(n,γ) Gammas (mrem/hr)**	**^{60}Co Gammas (mrem/hr)**	**Neutrons (mrem/hr)**	**Totals (mrem/hr)**	**Totals with BPRAs (mrem/hr)**
1	57	1	4	1	62	62
2	75	1	1	1	77	78
3	6	<1	5	<1	13	15
4 (center)	0.6	0.3	1.0	0.2	2.1	2.7

Notes:
- Refer to Figure 5.1.1 for dose locations.
- Values are rounded to nearest integer where appropriate.
- The "Fuel Gammas" category includes gammas from the spent fuel, ^{60}Co from the spacer grids, and ^{60}Co from the BPRAs in the active fuel region.

Table 5.1.8

DOSE RATES AT ONE METER FROM HI-STORM FW OVERPACK
FOR NORMAL CONDITIONS
MPC-89
BURNUP AND COOLING TIME
45,000 MWD/MTU AND 5-YEAR COOLING

Dose Point Location	**Fuel Gammas (mrem/hr)**	**(n,γ) Gammas (mrem/hr)**	**^{60}Co Gammas (mrem/hr)**	**Neutrons (mrem/hr)**	**Totals (mrem/hr)**
1	38	<1	7	<1	47
2	47	<1	<1	<1	50
3	3	<1	5	<1	10
4 (center)	0.2	0.2	1	0.1	2

Notes:
- Refer to Figure 5.1.1 for dose locations.
- Values are rounded to nearest integer where appropriate.
- The "Fuel Gammas" category includes gammas from the spent fuel and ^{60}Co from the spacer girds.

Table 5.1.9

DOSE FROM HI-TRAC VW WITH MPC-37
FOR ACCIDENT CONDITIONS
AT 100 METERS
65,000 MWD/MTU AND 8-YEAR COOLING

Dose Point Location	**Dose Rate (rem/hr)**	**Accident Duration (days)**	**Total Dose (rem)**	**Regulatory Limit (rem)**	**Time to Reach Regulatory Limit (days)**
2 (Accident Condition)	2.3E-3	30	1.66	5	90

Notes:
- Refer to Figure 5.1.2 for dose locations.
- Values are rounded to nearest integer where appropriate.
- Dose rates used to evaluated “ Total Dose (rem)” are from Table 5.1.4
- Regulatory Limit is from 10CFR72.106.

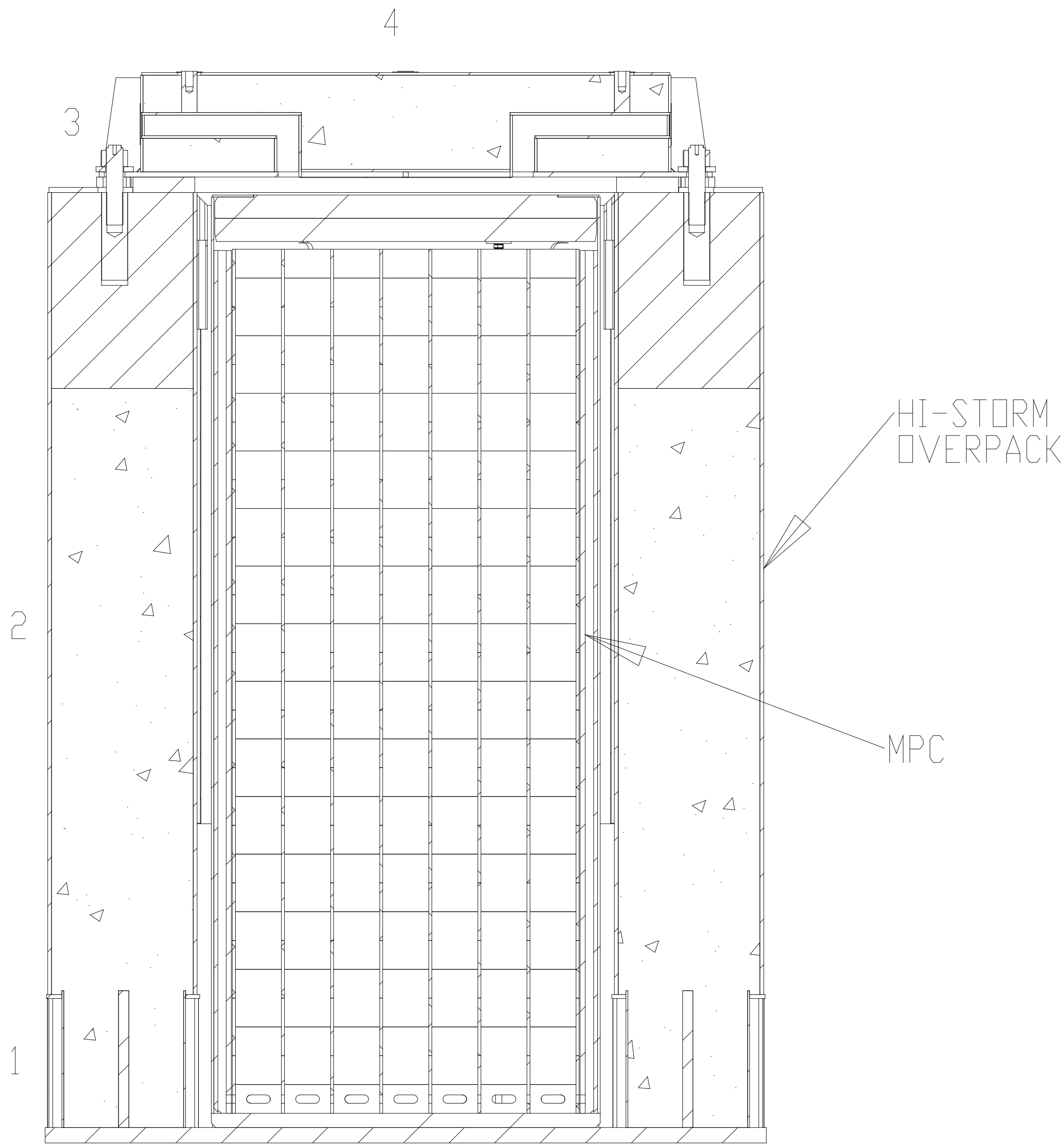

Figure 5.1.1

CROSS SECTION ELEVATION VIEW OF HI-STORM FW OVERPACK WITH DOSE POINT LOCATIONS

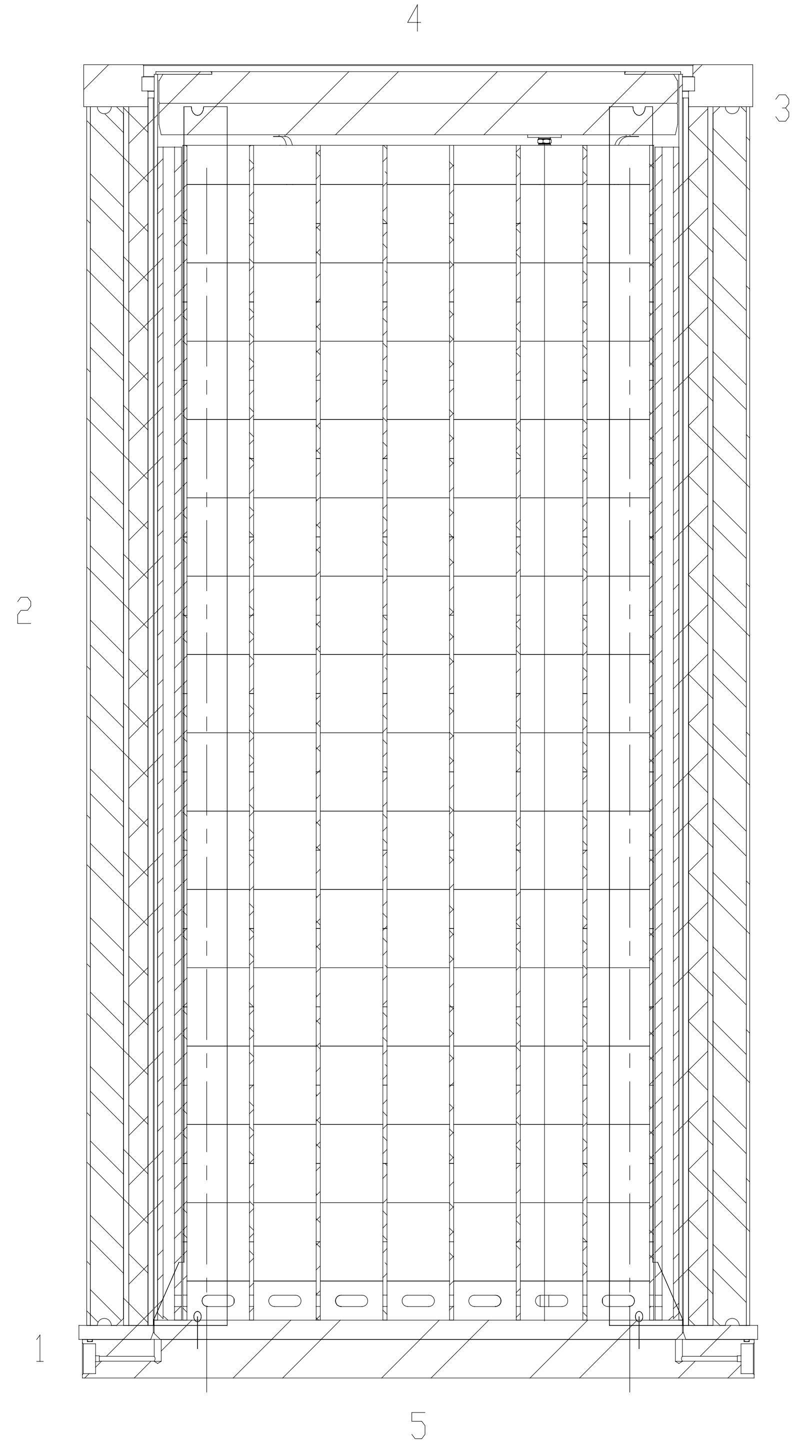

Figure 5.1.2
CROSS SECTION ELEVATION VIEW OF HI-TRAC VW TRANSFER CASK WITH DOSE POINT LOCATIONS

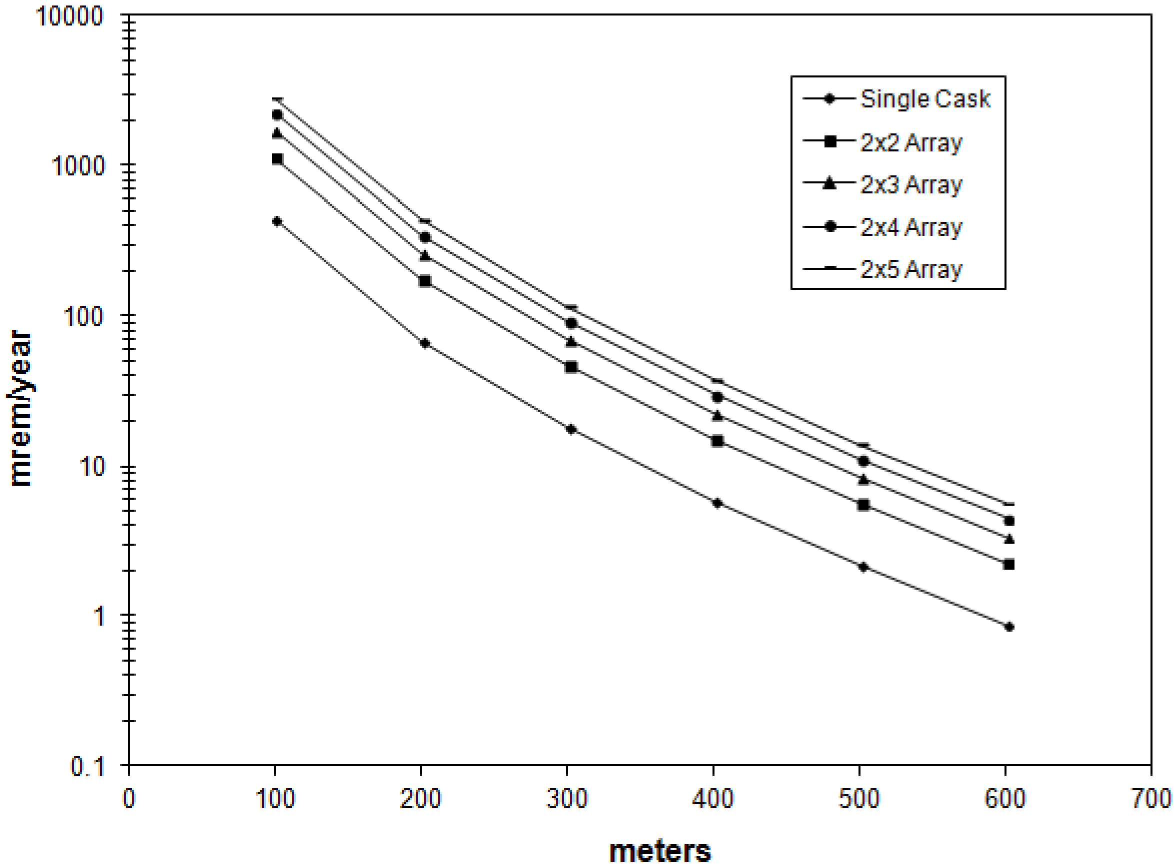

Figure 5.1.3

ANNUAL DOSE VERSUS DISTANCE FOR VARIOUS CONFIGURATIONS OF THE MPC-37 FOR 45,000 MWD/MTU AND 4.5 YEAR COOLING (8760 HOUR OCCUPANCY ASSUMED)

5.2 SOURCE SPECIFICATION

The neutron and gamma source terms, decay heat values, and quantities of radionuclides available for release were calculated with the SAS2H and ORIGEN-S modules of the SCALE 5 system [5.1.2, 5.1.3]. SAS2H has been extensively compared to experimental isotopic validations and decay heat measurements. References [5.2.8] through [5.2.12] and [5.2.15] present isotopic comparisons for PWR and BWR fuels for burnups ranging to 47 GWD/MTU and reference [5.2.13] presents results for BWR measurements to a burnup of 57 GWD/MTU. A comparison of calculated and measured decays heats is presented in reference [5.2.14]. All of these studies indicate good agreement between SAS2H and measured data. Additional comparisons of calculated values and measured data are being performed by various institutions for high burnup PWR and BWR fuel. These new results, when published, are expected to further confirm the validity of SAS2H for the analysis of PWR and BWR fuel.

Sample input files for SAS2H and ORIGEN-S are provided in Appendix 5.A. The gamma source term is actually comprised of three distinct sources. The first is a gamma source term from the active fuel region due to decay of fission products. The second source term is from ^{60}Co activity of the stainless steel structural material in the fuel element above and below the active fuel region. The third source is from (n,γ) reactions described below.

A description of the design basis fuel for the source term calculations is provided in Table 5.2.1. Subsection 5.2.5 discusses, in detail, the determination of the design basis fuel assemblies.

In performing the SAS2H and ORIGEN-S calculations, a single full power cycle was used to achieve the desired burnup. This assumption, in conjunction with the above-average specific powers listed in Table 5.2.1 resulted in conservative source term calculations.

5.2.1 Gamma Source

Tables 5.2.2 through 5.2.5 provide the gamma source in MeV/s and photons/s as calculated with SAS2H and ORIGEN-S for the design basis zircaloy clad fuel at the burnups and cooling times used for normal and accident conditions.

Previous analyses were performed for the HI-STORM 100 system to determine the dose contribution from gammas as a function of energy [5.2.17]. The results of these analyses have revealed that, due to the magnitude of the gamma source at lower energies, photons with energies as low as 0.45 MeV must be included in the shielding analysis, but photons with energies below 0.45 MeV are too weak to penetrate the HI-STORM overpack or HI-TRAC. The effect of gammas with energies above 3.0 MeV, on the other hand, was found to be insignificant. This is due to the fact that the source of gammas in this range (i.e., above 3.0 MeV) is extremely low. Therefore, all photons with energies in the range of 0.45 to 3.0 MeV are included in the shielding calculations.

The primary source of activity in the non-fuel regions of an assembly arises from the activation of ^{59}Co to ^{60}Co. The primary source of ^{59}Co in a fuel assembly is impurities in the steel structural material above and below the fuel. The zircaloy in these regions is neglected since it does not have a significant ^{59}Co impurity level. Reference [5.2.2] indicates that the impurity level in steel is 800 ppm or 0.8 gm/kg. Therefore, inconel and stainless steel in the non-fuel regions are both assumed to have the same 0.8 gm/kg impurity level.

Some of the PWR fuel assembly designs (B&W and WE 15x15) utilized inconel in-core grid spacers while other PWR fuel designs use zircaloy in-core grid spacers. In the mid 1980s, the fuel assembly designs using inconel in-core grid spacers were altered to use zircaloy in-core grid spacers. Since both designs may be loaded into the HI-STORM FW system, the gamma source for the PWR zircaloy clad fuel assembly includes the activation of the in-core grid spacers. Although BWR assembly grid spacers are zircaloy, some assembly designs have inconel springs in conjunction with the grid spacers. The gamma source for the BWR zircaloy clad fuel assembly includes the activation of these springs associated with the grid spacers.

The non-fuel data listed in Table 5.2.1 were taken from References [5.2.2], [5.2.4], and [5.2.5]. As stated above, a Cobalt-59 impurity level of 0.8 gm/kg was used for both inconel and stainless steel. Therefore, there is little distinction between stainless steel and inconel in the source term generation and since the shielding characteristics are similar, stainless steel was used in the MCNP calculations instead of inconel. The BWR masses for an 8x8 fuel assembly were used. These masses are also appropriate for the 10x10 assembly since the masses of the non-fuel hardware from a 10x10 and an 8x8 are approximately the same. The masses listed are those of the steel components. The zircaloy in these regions was not included because zircaloy does not produce significant activation.

The masses in Table 5.2.1 were used to calculate a ^{59}Co impurity level in the fuel assembly material. The grams of impurity were then used in ORIGEN-S to calculate a ^{60}Co activity level for the desired burnup and decay time. The methodology used to determine the activation level was developed from Reference [5.2.3] and is described here.

1. The activity of the ^{60}Co is calculated using ORIGEN-S. The flux used in the calculation was the in-core fuel region flux at full power.

2. The activity calculated in Step 1 for the region of interest was modified by the appropriate scaling factors listed in Table 5.2.6. These scaling factors were taken from Reference [5.2.3].

Tables 5.2.7 through 5.2.10 provide the ^{60}Co activity utilized in the shielding calculations for normal and accident conditions for the non-fuel regions of the assemblies in the MPC-37 and the MPC-89.

In addition to the two sources already mentioned, a third source arises from (n,γ) reactions in the material of the MPC and the overpack. This source of photons is properly accounted for in

MCNP when a neutron calculation is performed in a coupled neutron-gamma mode.

5.2.2 Neutron Source

It is well known that the neutron source strength increases as enrichment decreases, for a constant burnup and decay time. This is due to the increase in Pu content in the fuel, which increases the inventory of other transuranium nuclides such as Cm. The gamma source also varies with enrichment, although only slightly. Because of this effect and in order to obtain conservative source terms, low initial fuel enrichments of 3.2 and 3.6 wt% were chosen for the BWR and PWR design basis fuel assemblies under normal conditions, respectively. For the accident conditions, a fuel enrichment of 4.8 wt% was chosen to accommodate the higher burnups of the selected source terms (see Table 5.0.2) in accordance with Table 5.2.24 of reference [5.2.17].

The neutron source calculated for the design basis fuel assemblies for the MPCs and the design basis fuel are listed in Tables 5.2.11 through 5.2.14 in neutrons/s for the selected burnup and cooling times used in the shielding evaluations for normal and accident conditions. The neutron spectrum is generated in ORIGEN-S.

5.2.3 Non-Fuel Hardware

Burnable poison rod assemblies (BPRAs), thimble plug devices (TPDs), control rod assemblies (CRAs), and axial power shaping rods (APSRs) are permitted for storage in the HI-STORM FW system as an integral part of a PWR fuel assembly. BPRAs and TPDs may be stored in any fuel location while CRAs and APSRs are restricted as specified in Subsection 2.1.

5.2.3.1 BPRAs and TPDs

Burnable poison rod assemblies (BPRA) (including wet annular burnable absorbers) and thimble plug devices (TPD) (including orifice rod assemblies, guide tube plugs, and water displacement guide tube plugs) are an integral, yet removable, part of a large portion of PWR fuel. The TPDs are not used in all assemblies in a reactor core but are reused from cycle to cycle. Therefore, these devices can achieve very high burnups. In contrast, BPRAs are burned with a fuel assembly in core and are not reused. In fact, many BPRAs are removed after one or two cycles before the fuel assembly is discharged. Therefore, the achieved burnup for BPRAs is not significantly different from that of a fuel assembly. Vibration suppressor inserts are considered to be in the same category as BPRAs for the purposes of the analysis in this chapter since these devices have the same configuration (long non-absorbing thimbles which extend into the active fuel region) as a BPRA without the burnable poison.

TPDs are made of stainless steel and contain a small amount of inconel. These devices extend down into the plenum region of the fuel assembly but typically do not extend into the active fuel region. Since these devices are made of stainless steel, there is a significant amount of cobalt-60 produced during irradiation. This is the only significant radiation source from the activation of steel and inconel.

BPRAs are made of stainless steel in the region above the active fuel zone and may contain a small amount of inconel in this region. Within the active fuel zone the BPRAs may contain 2-24 rodlets which are burnable absorbers clad in either zircaloy or stainless steel. The stainless steel clad BPRAs create a significant radiation source (Co-60) while the zircaloy clad BPRAs create a negligible radiation source. Therefore, the stainless steel clad BPRAs are bounding.

SAS2H and ORIGEN-S were used to calculate a radiation source term for the TPDs and BPRAs. In the ORIGEN-S calculations the cobalt-59 impurity level was conservatively assumed to be 0.8 gm/kg for stainless steel and 4.7 gm/kg for inconel. These calculations were performed by irradiating the appropriate mass of steel and inconel using the flux calculated for the design basis W 17x17 fuel assembly. The mass of material in the regions above the active fuel zone was scaled by the appropriate scaling factors listed in Table 5.2.6 in order to account for the reduced flux levels above the fuel assembly. The total curies of cobalt were calculated for the TPDs and BPRAs as a function of burnup and cooling time.

Since the HI-STORM FW cask system is designed to store many varieties of PWR fuel, a representative TPD and BPRA had to be determined for the purposes of the analysis. This was accomplished in the HI-STORM 100 FSAR [5.2.17] by analyzing all of the BPRAs and TPDs (Westinghouse and B&W 14x14 through 17x17) found in references [5.2.5] and [5.2.7] to determine the TPD and BPRA which produced the highest Cobalt-60 source term and decay heat for a specific burnup and cooling time. The TPD was determined to be the Westinghouse 17x17 guide tube plug and the BPRA was actually determined by combining the higher masses of the Westinghouse 17x17 and 15x15 BPRAs into a single hypothetical BPRA. The masses of these devices are listed in Table 5.2.15.

Table 5.2.16 shows the curies of Co-60 that were calculated for BPRAs and TPDs in each region of the fuel assembly (e.g. incore, plenum, top). A burnup and cooling time, separate from the fuel assemblies, is used for BPRAs and TPDs. Table 2.1.25 of the HI-STORM 100 [5.2.17] lists the allowable burnups and cooling times for non-fuel hardware that corresponds to the BPRA. These burnup and cooling times assure that the Co-60 activity remains below the levels specified above. For specific site boundary evaluations, these levels/values can be used if they are bounding. Alternatively, more realistic values can be used.

The HI-STORM 100 [5.2.17] presents dose rates for both BPRAs and TPDs. The results indicate that BPRAs are bounding, therefore all dose rates in this chapter will contain a BPRA in every PWR fuel location. However, Section 5.4 also contains a quantitative dose rates comparison from BPRAs and TPDs to validate this approach. Subsection 5.4.4 discusses the increase in the cask dose rates due to the insertion of BPRAs into fuel assemblies.

5.2.3.2 CRAs and APSRs

Control rod assemblies (CRAs) (including control element assemblies and rod cluster control assemblies) and axial power shaping rod assemblies (APSRs) are an integral portion of a PWR fuel assembly. These devices are utilized for many years (upwards of 20 years) prior to discharge into the spent fuel pool. The manner in which the CRAs are utilized vary from plant to plant. Some utilities maintain the CRAs fully withdrawn during normal operation while others may operate with a bank of rods partially inserted (approximately 10%) during normal operation. Even when fully withdrawn, the ends of the CRAs are present in the upper portion of the fuel assembly since they are never fully removed from the fuel assembly during operation. The result of the different operating styles is a variation in the source term for the CRAs. In all cases, however, only the lower portion of the CRAs will be significantly activated. Therefore, when the CRAs are stored with the PWR fuel assembly, the activated portion of the CRAs will be in the lower portion of the cask. CRAs are fabricated of various materials. The cladding is typically stainless steel, although inconel has been used. The absorber can be a single material or a combination of materials. AgInCd is possibly the most common absorber although B_4C in aluminum is used, and hafnium has also been used. AgInCd produces a noticeable source term in the 0.3-1.0 MeV range due to the activation of Ag. The source term from the other absorbers is negligible, therefore the AgInCd CRAs are the bounding CRAs.

APSRs are used to flatten the power distribution during normal operation and as a result these devices achieve a considerably higher activation than CRAs. There are two types of B&W stainless steel clad APSRs: gray and black. According to reference [5.2.5], the black APSRs have 36 inches of AgInCd as the absorber while the gray ones use 63 inches of inconel as the absorber. Because of the cobalt-60 source from the activation of inconel, the gray APSRs produce a higher source term than the black APSRs and therefore are the bounding APSR.

Since the level of activation of CRAs and APSRs can vary, the quantity that can be stored in an MPC is being limited. These devices are required to be stored in the locations as outlined in Subsection 2.1.

Subsection 5.4.4 discusses the effect on dose rate of the insertion of APSRs or CRAs into fuel assemblies.

5.2.4 Choice of Design Basis Assembly

The Westinghouse 17x17 and GE 10x10 assemblies were selected as design basis assemblies since they are widely used throughout the industry. Site specific shielding evaluations should verify that those assemblies and assembly parameters are appropriate for the site-specific analyses.

5.2.5 Decay Heat Loads and Allowable Burnup and Cooling Times

Subsection 2.1 describes the MPC maximum decay heat limits per assembly. The allowable burnup and cooling time limits are derived based on the allowable decay heat limits.

5.2.6 Fuel Assembly Neutron Sources

Neutron source assemblies (NSAs) are used in reactors for startup. There are different types of neutron sources (e.g. californium, americium-beryllium, plutonium-beryllium, polonium-beryllium, antimony-beryllium). These neutron sources are typically inserted into the water rod of a fuel assembly and are usually removable.

During in-core operations, the stainless steel and inconel portions of the NSAs become activated, producing a significant amount of Co-60. A detailed discussion about NSAs is provided in reference [5.2.17], where it is concluded that activation from NSAs are bounded by activation from BPRAs.

For ease of implementation in the CoC, the restriction concerning the number of NSAs is being applied to all types of NSAs. In addition, conservatively NSAs are required to be stored in the inner region of the MPC basket as specified in Subsection 2.1. Further limitations allow for only one NSA to be stored in the MPC-37 (see Table 2.1.1).

Table 5.2.1 DESCRIPTION OF DESIGN BASIS CLAD FUEL		
	PWR	**BWR**
Assembly type/class	WE 17×17	GE 10×10
Active fuel length (in.)	144	144
No. of fuel rods	264	92
Rod pitch (in.)	0.496	0.51
Cladding material	Zircaloy-4	Zircaloy-2
Rod diameter (in.)	0.374	0.404
Cladding thickness (in.)	0.0225	0.026
Pellet diameter (in.)	0.3232	0.345
Pellet material	UO_2	UO_2
Pellet density (gm/cc)	10.412 (95% of theoretical)	10.522 (96% of theoretical)
Enrichment (w/o ^{235}U)	3.6	3.2
Specific power (MW/MTU)	43.48	30
Weight of UO_2 (kg)[††]	532.150	213.531
Weight of U (kg)[††]	469.144	188.249
No. of Water Rods/ Guide Tubes	25	2
Water Rod/ Guide Tube O.D. (in.)	0.474	0.98
Water Rod/ Guide Tube Thickness (in.)	0.016	0.03

[††] Derived from parameters in this table.

Table 5.2.1 (continued)

DESCRIPTION OF DESIGN BASIS FUEL

	PWR	BWR
Lower End Fitting (kg)	5.9 (steel)	4.8 (steel)
Gas Plenum Springs (kg)	1.150 (steel)	1.1 (steel)
Gas Plenum Spacer (kg)	0.793 (inconel) 0.841 (steel)	N/A
Expansion Springs (kg)	N/A	0.4 (steel)
Upper End Fitting (kg)	6.89 (steel) 0.96 (inconel)	2.0 (steel)
Handle (kg)	N/A	0.5 (steel)
Incore Grid Spacers (kg)	4.9 (inconel)	0.33 (inconel springs)

Table 5.2.2 CALCULATED MPC-37 PWR FUEL GAMMA SOURCE PER ASSEMBLY FOR DESIGN BASIS BURNUP AND COOLING TIME FOR NORMAL CONDITIONS			
Lower Energy	**Upper Energy**	**45,000 MWD/MTU 4.5-Year Cooling**	
(MeV)	(MeV)	(MeV/s)	(Photons/s)
0.45	0.7	2.11E+15	3.68E+15
0.7	1.0	7.67E+14	9.02E+14
1.0	1.5	1.74E+14	1.39E+14
1.5	2.0	1.45E+13	8.30E+12
2.0	2.5	1.01E+13	4.47E+12
2.5	3.0	4.05E+11	1.47E+11
Total		3.08E+15	4.73E+15

Table 5.2.3 CALCULATED MPC-37 PWR FUEL GAMMA SOURCE PER ASSEMBLY FOR BURNUP AND COOLING TIME FOR ACCIDENT CONDITIONS			
Lower Energy	**Upper Energy**	**65,000 MWD/MTU 8-Year Cooling**	
(MeV)	(MeV)	(MeV/s)	(Photons/s)
0.45	0.7	2.05E+15	3.56E+15
0.7	1.0	4.16E+14	4.89E+14
1.0	1.5	1.30E+14	1.04E+14
1.5	2.0	8.66E+12	4.95E+12
2.0	2.5	6.46E+11	2.87E+11
2.5	3.0	4.49E+10	1.63E+10
Total		2.60E+15	4.16E+15

Table 5.2.4 CALCULATED MPC-89 BWR FUEL GAMMA SOURCE PER ASSEMBLY FOR DESIGN BASIS BURNUP AND COOLING TIME FOR NORMAL CONDITIONS			
Lower Energy	**Upper Energy**	**45,000 MWD/MTU 5-Year Cooling**	
(MeV)	(MeV)	(MeV/s)	(Photons/s)
0.45	0.7	7.52E+14	1.31E+15
0.7	1.0	2.40E+14	2.82E+14
1.0	1.5	5.53E+13	4.42E+13
1.5	2.0	4.15E+12	2.37E+12
2.0	2.5	2.02E+12	8.97E+11
2.5	3.0	9.74E+10	3.54E+10
Total		1.05E+15	2.04E+15

Table 5.2.5 CALCULATED MPC-89 BWR FUEL GAMMA SOURCE PER ASSEMBLY FOR BURNUP AND COOLING TIME FOR ACCIDENT CONDITIONS			
Lower Energy	**Upper Energy**	**65,000 MWD/MTU 10-Year Cooling**	
(MeV)	(MeV)	(MeV/s)	(Photons/s)
0.45	0.7	6.98E+14	1.21E+15
0.7	1.0	8.37E+13	9.85E+13
1.0	1.5	3.50E+13	2.80E+13
1.5	2.0	2.52E+12	1.44E+12
2.0	2.5	4.49E+10	2.00E+10
2.5	3.0	3.90E+09	1.42E+09
Total		8.19E+14	1.34E+15

Table 5.2.6

SCALING FACTORS USED IN CALCULATING THE ^{60}Co SOURCE

Region	**PWR**	**BWR**
Handle	N/A	0.05
Upper End Fitting	0.1	0.1
Gas Plenum Spacer	0.1	N/A
Expansion Springs	N/A	0.1
Gas Plenum Springs	0.2	0.2
Incore Grid Spacer	1.0	1.0
Lower End Fitting	0.2	0.15

Table 5.2.7

CALCULATED MPC-37 ^{60}Co SOURCE PER ASSEMBLY FOR DESIGN BASIS FUEL AT DESIGN BASIS BURNUP AND COOLING TIME FOR NORMAL CONDITIONS

Location	**45,000 MWD/MTU and 4.5-Year Cooling (curies)**
Lower End Fitting	86.02
Gas Plenum Springs	16.77
Gas Plenum Spacer	11.91
Expansion Springs	NA
Incore Grid Spacers	357.19
Upper End Fitting	57.22
Handle	NA

Table 5.2.8

CALCULATED MPC-37 ^{60}Co SOURCE PER ASSEMBLY FOR DESIGN BASIS FUEL AT BURNUP AND COOLING TIME FOR ACCIDENT CONDITIONS

Location	**65,000 MWD/MTU and 8-Year Cooling (curies)**
Lower End Fitting	64.89
Gas Plenum Springs	12.65
Gas Plenum Spacer	8.99
Expansion Springs	NA
Incore Grid Spacers	269.46
Upper End Fitting	43.17
Handle	NA

Table 5.2.9

CALCULATED MPC-89 ^{60}Co SOURCE PER ASSEMBLY FOR DESIGN BASIS FUEL AT DESIGN BASIS BURNUP AND COOLING TIME FOR NORMAL CONDITIONS

Location	**45,000 MWD/MTU and 5-Year Cooling (curies)**
Lower End Fitting	158.66
Gas Plenum Springs	48.48
Gas Plenum Spacer	N/A
Expansion Springs	8.81
Grid Spacer Springs	72.72
Upper End Fitting	44.07
Handle	5.51

Table 5.2.10

CALCULATED MPC-89 ^{60}Co SOURCE PER ASSEMBLY FOR DESIGN BASIS FUEL AT BURNUP AND COOLING TIME FOR ACCIDENT CONDITIONS

Location	**65,000 MWD/MTU and 10-Year Cooling (curies)**
Lower End Fitting	90.17
Gas Plenum Springs	27.55
Gas Plenum Spacer	N/A
Expansion Springs	5.01
Grid Spacer Springs	41.33
Upper End Fitting	25.05
Handle	3.13

Table 5.2.11

CALCULATED MPC-37 PWR NEUTRON SOURCE PER ASSEMBLY
FOR 45,000 MWD/MTU BURNUP AND 4.5 YEAR COOLING

Lower Energy (MeV)	Upper Energy (MeV)	45,000 MWD/MTU 4.5-Year Cooling (Neutrons/s)
1.0e-01	4.0e-01	3.05E+07
4.0e-01	9.0e-01	6.64E+07
9.0e-01	1.4	6.63E+07
1.4	1.85	5.30E+07
1.85	3.0	9.88E+07
3.0	6.43	8.97E+07
6.43	20.0	8.56E+06
Totals		4.13E+08

Table 5.2.12

CALCULATED MPC-37 PWR NEUTRON SOURCE PER ASSEMBLY
FOR 65,000 MWD/MTU BURNUP AND 8 YEAR COOLING

Lower Energy (MeV)	Upper Energy (MeV)	65,000 MWD/MTU 8-Year Cooling (Neutrons/s)
1.0e-01	4.0e-01	6.80E+07
4.0e-01	9.0e-01	1.48E+08
9.0e-01	1.4	1.47E+08
1.4	1.85	1.17E+08
1.85	3.0	2.18E+08
3.0	6.43	1.98E+08
6.43	20.0	1.89E+07
Totals		9.16E+08

Table 5.2.13

CALCULATED MPC-89 BWR NEUTRON SOURCE
PER ASSEMBLY
FOR DESIGN BASIS FUEL
FOR 45,000 MWD/MTU BURNUP AND 5 YEAR COOLING

Lower Energy (MeV)	**Upper Energy (MeV)**	**45,000 MWD/MTU 5-Year Cooling (Neutrons/s)**
1.0e-01	4.0e-01	1.37E+07
4.0e-01	9.0e-01	2.99E+07
9.0e-01	1.4	2.99E+07
1.4	1.85	2.38E+07
1.85	3.0	4.44E+07
3.0	6.43	4.03E+07
6.43	20.0	3.86E+06
Totals		1.86E+08

Table 5.2.14

CALCULATED MPC-89 BWR NEUTRON SOURCE
PER ASSEMBLY
FOR DESIGN BASIS FUEL
FOR 65,000 MWD/MTU BURNUP AND 10 YEAR COOLING

Lower Energy (MeV)	**Upper Energy (MeV)**	**65,000 MWD/MTU 10-Year Cooling (Neutrons/s)**
1.0e-01	4.0e-01	2.40E+07
4.0e-01	9.0e-01	5.22E+07
9.0e-01	1.4	5.20E+07
1.4	1.85	4.15E+07
1.85	3.0	7.71E+07
3.0	6.43	7.00E+07
6.43	20.0	6.68E+06
Totals		3.24E+08

Table 5.2.15 DESCRIPTION OF DESIGN BASIS BURNABLE POISON ROD ASSEMBLY AND THIMBLE PLUG DEVICE		
Region	**BPRA**	**TPD**
Upper End Fitting (kg of steel)	2.62	2.3
Upper End Fitting (kg of inconel)	0.42	0.42
Gas Plenum Spacer (kg of steel)	0.77488	1.71008
Gas Plenum Springs (kg of steel)	0.67512	1.48992
In-core (kg of steel)	13.2	N/A

Table 5.2.16 DESIGN BASIS COBALT-60 ACTIVITIES FOR BURNABLE POISON ROD ASSEMBLIES AND THIMBLE PLUG DEVICES		
Region	**BPRA**	**TPD**
Upper End Fitting (curies Co-60)	32.7	25.21
Gas Plenum Spacer (curies Co-60)	5.0	9.04
Gas Plenum Springs (curies Co-60)	8.9	15.75
In-core (curies Co-60)	848.4	N/A

5.3 MODEL SPECIFICATIONS

The shielding analysis of the HI-STORM FW system was performed with MCNP5 [5.1.1]. MCNP is a Monte Carlo transport code that offers a full three-dimensional combinatorial geometry modeling capability including such complex surfaces as cones and tori. This means that no gross approximations were required to represent the HI-STORM FW system, including the HI-TRAC transfer casks, in the shielding analysis. A sample input file for MCNP is provided in Appendix 5.A.

As discussed in Subsection 5.1.1, off-normal conditions do not have any implications for the shielding analysis. Therefore, the MCNP models and results developed for the normal conditions also represent the off-normal conditions. Subsection 5.1.2 discussed the accident conditions and stated that the only accident that would impact the shielding analysis would be a loss of the neutron shield (water) in the HI-TRAC. Therefore, the MCNP model of the normal HI-TRAC condition has the neutron shield in place while the accident condition replaces the neutron shield with void. Subsection 5.1.2 also mentioned that there is no credible accident scenario that would impact the HI-STORM shielding analysis. Therefore, models and results for the normal and accident conditions are identical for the HI-STORM overpack.

5.3.1 Description of the Radial and Axial Shielding Configuration

Chapter 1 provides the drawings that describe the HI-STORM FW system, including the HI-TRAC transfer cask. These drawings, using nominal dimensions, were used to create the MCNP models used in the radiation transport calculations. Modeling deviations from these drawings are discussed below. Figures 5.3.1 and 5.3.2, as well as Figures 5.3.12 and 5.3.13, show cross sectional views of the HI-STORM FW overpack, MPCs, and basket cells as they are modeled in MCNP. Figures 5.3.1 and 5.3.2 were created in VISED and are drawn to scale. The inlet and outlet vents were modeled explicitly, therefore, streaming through these components is accounted for in the calculations of the dose adjacent to the overpack and at 1 meter. Figures 5.3.3 and 5.3.4 show a cross sectional view of the HI-TRAC VW with the MPC-37 and MPC-89, respectively, as it was modeled in MCNP. These figures were created in VISED and are drawn to scale.

Figure 5.3.5 shows a cross sectional view of the HI-STORM FW overpack with the as-modeled thickness of the various materials.

Figure 5.3.6 shows the axial representation of the HI-STORM FW overpack with the various as-modeled dimensions indicated.

Figure 5.3.7 shows axial cross-sectional views of the HI-TRAC VW transfer casks with the as-modeled dimensions and materials specified. Figures 5.3.8 and 5.3.9 shows fully labeled radial cross-sectional view of the HI-TRAC VW transfer casks and each of the MPCs.

Calculations were performed for the HI-STORM 100 [5.2.17] to determine the acceptability of homogenizing the fuel assembly versus explicit modeling. Based on these calculations it was concluded that it is acceptable to homogenize the fuel assembly without loss of accuracy. The width of the PWR and BWR homogenized fuel assembly is equal to 17 times the pitch and 10 times the pitch, respectively. Homogenization results in a noticeable decrease in run time.

Several conservative approximations were made in modeling the MPC. The conservative approximations are listed below.

1. The fuel shims are not modeled because they are not needed on all fuel assembly types. However, most PWR fuel assemblies will have fuel shims. The fuel shim length for the design basis fuel assembly type determines the positioning of the fuel assembly for the shielding analysis. This is conservative since it removes steel that would provide a small amount of additional shielding.

2. The MPC basket supports are not modeled. This is conservative since it removes material that would provide a small increase in shielding.

The zircaloy flow channels are included in the modeling of the BWR assemblies. The expected impact of this assumption on the dose rates is insignificant. Additionally, site specific analysis should consider site specific fuel characteristics as applicable.

5.3.1.1 Fuel Configuration

As described earlier, the active fuel region is modeled as a homogenous zone. The end fittings and the plenum regions are also modeled as homogenous regions of steel. The masses of steel used in these regions are shown in Table 5.2.1. The axial description of the design basis fuel assemblies is provided in Table 5.3.1. Figures 5.3.10 and 5.3.11 graphically depict the location of the PWR and BWR fuel assemblies within the HI-STORM FW system. The axial locations of the basket, inlet vents, and outlet vents are shown in these figures.

5.3.1.2 Streaming Considerations

The MCNP model of the HI-STORM overpack completely describes the inlet and outlet vents, thereby properly accounting for their streaming effect. Further, the top lid is properly modeled with its reduced diameter, which accounts for higher localized dose rates on the top surface of the HI-STORM.

The MCNP model of the HI-TRAC transfer cask accounts for the fins through the HI-TRAC water jacket, as discussed in Subsection 5.4.1, as well as the open annulus.

5.3.2 Regional Densities

Composition and densities of the various materials used in the HI-STORM FW system and HI-TRAC shielding analyses are given in Table 5.3.2. All of the materials and their actual geometries are represented in the MCNP model.

The concrete density shown in Table 5.3.2 is the minimum concrete density analyzed in this chapter. The HI-STORM FW overpacks are designed in such a way that the concrete density in the body of the overpack can be increased to approximately 3.2 g/cm^3 (200 lb/cu-ft). Increasing the density beyond the value in Table 5.3.2 would result in a significant reduction in the dose rates. This may be beneficial based on on-site and off-site ALARA considerations.

The water density inside the MPC corresponds to the maximum allowable water temperature within the MPC. The water density in the water jacket corresponds to the maximum allowable temperature at the maximum allowable pressure. As mentioned, the HI-TRAC transfer cask may be equipped with a water jacket to provide radial neutron shielding. Demineralized water (borated water) will be utilized in the water jacket. To ensure operability for low temperature conditions, ethylene glycol (25% in solution) may be added to reduce the freezing point for low temperature operations. Calculations were performed for the HI-STORM 100 system [5.2.17] to determine the effect of the ethylene glycol on the shielding effectiveness of the radial neutron shield. Based on these calculations, it was concluded that the addition of ethylene glycol (25% in solution) does not reduce the shielding effectiveness of the radial neutron shield.

Subsections 4.4 and 4.5 demonstrate that all materials used in the HI-STORM and HI-TRAC remain below their design temperatures as specified in Table 2.2.3 during all normal conditions. Therefore, the shielding analysis does not address changes in the material density or composition as a result of temperature changes.

Chapter 11 discusses the effect of the various accident conditions on the temperatures of the shielding materials and the resultant impact on their shielding effectiveness. As stated in Subsection 5.1.2, there is only one accident that has any significant impact on the shielding configuration. This accident is the loss of the neutron shield (water) in the HI-TRAC as a result of fire or other damage. The change in the neutron shield was conservatively analyzed by assuming that the entire volume of the liquid neutron shield was replaced by void.

Table 5.3.1

DESCRIPTION OF THE AXIAL MCNP MODEL OF THE FUEL ASSEMBLIES†

Region	Start (in.)	Finish (in.)	Length (in.)	Actual Material	Modeled Material
PWR					
Lower End Fitting	0.0	2.738	2.738	SS304	SS304
Space	2.738	3.738	1.0	zircaloy	void
Fuel	3.738	147.738	144.0	fuel & zircaloy	fuel & zircaloy
Gas Plenum Springs	147.738	151.916	4.178	SS304 & inconel	SS304
Gas Plenum Spacer	151.916	156.095	4.179	SS304 & inconel	SS304
Upper End Fitting	156.095	159.765	3.670	SS304 & inconel	SS304
BWR					
Lower End Fitting	0.0	7.385	7.385	SS304	SS304
Fuel	7.385	151.385	144.0	fuel & zircaloy	fuel & zircaloy
Space	151.385	157.385	6.0	zircaloy	void
Gas Plenum Springs	157.385	166.865	9.48	SS304 & zircaloy	SS304
Expansion Springs	166.865	168.215	1.35	SS304	SS304
Upper End Fitting	168.215	171.555	3.34	SS304	SS304
Handle	171.555	176	4.445	SS304	SS304

† All dimensions start at the bottom of the fuel assembly. The length of the fuel shims must be added to the distances to determine the distance from the top of the MPC baseplate.

Table 5.3.2

COMPOSITION OF THE MATERIALS IN THE HI-STORM FW SYSTEM

Component	Density (g/cm^3)	Elements	Mass Fraction (%)
Metamic-HT†	2.61 (9% B_4C)	**Withheld in Accordance with 10 CFR 2.390**	
SS304	7.94	Cr	19
		Mn	2
		Fe	69.5
		Ni	9.5
Carbon Steel	7.82	C	1.0
		Fe	99.0
Zircaloy	6.55	Zr	98.24
		Sn	1.45
		Fe	0.21
		Cr	0.10

† All B-10 loadings in the Metamic compositions are conservatively lower than the values defined in the Bill of Materials.

Table 5.3.2 (continued)

COMPOSITION OF THE MATERIALS IN THE HI-STORM FW SYSTEM

Component	Density (g/cm^3)	Elements	Mass Fraction (%)
BWR Fuel Region Mixture	4.781 (5.0 wt% U-235)	^{235}U	3.207
		^{238}U	60.935
		O	8.623
		Zr	26.752
		N	0.014
		Cr	0.027
		Fe	0.034
		Sn	0.409
PWR Fuel Region Mixture	3.769 (5.0 wt% U-235)	^{235}U	3.709
		^{238}U	70.474
		O	9.972
		Zr	15.565
		Cr	0.016
		Fe	0.033
		Sn	0.230

Table 5.3.2 (continued)

COMPOSITION OF THE MATERIALS IN THE HI-STORM FW SYSTEM

Component	Density (g/cm^3)	Elements	Mass Fraction (%)
Lower End Fitting (PWR)	1.849	SS304	100
Gas Plenum Springs (PWR)	0.23626	SS304	100
Gas Plenum Spacer (PWR)	0.33559	SS304	100
Upper End Fitting (PWR)	1.8359	SS304	100
Lower End Fitting (BWR)	1.5249	SS304	100
Gas Plenum Springs (BWR)	0.27223	SS304	100
Expansion Springs (BWR)	0.69514	SS304	100
Upper End Fitting (BWR)	1.4049	SS304	100
Handle (BWR)	0.26391	SS304	100
Lead	11.3	Pb	99.9
		Cu	0.08
		Ag	0.02
Water	0.919 (water jacket)	H	11.2
	0.958 (inside MPC)	O	88.8

Table 5.3.2 (continued)

COMPOSITION OF THE MATERIALS IN THE HI-STORM FW SYSTEM

Component	Density (g/cm^3)	Elements	Mass Fraction (%)
Water w/ 2000 ppm	0.958	B-10	0.036
		B-11	0.164
		H	11.17
		O	88.63
Concrete	2.4	H	1.0
		O	53.2
		Si	33.7
		Al	3.4
		Na	2.9
		Ca	4.4
		Fe	1.4

Withheld in Accordance with 10 CFR 2.390

Figure 5.3.1
HI-STORM FW OVERPACK WITH MPC-37 CROSS SECTIONAL VIEW AS MODELED IN MCNP†

† This figure is drawn to scale using VISED.

Withheld in Accordance with 10 CFR 2.390

Figure 5.3.2

HI-STORM FW OVERPACK WITH MPC-89 CROSS SECTIONAL VIEW AS MODELED IN MCNP†

† This figure is drawn to scale using VISED.

Withheld in Accordance with 10 CFR 2.390

Figure 5.3.3

HI-TRAC VW OVERPACK WITH MPC-37 CROSS SECTIONAL VIEW AS MODELED IN MCNP†

† This figure is drawn to scale using VISED.

Withheld in Accordance with 10 CFR 2.390

Figure 5.3.4

HI-TRAC VW OVERPACK WITH MPC-89 CROSS SECTIONAL VIEW AS MODELED IN MCNP†

† This figure is drawn to scale using VISED.

Withheld in Accordance with 10 CFR 2.390

Figure 5.3.5
CROSS SECTION OF HI-STORM FW OVERPACK

Withheld in Accordance with 10 CFR 2.390

Figure 5.3.6
HI-STORM FW OVERPACK CROSS SECTIONAL ELEVATION VIEW

Withheld in Accordance with 10 CFR 2.390

Figure 5.3.7
HI-TRAC VW TRANSFER CASK WITH POOL LID CROSS SECTIONAL ELEVATION VIEW (AS MODELED)

Withheld in Accordance with 10 CFR 2.390

Figure 5.3.8

HI-TRAC VW TRANSFER CASK CROSS SECTIONAL VIEW WITH MPC-37 (AS MODELED)

Withheld in Accordance with 10 CFR 2.390

Figure 5.3.9

HI-TRAC VW TRANSFER CASK CROSS SECTIONAL VIEW WITH MPC-89 (AS MODELED)

Withheld in Accordance with 10 CFR 2.390

Figure 5.3.10

AXIAL LOCATION OF PWR DESIGN BASIS FUEL IN THE HI-STORM FW OVERPACK

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Figure 5.3.11

AXIAL LOCATION OF BWR DESIGN BASIS FUEL IN THE HI-STORM FW OVERPACK

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Figure 5.3.12

CROSS SECTIONAL VIEW OF AN MPC-37 BASKET CELL AS MODELED IN MCNP

Withheld in Accordance with 10 CFR 2.390

Figure 5.3.13

CROSS SECTIONAL VIEW OF AN MPC-89 BASKET CELL AS MODELED IN MCNP

5.4 SHIELDING EVALUATION

The MCNP-5 code was used for all of the shielding analyses [5.1.1]. MCNP is a continuous energy, three-dimensional, coupled neutron-photon-electron Monte Carlo transport code. Continuous energy cross section data are represented with sufficient energy points to permit linear-linear interpolation between points. The individual cross section libraries used for each nuclide are those recommended by the MCNP manual Cross section libraries are based on ENDF/B-V and ENDF/B-VI, except for Sn isotopes where the ENDL92 library is used, and uranium isotopes where LANL/T16 libraries are used. These are the default libraries for the MCNP code version used here [5.1.1]. MCNP has been extensively benchmarked against experimental data by the large user community. References [5.4.2], [5.4.3], and [5.4.4] are three examples of the benchmarking that has been performed.

The energy distribution of the source term, as described earlier, is used explicitly in the MCNP model. A different MCNP calculation is performed for each of the three source terms (neutron, decay gamma, and ^{60}Co). The axial distribution of the fuel source term is described in Table 2.1.5 and Figures 2.1.3 and 2.1.4. The PWR and BWR axial burnup distributions were obtained from References [5.4.5] and [5.4.6], respectively and have previously been utilized in the HI-STORM FSAR [5.2.17]. These axial distributions were obtained from operating plants and are representative of PWR and BWR fuel with burnups greater than 30,000 MWD/MTU. The ^{60}Co source in the hardware was assumed to be uniformly distributed over the appropriate regions.

It has been shown that the neutron source strength varies as the burnup level raised by the power of 4.2. Since this relationship is non-linear and since the burnup in the axial center of a fuel assembly is greater than the average burnup, the neutron source strength in the axial center of the assembly is greater than the relative burnup times the average neutron source strength. In order to account for this effect, the neutron source strength in each of the 10 axial nodes listed in Table 2.1.5 was determined by multiplying the average source strength by the relative burnup level raised to the power of 4.2. The peak relative burnups listed in Table 2.1.5 for the PWR and BWR fuels are 1.105 and 1.195 respectively. Using the power of 4.2 relationship results in a 37.6% ($1.105^{4.2}/1.105$) and 76.8% ($1.195^{4.2}/1.195$) increase in the neutron source strength in the peak nodes for the PWR and BWR fuel, respectively. The total neutron source strength increases by 15.6% for the PWR fuel assemblies and 36.9% for the BWR fuel assemblies.

MCNP was used to calculate doses at the various desired locations. MCNP calculates neutron or photon flux and these values can be converted into dose by the use of dose response functions. This is done internally in MCNP and the dose response functions are listed in the input file in Appendix 5.A. The response functions used in these calculations are listed in Table 5.4.1 and were taken from ANSI/ANS 6.1.1, 1977 [5.4.1].

The dose rates at the various locations were calculated with MCNP using a two-step process. The first step was to calculate the dose rate for each dose location per starting particle for each neutron and gamma group in each basket region for each axial and azimuthal dose location. The second step is to multiply the dose rate per starting particle for each energy group and basket location (i.e., tally output/quantity) by the source strength (i.e. particles/sec) in that group and sum the resulting dose rates for all groups and basket locations in each dose location. The normalization of these results and calculation of the total dose rate from neutrons, fuel gammas or Co-60 gammas is performed with the following equation.

$$T_{final} = \sum_{j=1}^{M} \left[\sum_{i=1}^{N} \frac{T_{i,j}}{Fm_i} * F_{i,j} \right]$$ (Equation 5.4.1)

where,

T_{final} = Final dose rate (rem/h) from neutrons, fuel gammas, or Co-60

N = Number of groups (neutrons, fuel gammas) or Number of axial sections (Co-60 gammas)

M = Number of regions in the basket

$T_{i,j}$ = Tally quantity from particles originating in MCNP in group/section i and region j (rem/h)(particles/sec)

$F_{i,j}$ = Fuel Assembly source strength in group i and region j (particles/sec)

Fm_i = Source fraction used in MCNP for group i

Note that dividing by Fm_i (normalization) is necessary to account for the number of MCNP particles that actually start in group i. Also note that T_i is already multiplied by a dose conversion factor in MCNP.

The standard deviations of the various results were statistically combined to determine the standard deviation of the total dose in each dose location. The estimated variance of the total dose rate, S^2_{total}, is the sum of the estimated variances of the individual dose rates S^2_i. The estimated total dose rate, estimated variance, and relative error [5.1.1] are derived according to Equations 5.4.2 through 5.4.5.

$$R_i = \frac{\sqrt{S_i^2}}{T_i}$$ (Equation 5.4.2)

$$S^2_{Total} = \sum_{i=1}^{n} S_i^2 \qquad \text{(Equation 5.4.3)}$$

$$T_{Total} = \sum_{i=1}^{n} T_i \qquad \text{(Equation 5.4.4)}$$

$$R_{Total} = \frac{\sqrt{S^2_{Total}}}{T_{Total}} = \frac{\sqrt{\sum_{i=1}^{n} S_i^2}}{T_{Total}} = \frac{\sqrt{\sum_{i=1}^{n} (R_i \times T_i)^2}}{T_{Total}} \qquad \text{(Equation 5.4.5)}$$

where,

i	=	tally component index
n	=	total number of components
T_{Total}	=	total estimated tally
T_i	=	tally i component
S^2_{Total}	=	total estimated variance
S^2_i	=	variance of the i component
R_i	=	relative error of the i component
R_{Total}	=	total estimated relative error

Note that the two-step approach outlined above allows the accurate consideration of the neutron and gamma source spectrum, and the location of the individual assemblies, since the tallies are calculated in MCNP as a function of the starting energy group and the assembly location, and then in the second step multiplied with the source strength in each group in each location. It is therefore equivalent to a one-step calculation where source terms are directly specified in the MCNP input files, except for the following approximations:

The first approximation is that fuel is modeled as fresh UO_2 fuel (rather than spent fuel) in MCNP, with an upper bound enrichment. The second approximation is related to the axial burnup profile. The profile is modeled by assigning a source probability to each of the 10 axial sections of the active region, based on a representative axial burnup profile [5.2.17]. For fuel gammas, the probability is proportional to the burnup, since the gamma source strength changes essentially linearly with burnup. For neutrons, the probability is proportional to the burnup raised to the power of 4.2, since the neutron source strength is proportional to the burnup raised to about that power [5.4.7]. This is a standard approach that has been previously used in the licensing calculations for the HI-STAR 100 cask [5.4.8] and HI-STORM 100 system [5.2.17].

Tables 5.1.6 and 5.1.7 provide the design basis dose rates adjacent to the HI-STORM overpack during normal conditions for the MPC types in Table 1.0.1. Table 5.1.8 provides the design basis dose rates at one meter from the overpack containing the MPC-37. A detailed discussion of the normal, off-normal, and accident condition dose rates is provided in Subsections 5.1.1 and 5.1.2.

Table 5.4.2 shows the corresponding dose rates adjacent to and one meter away from the HI-TRAC for the fully flooded MPC-37 condition with an empty water-jacket (condition in which the HI-TRAC is removed from the spent fuel pool). Table 5.4.3 shows the dose rates adjacent to and one meter away from the HI-TRAC for the fully flooded MPC-37 condition with the water jacket filled with water (condition in which welding operations are performed). For the conditions involving a fully flooded MPC-37, the internal water level was 5 inches below the MPC lid. These dose rates represent the various conditions of the HI-TRAC during operations. Comparing these results to Table 5.1.1 (dry MPC-37 and HI-TRAC water jacket filled with water) indicates that the dose rates in the upper and lower portions of the HI-TRAC are significantly reduced with water in the MPC.

Table 5.4.4 shows the corresponding dose rates adjacent to and one meter away from the HI-TRAC for the fully flooded MPC-89 condition with an empty water-jacket. Table 5.4.5 shows the dose rates adjacent to and one meter away from the HI-TRAC for the fully flooded MPC-89 condition with the water jacket filled with water. These results demonstrate that the dose rates on contact at the top and bottom of the HI-TRAC VW are somewhat higher in the MPC-89 case than in the MPC-37 case. However, the MPC-37 produces higher dose rates than the MPC-89 at the center of the HI-TRAC, on-contact, and at locations 1 meter away from the HI-TRAC. Therefore, the MPC-37 is used for the exposure calculations in Chapter 11 of the SAR.

The calculations presented herein are using a uniform loading pattern. All MPCs, however, also offer regionalized loading patterns, as mentioned in Section 5.0 and described in Subsection 1.2. These loading patterns authorize fuel of higher decay heat (i.e., higher burnups and shorter cooling times) to be stored in certain regions of the basket. Evaluations have been performed for the HI-STORM 100 [5.2.17] where analysis of the MPC-32 and MPC-68 using the same burnup and cooling times in Region 1 and Region 2. Region 1 contains 38% of total number of assemblies for the MPC-32 and 47% for the MPC-68. The evaluations show that approximately 21% and 27% of the neutron dose at the edge of the water jacket comes from region 1 fuel assemblies in the MPC-32 and MPC-68, respectively. Further, approximately 1% and 2% of the photon dose at the edge of the water jacket comes from region 1 fuel assemblies in the MPC-32 and MPC-68, respectively. These results clearly indicate that the outer fuel assemblies shield almost the entire gamma source from the inner assemblies in the radial direction and a significant percentage of the neutron source. The conclusion from this analysis is that the total dose rate on the external radial surfaces of the cask can be greatly reduced by placing longer cooled and lower burnup fuels on the outside of the basket. Using a uniform loading pattern, rather than employing the regionalized loading scheme, in these HI-STORM FW calculations is therefore acceptable as it produces conservative dose rate values on the radial surfaces.

Since MCNP is a statistical code, there is an uncertainty associated with the calculated values. In MCNP the uncertainty is expressed as the relative error which is defined as the standard deviation of the mean divided by the mean. Therefore, the standard deviation is represented as a percentage of the mean. The relative error for the total dose rates presented in this chapter were typically less than 5% and the relative error for the individual dose components was typically less than 10%.

5.4.1 Streaming Through Radial Steel Fins

The HI-STORM FW overpack and the HI-TRAC VW cask utilize radial steel fins for structural support and cooling. The attenuation of neutrons through steel is substantially less than the attenuation of neutrons through concrete and water. Therefore, it is possible to have neutron streaming through the fins that could result in a localized dose peak. The reverse is true for photons, which would result in a localized reduction in the photon dose.

Analysis of the steel fins in the HI-TRAC has previously been performed in the HI-STORM 100 FSAR [5.2.17] and indicates that neutron streaming is noticeable at the surface of the cask. The neutron dose rate on the surface of the steel fin is somewhat higher than the circumferential average dose rate at that location. The gamma dose rate, however, is slightly lower than the circumferential average dose rate at that location. At one meter from the cask surface there is little difference between the dose rates calculated over the fins compared to the other areas of the water jackets.

These conclusions indicate that localized neutron streaming is noticeable on the surface of the transfer casks. However, at one meter from the surface the streaming has dissipated. Since most HI-TRAC operations will involve personnel moving around the transfer cask at some distance from the cask, only surface average dose rates are reported in this chapter.

5.4.2 Damaged Fuel Post-Accident Shielding Evaluation

The Holtec Generic PWR and BWR DFCs are designed to accommodate any PWR or BWR fuel assembly that can physically fit inside the DFC. Damaged fuel assemblies under normal conditions, for the most part, resemble intact fuel assemblies from a shielding perspective. Under accident conditions, it can not be guaranteed that the damaged fuel assembly will remain intact. As a result, the damaged fuel assembly may begin to resemble fuel debris in its possible configuration after an accident.

Since damaged fuel is identical to intact fuel from a shielding perspective no specific analysis is required for damaged fuel under normal conditions. However, a generic shielding evaluation was previously performed for the HI-STORM 100 [5.2.17] to demonstrate that fuel debris under normal or accident conditions, or damaged fuel in a post-accident configuration, will not result in a significant increase in the dose rates around the 100-ton HI-TRAC. Since the 100-ton HI-TRAC and the HI-TRAC VW are similar in design, the conclusions from the 100-ton HI-RAC evaluations are also applicable to the HI-TRAC VW.

The scenario analyzed to determine the potential change in dose rate as a result of fuel debris or a damaged fuel assembly collapse in the HI-STORM 100 [5.2.17] feature fuel debris or a damaged fuel assembly that has collapsed (which can have a higher average fuel density than an intact fuel assembly). If the damaged fuel assembly would fully or partially collapse, the fuel density in one portion of the assembly would increase and the density in the other portion of the assembly would decrease. The analysis consisted of modeling the fuel assemblies in the damaged fuel locations in the MPC-24 and MPC-68 with a fuel density that was twice the normal fuel density and correspondingly increasing the source rate for these locations by a factor of two. A flat axial power distribution was used which is approximately representative of the source distribution if the top half of an assembly collapsed into the bottom half of the assembly. Increasing the fuel density over the entire fuel length, rather than in the top half or bottom half of the fuel assembly, is conservative and provides the dose rate change in both the top and bottom portion of the cask.

The results for the MPC-24 and MPC-68 calculations [5.2.17] show that the potential effect on the dose rate is not very significant for the storage of damaged fuel and/or fuel debris. This conclusion is further reinforced by the fact that the majority of the significantly damaged fuel assemblies in the spent fuel inventories are older assemblies from the earlier days of nuclear plant operations. Therefore, these assemblies will have a considerably lower burnup and longer cooling times than the assemblies analyzed in this chapter. In addition, since the dose rate change is not significant for the 100-ton HI-TRAC, the dose rate change will not be significant for the HI-TRAC VW or the HI-STORM FW overpacks.

5.4.3 Site Boundary Evaluation

NUREG-1536 [5.2.1] states that detailed calculations need not be presented since SAR Chapter 12 assigns ultimate compliance responsibilities to the site licensee. Therefore, this subsection describes, by example, the general methodology for performing site boundary dose calculations. The site-specific fuel characteristics, burnup, cooling time, and the site characteristics would be factored into the evaluation performed by the licensee.

The methodology of calculating the dose from a single HI-STORM overpack loaded with an MPC and various arrays of loaded HI-STORMs at distances equal to and greater than 100 meters is described in the HI-STORM 100 FSAR [5.2.17]. A back row factor of 0.20 was calculated in

[5.2.17], and utilized herein to calculate dose value C below, based on the results that the dose from the side of the back row of casks is approximately 16 % of the total dose.

The annual dose, assuming 100% occupancy (8760 hours), at 300 meters from a single HI-STORM FW cask is presented in Table 5.4.6 for the design basis burnup and cooling time analyzed.

The annual dose, assuming 8760 hour occupancy, at distance from an array of casks was calculated in three steps.

1. The annual dose from the radiation leaving the side of the HI-STORM FW overpack was calculated at the distance desired. Dose value = A.

2. The annual dose from the radiation leaving the top of the HI-STORM FW overpack was calculated at the distance desired. Dose value = B.

3. The annual dose from the radiation leaving the side of a HI-STORM FW overpack, when it is behind another cask, was calculated at the distance desired. The casks have an assumed 15-foot pitch. Dose value = C.

The doses calculated in the steps above are listed in Table 5.4.7. Using these values, the annual dose (at the center of the long side) from an arbitrary 2 by Z array of HI-STORM FW overpacks can easily be calculated. The following formula describes the method.

Z = number of casks along long side

Dose = ZA + 2ZB + ZC

The results for various typical arrays of HI-STORM overpacks can be found in Section 5.1. While the off-site dose analyses were performed for typical arrays of casks containing design basis fuel, compliance with the requirements of 10CFR72.104(a) can only be demonstrated on a site-specific basis, as stated earlier. Therefore, a site-specific evaluation of dose at the controlled area boundary must be performed for each ISFSI in accordance with 10CFR72.212. The site-specific evaluation will consider the site-specific characteristics (such as exposure duration and the number of casks deployed), dose from other portions of the facility and the specifics of the fuel being stored (burnup and cooling time).

5.4.4 Non-Fuel Hardware

As discussed in Subsection 5.2.3, non-fuel hardware in the form of BPRAs, TPDs, CRAs, and APSRs are permitted for storage, integral with a PWR fuel assembly, in the HI-STORM FW system. Since each device occupies the same location within an assembly, only one device will be present in a given assembly. ITTRs, which are installed after core discharge and do not contain radioactive material, may also be stored in the assembly. BPRAs, TPDs and ITTRs are authorized for unrestricted storage in an MPC. The permissible locations of the CRAs and APSRs are shown in Figure 2.1.5.

Table 5.4.8 provides the dose rates at various locations on the surface and one meter from the HI-TRAC VW due to the BPRAs and TPDs for the MPC-37. The results in Table 5.4.8 show that the BPRAs essentially bound TPDs. All dose rates with NFH in this chapter therefore assume BPRA in every assembly. Note that, even for calculations without NFH, the dose from the active region conservatively contains the contribution of the BPRA. This mainly affects dose location 1 and 2, and results for these locations are therefore identical in most tables, and don't show the dose rate difference indicated in Table 5.4.8.

The analyses in this chapter that consider presence of BPRAs assume that a full-length rod with burnable poison is present in all principal locations. In reality, many BPRAs contain full-length poison rods in some locations, and thimble rodlets in others. The burnup and cooling time combinations listed in Table 2.1.25 of HI-STORM 100 FSAR [5.2.17] for BPRAs and TPDs were selected to ensure the Co-60 activity of those devices is below the value of 895 Ci (BPRA) and 50 Ci (TPD). These activities are used in the dose evaluations presented in this chapter. Apart from the total activity, the axial distribution of the material in those devices is important for the dose rates. This axial distribution is shown in Table 5.2.15 (masses) and 5.2.16 (activities). It can be observed from Table 5.2.16, while TPDs have a lower overall activity, their activity in the gas plenum region of the assembly is higher compared to that of the BPRAs. These activities were used to calculate the dose rates in Table 5.4.8. The results in this table show that the maximum dose effect for BPRAs is at the side of the cask, while the maximum dose effect of TPDs is near and on the top of the cask. Nevertheless, Table 5.4.8 demonstrates that even near and on the top of the cask, the TPD doses are bounded by the BPRA doses. It is to be noted that BPRAs with several thimble plugs may result higher dose rate near and on the top of the cask than that reported in Table 5.4.8. However, the potential local increase in dose near and on the top of the cask due the presence of several thimble plug rodlets instead of full length BPRA rods would be more than compensated by the reduction of the dose from the side of the cask at larger distances. Therefore, using BPRAs with all burnable poison rods in the analyses that demonstrate compliance with the site boundary dose limits would be bounding, and hence the burnup and cooling time combinations for BPRAs in Table 2.1.25 of the HI-STORM 100 FSAR [5.2.17] are conservative.

Two different configurations were analyzed for CRAs and three different configurations were analyzed for APSRs in the HI-STORM FSAR [5.2.17]. The dose rate due to CRAs and APSRs was explicitly calculated for dose locations around the HI-TRAC and results were provided for the different configurations of CRAs and APSRs, respectively, in the MPCs. These results indicate the dose rate on the radial surfaces of the overpack due to the storage of these devices is less than the dose rate from BPRAs (the increase in dose rate on the radial surface due to CRAs and APSRs are virtually negligible). For the surface dose rate at the bottom, the value for the CRA is comparable to or higher than the value from the BPRA. The increase in the bottom dose rates due to the presence of CRAs is on the order of 10-15% (based on bounding configuration 1 in [5.2.17]). The dose rate out the top of the overpack is essentially 0. The latter is due to the fact that CRAs and APSRs do not achieve significant activation in the upper portion of the devices due to the manner in which they are utilized during normal reactor operations. In contrast, the dose rate out the bottom of the overpack is substantial due to these devices. However, these dose rates occur in an area (below the pool lid and transfer doors) which is not normally occupied.

While the evaluations described above are based on conservative assumptions, the conclusions can vary slightly depending on the number of CRAs and their operating conditions.

5.4.5 Effect of Uncertainties

The design basis calculations presented in this chapter are based on a range of conservative assumptions, but do not explicitly account for uncertainties in the methodologies, codes and input parameters, that is, it is assumed that the effect of uncertainties is small compared to the numerous conservatisms in the analyses. To show that this assumption is valid, calculations have previously been performed as "best estimate" calculations and with estimated uncertainties added [5.4.9]. In all scenarios considered (e.g., evaluation of conservatisms in modeling assumptions, uncertainties associated with MCNP as well as the depletion analysis (including input parameters), etc.), the total dose rates long with uncertainties are comparable to, or lower than, the corresponding values from the design basis calculations. This provides further confirmation that the design basis calculations are reasonable and conservative.

Table 5.4.1

FLUX-TO-DOSE CONVERSION FACTORS (FROM [5.4.1])

Gamma Energy (MeV)	(rem/hr)/ (photon/cm^2-s)
0.01	3.96E-06
0.03	5.82E-07
0.05	2.90E-07
0.07	2.58E-07
0.1	2.83E-07
0.15	3.79E-07
0.2	5.01E-07
0.25	6.31E-07
0.3	7.59E-07
0.35	8.78E-07
0.4	9.85E-07
0.45	1.08E-06
0.5	1.17E-06
0.55	1.27E-06
0.6	1.36E-06
0.65	1.44E-06
0.7	1.52E-06
0.8	1.68E-06
1.0	1.98E-06
1.4	2.51E-06
1.8	2.99E-06
2.2	3.42E-06

Table 5.4.1 (continued) FLUX-TO-DOSE CONVERSION FACTORS (FROM [5.4.1])	
Gamma Energy (MeV)	**(rem/hr)/ (photon/cm^2-s)**
2.6	3.82E-06
2.8	4.01E-06
3.25	4.41E-06
3.75	4.83E-06
4.25	5.23E-06
4.75	5.60E-06
5.0	5.80E-06
5.25	6.01E-06
5.75	6.37E-06
6.25	6.74E-06
6.75	7.11E-06
7.5	7.66E-06
9.0	8.77E-06
11.0	1.03E-05
13.0	1.18E-05
15.0	1.33E-05

Table 5.4.1 (continued) FLUX-TO-DOSE CONVERSION FACTORS (FROM [5.4.1])		
Neutron Energy (MeV)	**Quality Factor**	**(rem/hr)[†]/(n/cm^2-s)**
2.5E-8	2.0	3.67E-6
1.0E-7	2.0	3.67E-6
1.0E-6	2.0	4.46E-6
1.0E-5	2.0	4.54E-6
1.0E-4	2.0	4.18E-6
1.0E-3	2.0	3.76E-6
1.0E-2	2.5	3.56E-6
0.1	7.5	2.17E-5
0.5	11.0	9.26E-5
1.0	11.0	1.32E-4
2.5	9.0	1.25E-4
5.0	8.0	1.56E-4
7.0	7.0	1.47E-4
10.0	6.5	1.47E-4
14.0	7.5	2.08E-4
20.0	8.0	2.27E-4

† Includes the Quality Factor.

Table 5.4.2

DOSE RATES FOR THE HI-TRAC VW FOR THE FULLY FLOODED MPC CONDITION WITH AN EMPTY NEUTRON SHIELD

MPC-37 DESIGN BASIS ZIRCALOY CLAD FUEL AT 45,000 MWD/MTU AND 4.5-YEAR COOLING

Dose Point Location	**Fuel Gammas (mrem/hr)**	**(n,γ) Gammas (mrem/hr)**	**^{60}Co Gammas (mrem/hr)**	**Neutrons (mrem/hr)**	**Totals (mrem/hr)**	**Totals with BPRAs (mrem/hr)**
ADJACENT TO THE HI-TRAC VW						
1	711	<1	489	67	1268	1268
2	2242	2	<1	319	2563	2563
3	7	<1	128	2	139	210
4	15	<1	227	<1	244	371
5 (bottom lid)	465	<1	1802	79	2346	2346
ONE METER FROM THE HI-TRAC VW						
1	541	<1	61	53	657	657
2	1141	<1	5	116	1263	1264
3	191	<1	75	20	286	326
4	8	<1	127	<1	137	208
5	259	<1	985	20	1266	1266

Notes:

- Refer to Figure 5.1.2 for dose point locations.
- Values are rounded to nearest integer.
- MPC internal water level is 5 inches below the MPC lid.
- The "Fuel Gammas" category includes gammas from the spent fuel, ^{60}Co from the spacer grids, and ^{60}Co from the BPRAs in the active fuel region.

Table 5.4.3

DOSE RATES FOR THE HI-TRAC VW FOR THE FULLY FLOODED MPC CONDITION WITH A FULL NEUTRON SHIELD

MPC-37 DESIGN BASIS ZIRCALOY CLAD FUEL AT 45,000 MWD/MTU AND 4.5-YEAR COOLING

Dose Point Location	Fuel Gammas (mrem/hr)	(n,γ) Gammas (mrem/hr)	^{60}Co Gammas (mrem/hr)	Neutrons (mrem/hr)	Totals (mrem/hr)	Totals with BPRAs (mrem/hr)
ADJACENT TO THE HI-TRAC VW						
1	433	1	301	5	740	740
2	1266	5	<1	27	1298	1298
3	2	<1	69	<1	73	111
4	15	<1	227	<1	244	371
5 (bottom lid)	465	<1	1802	80	2347	2347
ONE METER FROM THE HI-TRAC VW						
1	294	1	34	4	333	334
2	657	2	3	10	671	672
3	95	<1	41	1	138	160
4	8	<1	127	<1	137	208
5	259	<1	985	20	1265	1265

Notes:

- Refer to Figure 5.1.2 for dose point locations.
- Values are rounded to nearest integer.
- MPC internal water level is 5 inches below the MPC lid.
- The “Fuel Gammas” category includes gammas from the spent fuel, ^{60}Co from the spacer grids, and ^{60}Co from the BPRAs in the active fuel region.

Table 5.4.4

DOSE RATES FOR THE HI-TRAC VW FOR THE FULLY FLOODED MPC CONDITION WITH AN EMPTY NEUTRON SHIELD

MPC-89 DESIGN BASIS ZIRCALOY CLAD FUEL AT 45,000 MWD/MTU AND 5-YEAR COOLING

Dose Point Location	Fuel Gammas (mrem/hr)	(n,γ) Gammas (mrem/hr)	^{60}Co Gammas (mrem/hr)	Neutrons (mrem/hr)	Totals (mrem/hr)
ADJACENT TO THE HI-TRAC VW					
1	195	<1	1513	43	1752
2	2435	3	<1	579	3018
3	<1	<1	351	2	355
4	3	<1	217	<1	222
5 (bottom lid)	40	<1	1530	5	1576
ONE METER FROM THE HI-TRAC VW					
1	387	<1	197	69	654
2	1180	<1	12	168	1361
3	118	<1	154	26	299
4	<1	<1	132	<1	135
5	19	<1	864	2	886

Notes:
- Refer to Figure 5.1.2 for dose point locations.
- Values are rounded to nearest integer.
- MPC internal water level is 5 inches below the MPC lid.
- The “Fuel Gammas” category includes gammas from the spent fuel and ^{60}Co from the spacer grids.

Table 5.4.5

DOSE RATES FOR THE HI-TRAC VW FOR THE FULLY FLOODED MPC CONDITION WITH A FULL NEUTRON SHIELD
MPC-89 DESIGN BASIS ZIRCALOY CLAD FUEL AT 45,000 MWD/MTU AND 5-YEAR COOLING

Dose Point Location	**Fuel Gammas (mrem/hr)**	**(n,γ) Gammas (mrem/hr)**	**^{60}Co Gammas (mrem/hr)**	**Neutrons (mrem/hr)**	**Totals (mrem/hr)**
ADJACENT TO THE HI-TRAC VW					
1	102	<1	926	2	1031
2	1373	10	<1	47	1431
3	<1	<1	189	<1	192
4	3	<1	217	<1	222
5 (bottom lid)	40	<1	1530	5	1576
ONE METER FROM THE HI-TRAC VW					
1	211	1	115	5	332
2	627	3	7	16	653
3	72	<1	83	1	157
4	<1	<1	132	<1	135
5	19	<1	864	2	886

Notes:
- Refer to Figure 5.1.2 for dose point locations.
- Values are rounded to nearest integer.
- MPC internal water level is 5 inches below the MPC lid.
- The "Fuel Gammas" category includes gammas from the spent fuel and ^{60}Co from the spacer grids.

Table 5.4.6 ANNUAL DOSE AT 300 METERS FROM A SINGLE HI-STORM FW OVERPACK WITH AN MPC-37 WITH DESIGN BASIS ZIRCALOY CLAD FUEL	
Dose Component	**45,000 MWD/MTU 4.5-Year Cooling (mrem/yr)**
Fuel gammas	15.8
^{60}Co Gammas	2.2
Neutrons	0.2
Total	18.2

Notes:

- Gammas generated by neutron capture are included with fuel gammas.
- The Co-60 gammas include BPRAs.
- The "Fuel Gammas" category includes gammas from the spent fuel, ^{60}Co from the spacer grids, and ^{60}Co from the BPRAs in the active fuel region.

Table 5.4.7

DOSE VALUES USED IN CALCULATING ANNUAL DOSE FROM VARIOUS HI-STORM FW ISFSI CONFIGURATIONS
45,000 MWD/MTU AND 4.5-YEAR COOLING ZIRCALOY CLAD FUEL

Distance	**A Side of Overpack (mrem/yr)**	**B Top of Overpack (mrem/yr)**	**C Side of Shielded Overpack (mrem/yr)**
100 meters	396.0	44.0	79.2
200 meters	61.7	6.9	12.3
300 meters	16.4	1.8	3.3
400 meters	5.3	0.6	1.1
500 meters	2.0	0.2	0.4
600 meters	0.8	0.1	0.2

Notes:
- 8760 hour annual occupancy is assumed.
- Values are rounded to nearest integer where appropriate.

Table 5.4.8

DOSE RATES DUE TO BPRAs AND TPDs FROM THE HI-TRAC VW FOR NORMAL CONDITIONS

Dose Point Location	BPRAs (mrem/hr)	TPDs (mrem/hr)
ADJACENT TO THE HI-TRAC VW		
1	159.09	0.0
2	509.04	0.0
3	192.78	165.31
4	304.15	275.53
5	137.27	0.0
ONE METER FROM THE HI-TRAC VW		
1	122.06	0.40
2	240.70	3.10
3	128.50	86.95
4	174.25	153.49
5	63.13	0.0

Notes:

- Refer to Figure 5.1.2 for dose locations.
- Dose rates are based on no water within the MPC, an empty annulus, and a water jacket full of water. For the majority of the duration that the HI-TRAC bottom lid is installed, the MPC cavity will be flooded with water. The water within the MPC greatly reduces the dose rate
- Includes the BPRAs from both the active and non-active region.

5.5 REGULATORY COMPLIANCE

Chapters 1 and 2 and this chapter of this SAR describe in detail the shielding structures, systems, and components (SSCs) important to safety.

The shielding-significant SSCs important to safety have been valuated in this chapter and their impact on personnel and public health and safety resulting from operation of an independent spent fuel storage installation (ISFSI) utilizing the HI-STORM FW system has been evaluated.

It has been shown that the design of the shielding system of the HI-STORM FW system is in compliance with 10CFR72 and that the applicable design and acceptance criteria including 10CFR20 have been satisfied. Thus, this shielding evaluation provides reasonable assurance that the HI-STORM FW system will allow safe storage of spent fuel in full conformance with 10CFR72.

5.6 REFERENCES

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APPENDIX 5.A

SAMPLE INPUT FILES FOR SAS2H, ORIGEN-S, AND MCNP

Withheld in Accordance with 10 CFR 2.390