

CHAPTER 5[†]: SHIELDING EVALUATION

5.0 INTRODUCTION

The shielding analysis of the HI-STORM 100 System, including the HI-STORM 100 overpack, HI-STORM 100S overpack, and the 100-ton and 125-ton (including the 125D) HI-TRAC transfer casks, is presented in this chapter. The HI-STORM 100 System is designed to accommodate different MPCs within two HI-STORM overpacks (the HI-STORM 100S overpack is a shorter version of the HI-STORM 100 overpack). The MPCs are designated as MPC-24, MPC-24E and MPC-24EF (24 PWR fuel assemblies), MPC-32 (32 PWR fuel assemblies), and MPC-68, MPC-68F, and MPC-68FF (68 BWR fuel assemblies). The MPC-24E and MPC-24EF are essentially identical to the MPC-24 from a shielding perspective. Therefore only the MPC-24 is analyzed in this chapter. Likewise, the MPC-68, MPC-68F and MPC-68FF are identical from a shielding perspective and therefore only the MPC-68 is analyzed. Throughout this chapter, unless stated otherwise, MPC-24 refers to either the MPC-24, MPC-24E, or MPC-24EF and MPC-68 refers to the MPC-68, MPC-68F, and MPC-68FF.

In addition to storing intact PWR and BWR fuel assemblies, the HI-STORM 100 System is designed to store BWR and PWR damaged fuel assemblies and fuel debris. Damaged fuel assemblies and fuel debris are defined in Section 2.1.3 and the approved contents section of Appendix B to the CoC. Both damaged fuel assemblies and fuel debris are required to be loaded into Damaged Fuel Containers (DFCs) prior to being loaded into the MPC. DFCs containing BWR fuel debris must be stored in the MPC-68F or MPC-68FF. DFCs containing BWR damaged fuel assemblies may be stored in either the MPC-68, the MPC-68F, or the MPC-68FF. DFCs containing PWR fuel debris must be stored in the MPC-24EF while DFCs containing PWR damaged fuel assemblies may be stored in either the MPC-24E or MPC-24EF.

The MPC-68, MPC-68F, and MPC-68FF are also capable of storing Dresden Unit 1 antimony-beryllium neutron sources and the single Thoria rod canister which contains 18 thoria rods that were irradiated in two separate fuel assemblies.

PWR fuel assemblies may contain burnable poison rod assemblies (BPRAs), thimble plug devices (TPDs), control rod assemblies (CRAs) or axial power shaping rod assemblies (APSRs) or similarly named devices. These non-fuel hardware devices are an integral yet removable part of PWR fuel assemblies and therefore the HI-STORM 100 System has been designed to store

[†] 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 (Table 1.0.1) and component nomenclature of the Bill-of-Materials (Section 1.5).

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.

In order to offer the user more flexibility in fuel storage, the HI-STORM 100 System offers two different loading patterns in the MPC-24, MPC-24E, MPC-24EF, MPC-32, MPC-68, and the MPC-68FF. These patterns are uniform and regionalized loading as described in Section 2.0.1 and 2.1.6. Since the different loading patterns have different allowable burnup and cooling times combinations, both loading patterns are discussed in this chapter.

The sections that follow will demonstrate that the design of the HI-STORM 100 dry cask storage system fulfills the following acceptance criteria outlined in the Standard Review Plan, NUREG-1536 [5.2.1]:

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 cask vendor 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 10CFR 72.106.
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 10 CFR Part 20, Subparts C and D.

This chapter contains the following information which demonstrates full compliance with the Standard Review Plan, NUREG-1536:

- A description of the shielding features of the HI-STORM 100 System, including the HI-TRAC transfer cask.
- A description of the bounding source terms.
- A general description of the shielding analysis methodology.
- A description of the analysis assumptions and results for the HI-STORM 100 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.
- The HI-STORM 100 System has been analyzed to show that the 10CFR72.104 and 10CFR72.106 controlled area boundary radiation dose limits are met during normal, off-normal, and accident conditions of storage for non-effluent radiation from illustrative ISFSI configurations at a minimum distance of 100 meters.
- Analyses are also presented which demonstrate that the storage of damaged fuel and fuel debris in the HI-STORM 100 System is acceptable during normal, off-normal, and accident conditions.

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

Chapter 7 contains an analysis of the estimated dose at the controlled area boundary during normal, off-normal, and accident conditions from the release of radioactive materials. Therefore, this chapter only calculates the dose from direct neutron and gamma radiation emanating from the HI-STORM 100 System.

Chapter 10, Radiation Protection, contains the following information:

- A discussion of the estimated occupational exposures for the HI-STORM 100 System, including the HI-TRAC transfer cask.
- A summary of the estimated radiation exposure to the public.

5.1 DISCUSSION AND RESULTS

The principal sources of radiation in the HI-STORM 100 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)
 5. Dresden Unit 1 antimony-beryllium neutron sources

During loading, unloading, and transfer operations, shielding from gamma radiation is provided by the steel structure of the MPC and the steel, lead, and water of the HI-TRAC transfer cask. For storage, the gamma shielding is provided by the MPC, and the steel and concrete 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. Additionally, in the HI-TRAC 125 and 125D top lid and the transfer lid of the HI-TRAC 125, a solid neutron shielding material, Holtite-A is used to thermalize the neutrons. Boron carbide, dispersed in the solid neutron shield material utilizes the high neutron absorption cross section of ^{10}B to absorb the thermalized neutrons.

The shielding analyses were performed with MCNP-4A [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 4.3 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 B&W 15x15 and the GE 7x7, for PWR and BWR fuel types, respectively. The design basis intact 6x6 and mixed oxide (MOX) fuel assemblies are the GE 6x6. The GE 6x6 is also the design basis damaged fuel assembly for the Dresden Unit 1 and Humboldt Bay array classes. Table 2.1.6 specifies the acceptable intact zircaloy clad fuel characteristics for storage. Table 2.1.7 specifies the acceptable damaged fuel characteristics for storage.

The design basis stainless steel clad fuels are the WE 15x15 and the A/C 10x10, for PWR and BWR fuel types, respectively. Table 2.1.8 specifies the acceptable fuel characteristics of stainless steel clad fuel for storage.

The MPC-24, MPC-24E, MPC-24EF, MPC-32, MPC-68, and MPC-68FF are qualified for storage of SNF with different combinations of maximum burnup levels and minimum cooling times. The approved contents section of Appendix B to the CoC specifies the acceptable maximum burnup levels and minimum cooling times for storage of zircaloy clad fuel in these MPCs. Appendix B to the CoC also specifies the acceptable maximum burnup levels and minimum cooling times for storage of stainless steel clad fuel. The values in Appendix B to the CoC were chosen based on an analysis of the maximum decay heat load that could be accommodated within each MPC.

The dose rates surrounding the HI-STORM overpack are very low, and thus, the shielding analysis of the HI-STORM overpack conservatively considered the burnup and cooling time combinations listed below, which bound the acceptable burnup levels and cooling times from Appendix B to the CoC. This large conservatism is included in the analysis of the HI-STORM overpack to unequivocally demonstrate that the HI-STORM overpack meets the Part 72 dose requirements.

Zircaloy Clad Fuel		
MPC-24	MPC-32	MPC-68
52,500 MWD/MTU 5 year cooling	45,000 MWD/MTU 5 year cooling	47,500 MWD/MTU 5 year cooling
Stainless Steel Clad Fuel		
MPC-24	MPC-32	MPC-68
40,000 MWD/MTU 8 year cooling	40,000 MWD/MTU 9 year cooling	22,500 MWD/MTU 10 year cooling

The burnup and cooling time combinations analyzed for zircaloy clad fuel produce dose rates at the midplane of the HI-STORM overpack which bound all uniform and regionalized loading burnup and cooling time combinations listed in Appendix B to the CoC. Therefore, the HI-STORM shielding analysis presented in this chapter is conservatively bounding for the MPC-24, MPC-32, and MPC-68.

The dose rates surrounding the HI-TRAC transfer cask are significantly higher than the dose rates surrounding the HI-STORM overpack, and although no specific regulatory limits are defined, dose rates are based on the ALARA principle. Therefore, the cited dose rates were based on the actual burnups and cooling times requested in Appendix B to the CoC. Two different burnup and cooling times, listed below, were analyzed for the MPC-24, MPC-32, and

the MPC-68 in the 100-ton HI-TRAC. The burnups corresponding to 5-year cooling times produce dose rates at 1 meter from the surface of the overpack, for the locations reported in this chapter, which bound the dose rates from all other uniform loading burnup and cooling time combinations listed in Appendix B to the CoC. Since it is reasonable to assume that the majority of fuel which will be loaded in casks will be 10 years or older, the dose rates from conservative burnups for 10-year cooling are also presented in this chapter.

100-ton HI-TRAC		
MPC-24	MPC-32	MPC-68
42,500 MWD/MTU 5 year cooling	32,500 MWD/MTU 5 year cooling	40,000 MWD/MTU 5 year cooling
52,500 MWD/MTU 10 year cooling	45,000 MWD/MTU 10 year cooling	50,000 MWD/MTU 10 year cooling

The 100-ton HI-TRAC with the MPC-24 has higher dose rates at the mid-plane than the 100-ton HI-TRAC with the MPC-32 or the MPC-68. Therefore, the MPC-24 results for 5-year cooling are presented in this section and the MPC-24 was used for the dose exposure estimates in Chapter 10. The MPC-32 results, MPC-68 results, and additional MPC-24 results are provided in Section 5.4 for comparison.

The 100-ton HI-TRAC dose rates bound the HI-TRAC 125 and 125D dose rates for the same burnup and cooling time combinations. Therefore, for illustrative purposes, the MPC-24 was the only MPC analyzed in the HI-TRAC 125 and 125D. Since the HI-TRAC 125D has fewer radial ribs, the dose rate at the midplane of the HI-TRAC 125D is higher than the dose rate at the midplane of the HI-TRAC 125. Therefore, the results on the radial surface are only presented for the HI-TRAC 125D in this chapter. Dose rates are presented for two different burnup and cooling time combinations for the MPC-24 in the HI-TRAC 125D: 42,500 MWD/MTU with 5-year cooling and 57,500 MWD/MTU with 12-year cooling. The dose rates for the later combination are presented in this section because it produces the highest dose rate at the cask midplane. Dose rates for the other burnup and cooling time combination are presented in Section 5.4.

As a general statement, the dose rates for uniform loading presented in this chapter bound the dose rates for regionalized loading at 1 meter distance from the overpack. Therefore, dose rates for specific burnup and cooling time combinations in a regionalized loading pattern are not presented in this chapter. Section 5.4.9 provides an additional brief discussion on regionalized loading.

Unless otherwise stated all tables containing dose rates for design basis fuel refer to design basis intact zircaloy clad fuel.

5.1.1 Normal and Off-Normal Operations

Chapter 11 discusses the potential off-normal conditions and their effect on the HI-STORM 100 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 10 specifically addresses these regulations.

In accordance with ALARA practices, design objective dose rates are established for the HI-STORM 100 System in Section 2.3.5.2 as: 60 mrem/hour on the radial surface of the overpack, 60 mrem/hour at the openings of the air vents, and 60 mrem/hour on the top of the overpack.

The HI-STORM overpack dose rates presented in this section are conservatively evaluated for the MPC-32, the MPC-68, and the MPC-24. All burnup and cooling time combinations analyzed bound the allowable burnup and cooling times specified in Appendix B to the CoC.

Figure 5.1.1 and 5.1.12 identify the locations of the dose points referenced in the dose rate summary tables for the HI-STORM 100 and HI-STORM 100S overpacks, respectively. 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 peak dose location above the overpack shield block. For the adjacent top dose, this dose point is located over the air annulus between the MPC and the overpack. Dose Point #4a in Figure 5.1.12 is located directly above the exit duct and next to the concrete shield block. The dose values reported at the locations shown on Figure 5.1.1 and 5.1.12 are averaged over a region that is approximately 1 foot in width.

The total dose rates presented in this chapter for the MPC-24 and MPC-32 are presented for two cases: with and without BPRAs. The dose from the BPRAs was conservatively assumed to be the maximum calculated in Section 5.2.4.1. This is conservative because it is not expected that the cooling times for both the BPRAs and fuel assemblies would be such that they are both at the maximum design basis values.

Tables 5.1.1 and 5.1.3 provide the maximum dose rates adjacent to the HI-STORM 100S overpack during normal conditions for the MPC-32 and MPC-68. Tables 5.1.4 and 5.1.6 provide the maximum dose rates at one meter from the HI-STORM 100S overpack. Tables 5.1.2 and 5.1.5 provide the maximum dose rates adjacent to and one meter from the HI-STORM 100 overpack for the MPC-24.

Although the dose rates for the MPC-32 in HI-STORM 100s are equivalent to or greater than those for the MPC-24 in HI-STORM 100, as shown in Tables 5.1.1, 5.1.2, 5.1.4, and 5.1.5, the MPC-24 was used in the calculations for the dose rates at the controlled area boundary. The MPC-24 was chosen because, for a given cooling time, the MPC-24 has a higher allowable burnup than the MPC-32 or the MPC-68 (see Appendix B to the CoC). Consequently, for the allowable burnup and cooling times, the MPC-24 will have dose rates that are greater than or equivalent to those from the MPC-68 and MPC-32. The dose rates at the controlled area boundary were calculated for the HI-STORM 100 overpack rather than the HI-STORM 100S overpack. The difference in height will have little impact on the dose rates at the controlled area boundary since the surface dose rates are very similar. The controlled area boundary dose rates were also calculated without including non-fuel hardware. This is acceptable because the dose rates for the HI-STORM 100 overpack calculated in Table 5.1.2 without BPRAs are conservative enough to bound the dose rates for actual burnup and cooling times from Appendix B to the CoC including BPRAs.

Table 5.1.7 provides dose rates adjacent to and one meter from the 100-ton HI-TRAC. Table 5.1.8 provides dose rates adjacent to and one meter from the 125-ton HI-TRACs. Figures 5.1.2 and 5.1.4 identify the locations of the dose points referenced in Tables 5.1.7 and 5.1.8 for the HI-TRAC 125 and 100 transfer casks, respectively. The dose rates listed in Tables 5.1.7 and 5.1.8 correspond to the normal condition in which the MPC is dry and the HI-TRAC water jacket is filled with water. The dose rates below the HI-TRAC (Dose Point #5) are provided for two conditions. The first condition is when the pool lid is in use and the second condition is when the transfer lid is in use. The HI-TRAC 125D does not utilize the transfer lid, rather it utilizes the pool lid in conjunction with the mating device. Therefore the dose rates reported for the pool lid are applicable to both the HI-TRAC 125 and 125D while the dose rates reported for the transfer lid are applicable only to the HI-TRAC 125. The calculational model of the 100-ton HI-TRAC included a concrete floor positioned 6 inches (the typical carry height) below the pool lid to account for ground scatter. As a result of the modeling, the dose rate at 1 meter from the pool lid for the 100-ton HI-TRAC was not calculated. The dose rates provided in Tables 5.1.7 and 5.1.8 are for the MPC-24 with design basis fuel at burnups and cooling times, based on the allowed burnup and cooling times specified in Appendix B to the CoC, that result in dose rates that are generally higher in each of the two HI-TRAC designs. The burnup and cooling time combination used for both the 100-ton and 125-ton HI-TRAC was chosen based on the allowable burnup and cooling times in Appendix B to the CoC. Results for other burnup and cooling times and for the MPC-68 and MPC-32 are provided in Section 5.4.

Because the dose rates for the 100-ton HI-TRAC transfer cask are significantly higher than the dose rates for the 125-ton HI-TRACs or the HI-STORM overpack, it is important to understand

the behavior of the dose rates surrounding the external surface. To assist in this understanding, several figures, showing the dose rate profiles on the top, bottom and sides of the 100-ton HI-TRAC transfer cask, are presented below. The figures discussed below were all calculated without the gamma source from BPRAs and were calculated for an earlier design of the HI-TRAC which utilized 30 steel fins 0.375 inches thick compared to 10 steel fins 1.25 inches thick. The change in rib design only affects the magnitude of the dose rates presented for the radial surface but does not affect the conclusions discussed below.

Figure 5.1.5 shows the dose rate profile at 1 foot from the side of the 100-ton HI-TRAC transfer cask with the MPC-24 for 35,000 MWD/MTU and 5 year cooling. This figure clearly shows the behavior of the total dose rate and each of the dose components as a function of the cask height. To capture the effect of scattering off the concrete floor, the calculational model simulates the 100-ton HI-TRAC at a height of 6 inches (the typical cask carry height) above the concrete floor. As expected, the total dose rate on the side near the top and bottom is dominated by the Co-60 gamma dose component, while the center dose rate is dominated by the fuel gamma dose component.

The total dose rate and individual dose rate components on the surface of the pool lid on the 100-ton HI-TRAC are provided in Figure 5.1.6, illustrating the significant reduction in dose rate with increasing distance from the center of the pool lid. Specifically, the total dose rate is shown to drop by a factor of more than 20 from the center of the pool lid to the outer edge of the HI-TRAC. Therefore, even though the dose rate in Table 5.1.7 at the center of the pool lid is substantial, the dose rate contribution, from the pool lid, to the personnel exposure is minimal.

The behavior of the dose rate 1-foot from the transfer lid is shown in Figure 5.1.7. Similarly, the total dose rate and the individual dose rate components 1-foot from the top lid, as a function of distance from the axis of the 100-ton HI-TRAC, are shown in Figure 5.1.8. For both lids (transfer and top), the reduction in dose rate with increased distance from the cask axial centerline is substantial.

To reduce the dose rate above the water jacket, a localized temporary shield ring, described in Chapter 8, may be employed on the 125-ton HI-TRACs and on the 100-ton HI-TRAC. This temporary shielding, which is water, essentially extends the water jacket to the top of the HI-TRAC. The effect of the temporary shielding on the side dose rate above the water jacket (in the area around the lifting trunnions and the upper flange) is shown on Figure 5.1.9, which shows the dose profile on the side of the 100-ton HI-TRAC with the temporary shielding installed. For comparison, the total dose rate without temporary shielding installed is also shown on Figure 5.1.9. The results indicate that the temporary shielding reduces the dose rate by approximately a factor of 2 in the area above the water jacket.

To illustrate the reduction in dose rate with distance from the side of the 100-ton HI-TRAC, Figure 5.1.10 shows the total dose rate on the surface and at distances of 1-foot and 1-meter.

Figure 5.1.11 plots the total dose rate at various distances from the bottom of the transfer lid, including distances of 1, 5, 10, and 15 feet. Near the transfer lid, the total dose rate is shown to decrease significantly as a function of distance from the 100-ton HI-TRAC axial centerline. Near the axis of the HI-TRAC, the reduction in dose rate from the 1-foot distance to the 15-foot distance is approximately a factor of 15. The dose rate beyond the radial edge of the HI-TRAC is also shown to be relatively low at all distances from the HI-TRAC transfer lid. Thus, prudent transfer operating procedures will employ the use of distance to reduce personnel exposure. In addition, when the HI-TRAC is in the horizontal position and is being transported on site, a missile shield may be positioned in front of the HI-TRAC transfer lid or pool lid. If present, this shield would also serve as temporary gamma shielding which would greatly reduce the dose rate in the vicinity of the transfer lid or pool lid. For example, if the missile shield was a 2 inch thick steel plate, the gamma dose rate would be reduced by approximately 90%.

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. As mentioned, only the MPC-24 was used in the calculation of the dose rates at the controlled area boundary. Table 5.1.9 presents the annual dose to an individual from a single HI-STORM 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. These values were conservatively calculated for a burnup of 52,500 MWD/MTU and a 5-year cooling time. In addition, the annual dose was calculated for a burnup of 45,000 MWD/MTU and a 9-year cooling time. BPRAs were not included in these dose estimates. 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, as stated in Chapter 12, "Operating Controls and Limits". 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 cask array configurations provided in Table 5.1.9. 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.

Section 5.2 lists the gamma and neutron sources for the design basis fuels. Since the source strengths of the GE 6x6 intact and damaged fuel and the GE 6x6 MOX fuel are significantly smaller in all energy groups than the intact design basis fuel source strengths, the dose rates from the GE 6x6 fuels for normal conditions are bounded by the MPC-68 analysis with the design basis intact fuel. Therefore, no explicit analysis of the MPC-68 with either GE 6x6 intact or damaged or GE 6x6 MOX fuel for normal conditions is required to demonstrate that the MPC-68 with GE 6x6 fuels will meet the normal condition regulatory requirements. Section 5.4.2 evaluates the effect of generic damaged fuel in the MPC-24E and the MPC-68.

Section 5.2.6 lists the gamma and neutron sources from the Dresden Unit 1 Thoria rod canister and demonstrates that the Thoria rod canister is bounded by the design basis Dresden Unit 1 6x6 intact fuel.

Section 5.2.4 presents the Co-60 sources from the BPRAs, TPDs, CRAs and APSRs that are permitted for storage in the HI-STORM 100 System. Section 5.4.6 discusses the increase in dose rate as a result of adding non-fuel hardware in the MPCs.

Section 5.4.7 demonstrates that the Dresden Unit 1 fuel assemblies containing antimony-beryllium neutron sources are bounded by the shielding analysis presented in this section.

Section 5.2.3 lists the gamma and neutron sources for the design basis stainless steel clad fuel. The dose rates from this fuel are provided in Section 5.4.4.

The analyses summarized in this section demonstrate that the HI-STORM 100 System, including the HI-TRAC transfer cask, are in compliance with the 10CFR72.104 limits and ALARA practices.

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.

Design basis accidents which may affect the HI-STORM 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, adjacent, and one meter doses for the loaded HI-STORM overpack for accident conditions are equivalent to the normal condition doses, which meet the 10CFR72.106 radiation dose limits.

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 fuel spacers. The HI-STAR 100 System (Docket Number 72-1008) documentation provides analysis to demonstrate that the fuel spacers will not fail under

any normal, off-normal, or accident condition of storage. Chapter 3 also shows that the HI-TRAC inner shell, lead, and outer shell remain intact throughout all design basis accident conditions. Localized damage of the HI-TRAC outer shell could be experienced. However, the 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 Figures 5.1.2 and 5.1.4) are provided in Table 5.1.10. The normal condition dose rates are provided for reference. Table 5.1.10 provides a comparison of the normal and accident condition dose rates at one meter from the HI-TRAC. The burnup and cooling time combinations used in Table 5.1.10 were the combinations that resulted in the highest post-accident condition dose rates. These burnup and cooling time combinations do not necessarily correspond to the burnup and cooling time combinations that result in the highest dose rate during normal conditions. Scaling this accident dose rate by the dose rate reduction seen in HI-STORM yields a dose rate at the 100 meter controlled area boundary that would be approximately 1.48[†] mrem/hr for the HI-TRAC accident condition. At this dose rate, it would take 3378 hours (~141 days) for the dose at the controlled area boundary to reach 5 Rem. 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.

The consequences of the design basis accident conditions for the MPC-68 and MPC-24E storing damaged fuel and the MPC-68F, MPC-68FF, or MPC-24EF storing damaged fuel and/or fuel debris differ slightly from those with intact fuel. It is conservatively assumed that during a drop accident (vertical, horizontal, or tip-over) the damaged fuel collapses and the pellets rest in the bottom of the damaged fuel container. Analyses in Section 5.4.2 demonstrates that the damaged fuel in the post-accident condition does not significantly affect the dose rates around the cask. Therefore, the damaged fuel post-accident dose rates are bounded by the intact fuel post-accident dose rates.

Analyses summarized in this section demonstrate that the HI-STORM 100 System, including the HI-TRAC transfer cask, are in compliance with the 10CFR72.106 limits.

[†] 2098.54 mrem/hr (Table 5.1.10) x [129 mrem/yr (Table 5.4.7) / 8760 hrs / 20.9 mrem/hr (Table 5.1.5)]

Table 5.1.1

DOSE RATES ADJACENT TO HI-STORM 100S OVERPACK
 FOR NORMAL CONDITIONS
 MPC-32 DESIGN BASIS ZIRCALOY CLAD FUEL AT BOUNDING
 BURNUP AND COOLING TIME
 45,000 MWD/MTU AND 5-YEAR COOLING

Dose Point [†] Location	Fuel Gammas ^{††} (mrem/hr)	⁶⁰ Co Gammas (mrem/hr)	Neutrons (mrem/hr)	Totals (mrem/hr)	Totals with BPRAs (mrem/hr)
1	10.45	16.45	7.17	34.07	34.94
2	37.19 ^{†††}	0.05	2.13	39.37	45.15
3	11.74	17.18	5.63	34.55	42.17
4	2.41	1.07	1.98	5.47	6.16
4a	3.86	9.48	27.59	40.93	45.45

[†] Refer to Figure 5.1.12.

^{††} Gammas generated by neutron capture are included with fuel gammas.

^{†††} The cobalt activation of incore grid spacers accounts for 8.5 % of this dose rate.

Table 5.1.2

DOSE RATES ADJACENT TO HI-STORM 100 OVERPACK
 FOR NORMAL CONDITIONS
 MPC-24 DESIGN BASIS ZIRCALOY CLAD FUEL AT BOUNDING
 BURNUP AND COOLING TIME
 52,500 MWD/MTU AND 5-YEAR COOLING

Dose Point [†] Location	Fuel Gammas ^{††} (mrem/hr)	⁶⁰ Co Gammas (mrem/hr)	Neutrons (mrem/hr)	Totals (mrem/hr)	Totals with BPRAs (mrem/hr)
1	7.20	5.34	4.46	17.00	17.35
2	37.65 ^{†††}	0.03	3.04	40.72	45.77
3	4.87	3.52	2.23	10.61	12.16
4	1.28	0.39	5.82	7.49	7.70

[†] Refer to Figure 5.1.1.

^{††} Gammas generated by neutron capture are included with fuel gammas.

^{†††} The cobalt activation of incore grid spacers accounts for 8.0 % of this dose rate.

Table 5.1.3

DOSE RATES ADJACENT TO HI-STORM 100S OVERPACK FOR NORMAL
 CONDITIONS
 MPC-68 DESIGN BASIS ZIRCALOY CLAD FUEL AT BOUNDING
 BURNUP AND COOLING TIME
 47,500 MWD/MTU AND 5-YEAR COOLING

Dose Point [†] Location	Fuel Gammas ^{††} (mrem/hr)	⁶⁰ Co Gammas (mrem/hr)	Neutrons (mrem/hr)	Totals (mrem/hr)
1	10.45	12.45	9.57	32.47
2	33.88	0.01	2.91	36.80
3	4.51	16.08	4.27	24.86
4	1.42	1.22	1.55	4.19
4a	1.17	9.88	20.74	31.79

[†] Refer to Figure 5.1.12.

^{††} Gammas generated by neutron capture are included with fuel gammas.

Table 5.1.4

DOSE RATES AT ONE METER FROM HI-STORM 100S OVERPACK
 FOR NORMAL CONDITIONS
 MPC-32 DESIGN BASIS ZIRCALOY CLAD FUEL AT BOUNDING
 BURNUP AND COOLING TIME
 45,000 MWD/MTU AND 5-YEAR COOLING

Dose Point [†] Location	Fuel Gammas ^{††} (mrem/hr)	⁶⁰ Co Gammas (mrem/hr)	Neutrons (mrem/hr)	Totals (mrem/hr)	Totals with BPRAs (mrem/hr)
1	5.66	5.51	1.03	12.20	12.96
2	18.83 ^{†††}	0.66	0.91	20.40	23.44
3	4.84	4.84	0.91	10.59	13.09
4	0.67	0.30	1.03	2.00	2.16

[†] Refer to Figure 5.1.12.

^{††} Gammas generated by neutron capture are included with fuel gammas.

^{†††} The cobalt activation of incore grid spacers accounts for 8.6 % of this dose rate.

Table 5.1.5

DOSE RATES AT ONE METER FROM HI-STORM 100 OVERPACK
 FOR NORMAL CONDITIONS
 MPC-24 DESIGN BASIS ZIRCALOY CLAD FUEL AT BOUNDING
 BURNUP AND COOLING TIME
 52,500 MWD/MTU AND 5-YEAR COOLING

Dose Point [†] Location	Fuel Gammas ^{††} (mrem/hr)	⁶⁰ Co Gammas (mrem/hr)	Neutrons (mrem/hr)	Totals (mrem/hr)	Totals with BPRAs (mrem/hr)
1	5.73	3.18	0.87	9.79	10.29
2	19.38 ^{†††}	0.27	1.26	20.90	23.48
3	3.28	2.29	0.34	5.91	7.05
4	0.58	0.18	1.77	2.53	2.63

[†] Refer to Figure 5.1.1.

^{††} Gammas generated by neutron capture are included with fuel gammas.

^{†††} The cobalt activation of incore grid spacers accounts for 8.0 % of this dose rate.

Table 5.1.6

DOSE RATES AT ONE METER FROM HI-STORM 100S OVERPACK
 FOR NORMAL CONDITIONS
 MPC-68 DESIGN BASIS ZIRCALOY CLAD FUEL AT BOUNDING
 BURNUP AND COOLING TIME
 47,500 MWD/MTU AND 5-YEAR COOLING

Dose Point [†] Location	Fuel Gammas ^{††} (mrem/hr)	⁶⁰ Co Gammas (mrem/hr)	Neutrons (mrem/hr)	Totals (mrem/hr)
1	5.69	3.92	1.27	10.88
2	16.77	0.29	1.22	18.27
3	2.33	4.92	0.70	7.95
4	0.43	0.30	0.76	1.49

[†] Refer to Figure 5.1.12.

^{††} Gammas generated by neutron capture are included with fuel gammas.

Table 5.1.7

DOSE RATES FROM THE 100-TON HI-TRAC FOR NORMAL CONDITIONS
MPC-24 DESIGN BASIS ZIRCALOY CLAD FUEL
42,500 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)	Totals with BPRAs (mrem/hr)
ADJACENT TO THE 100-TON HI-TRAC						
1	37.24	12.56	627.06	177.78	854.64	862.57
2	1007.72 [†]	51.10	0.63	94.31	1153.77	1383.33
3	10.41	2.46	345.74	148.75	507.37	645.28
3 (temp)	4.70	4.38	160.25	2.39	171.72	234.99
4	24.53	0.97	278.26	183.11	486.87	602.47
4 (outer)	7.01	0.62	69.28	123.61	200.52	229.61
5 (pool lid)	263.06	16.65	3174.00	1102.22	4555.94	4616.08
5 (transfer)	416.74	0.98	4707.91	683.81	5809.44	5899.35
5(t-outer)	102.21	0.34	455.69	270.15	828.38	848.32
ONE METER FROM THE 100-TON HI-TRAC						
1	132.96	6.75	93.21	28.90	261.82	291.98
2	441.79 [†]	15.62	7.37	35.36	500.15	601.80
3	55.58	3.76	77.43	13.88	150.65	196.55
3 (temp)	55.27	4.04	65.95	5.25	130.51	171.89
4	8.40	0.17	85.91	45.62	140.10	175.91
5 (transfer)	179.95	0.18	1923.20	190.81	2294.14	2330.47
5(t-outer)	24.17	0.58	173.35	54.78	252.89	256.51

Notes:

- Refer to Figures 5.1.2 and 5.1.4 for dose locations.
- Dose location 3(temp) represents dose location 3 with temporary shielding installed.
- Dose location 4(outer) is the radial segment at dose location 4 which is 18-30 inches from the center of the overpack.
- Dose location 5(t-outer) is the radial segment at dose location 5 (transfer lid) which is 30-42 and 54-66 inches from the center of the lid for the adjacent and one meter locations, respectively. The inner radius of the HI-TRAC is 34.375 in. and the outer radius of the water jacket is 44.375 in.
- Dose rate based on no water within the MPC. For the majority of the duration that the HI-TRAC pool lid is installed, the MPC cavity will be flooded with water. The water within the MPC greatly reduces the dose rate.

[†] The cobalt activation of incore grid spacers accounts for 12.3% of the surface and one-meter dose rates.

Table 5.1.8

DOSE RATES FROM THE 125-TON HI-TRACS FOR NORMAL CONDITIONS
MPC-24 DESIGN BASIS ZIRCALOY CLAD FUEL
57,500 MWD/MTU AND 12-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 125-TON HI-TRACs						
1	1.25	22.96	34.87	154.47	213.56	214.22
2	23.63 [†]	68.02	0.00	106.99	198.65	213.38
3	0.31	2.43	21.58	246.42	270.74	289.05
4	8.37	3.12	118.06	285.23	414.78	521.95
4 (outer)	0.94	2.23	14.66	5.99	23.82	36.99
5 (pool)	10.77	1.36	157.53	1070.88	1240.55	1247.70
5 (transfer)	11.50	1.78	208.42	163.54	385.23	390.32
ONE METER FROM THE 125-TON HI-TRACs						
1	3.06	9.16	4.47	25.41	42.11	44.00
2	10.42 [†]	22.05	0.18	36.49	69.14	75.70
3	1.15	5.18	4.36	22.68	33.37	37.93
4	2.35	0.75	28.42	29.37	60.90	86.57
5 (transfer)	5.24	0.34	100.76	28.33	134.68	137.39

Notes:

- Refer to Figures 5.1.2 and 5.1.4 for dose locations.
- Dose location 4(outer) is the radial segment at dose location 4 which is 18-24 inches from the center of the overpack.
- Dose rate based on no water within the MPC. For the majority of the duration that the HI-TRAC pool lid is installed, the MPC cavity will be flooded with water. The water within the MPC greatly reduces the dose rate.

[†] The cobalt activation of incore grid spacers accounts for 15.5% of the surface and one-meter dose rates.

Table 5.1.9

DOSE RATES FOR ARRAYS OF MPC-24
WITH DESIGN BASIS ZIRCALOY CLAD FUEL
AT VARYING BURNUP AND COOLING TIMES

Array Configuration	1 cask	2x2	2x3	2x4	2x5
52,500 MWD/MTU AND 5-YEAR COOLING					
Annual Dose (mrem/year) [†]	20.19	23.83	19.13	14.91	18.64
Distance to Controlled Area Boundary (meters) ^{††,†††}	200	250	300	350	350
45,000 MWD/MTU AND 9-YEAR COOLING					
Annual Dose (mrem/year) [†]	16.03	16.95	12.19	16.26	20.32
Distance to Controlled Area Boundary (meters) ^{††}	150	200	250	250	250

† 8760 hr. annual occupancy is assumed.

†† Dose location is at the center of the long side of the array.

††† Actual controlled area boundary dose rates will be lower because the maximum permissible burnup for 5-year cooling, as specified in the Appendix B to the CoC, is lower than the burnup used for this analysis.

Table 5.1.10

DOSE RATES AT ONE METER FROM HI-TRAC
 FOR ACCIDENT CONDITIONS
 MPC-24 DESIGN BASIS ZIRCALOY CLAD FUEL
 AT BOUNDING BURNUP AND COOLING TIMES

Dose Point [†] Location	Fuel Gammas ^{††} (mrem/hr)	⁶⁰ Co Gammas (mrem/hr)	Neutrons (mrem/hr)	Totals (mrem/hr)	Totals with BPRAs (mrem/hr)
125-TON HI-TRACs					
57,500 MWD/MTU AND 12-YEAR COOLING					
2 (Accident Condition)	19.73	0.35	1291.13	1311.22	1323.19
2 (Normal Condition)	32.47	0.18	36.49	69.14	75.70
100-TON HI-TRAC					
57,500 MWD/MTU AND 12-YEAR COOLING					
2 (Accident Condition)	291.47	6.20	1618.68	1916.34	2098.54
2 (Normal Condition)	189.12	3.43	62.72	255.28	356.93

† Refer to Figures 5.1.2 and 5.1.4.

†† Gammas generated by neutron capture are included with fuel gammas.

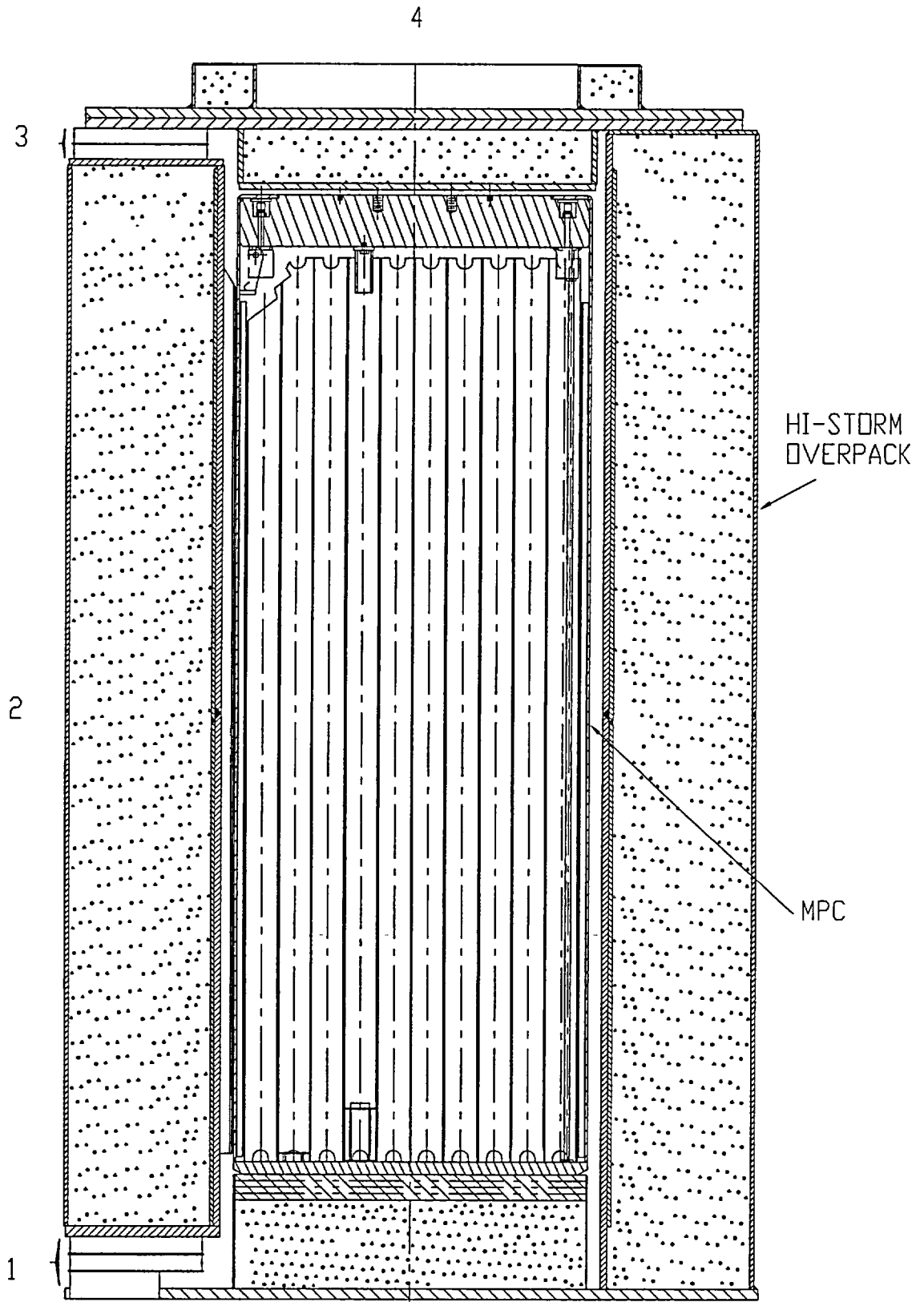


FIGURE 5.1.1; CROSS SECTION ELEVATION VIEW OF HI-STORM 100 OVERPACK WITH DOSE POINT LOCATION

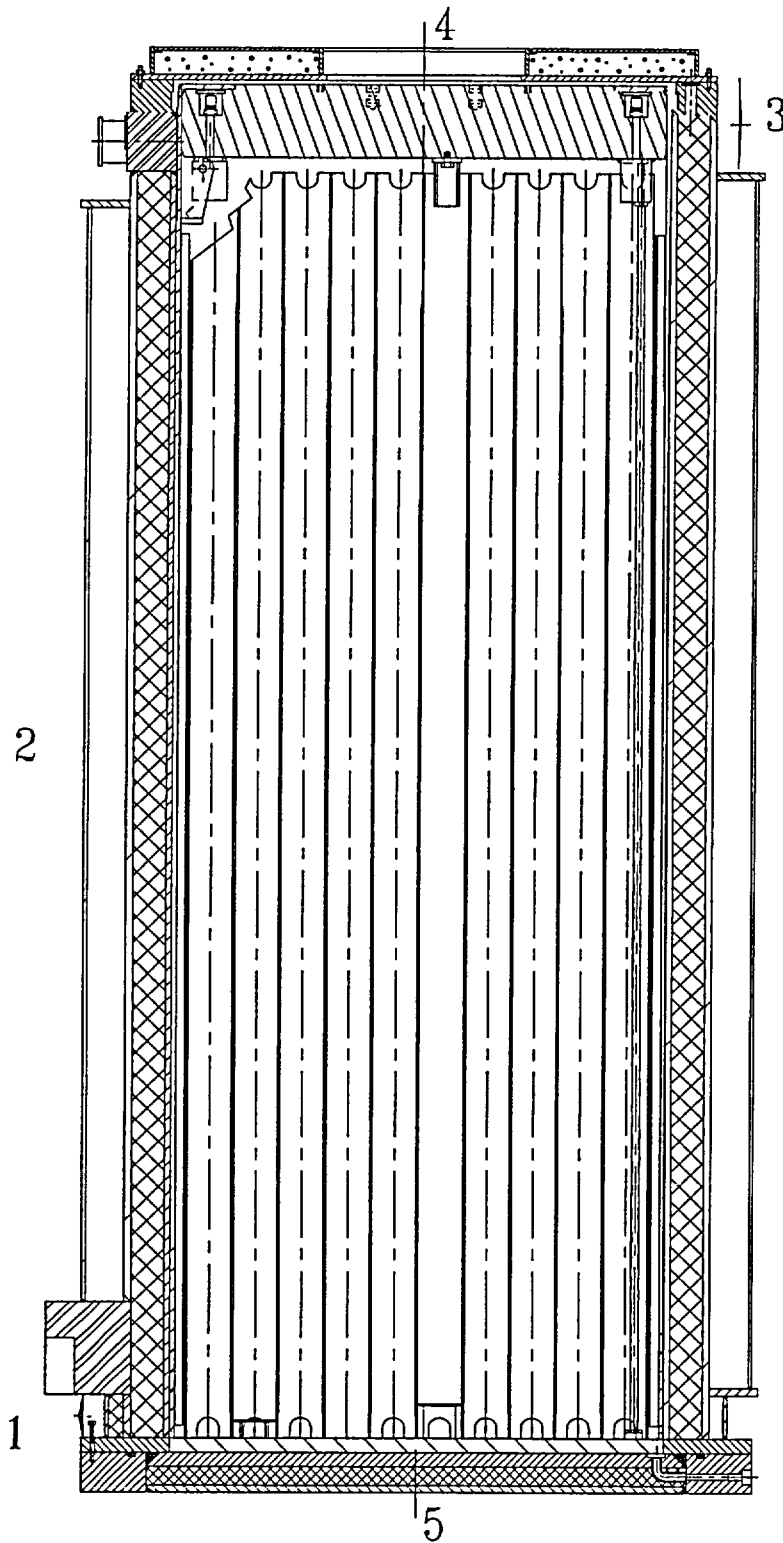


FIGURE 5.1.2; CROSS SECTION ELEVATION VIEW OF 125 TON HI-TRAC TRANSFER CASK WITH DOSE POINT LOCATIONS

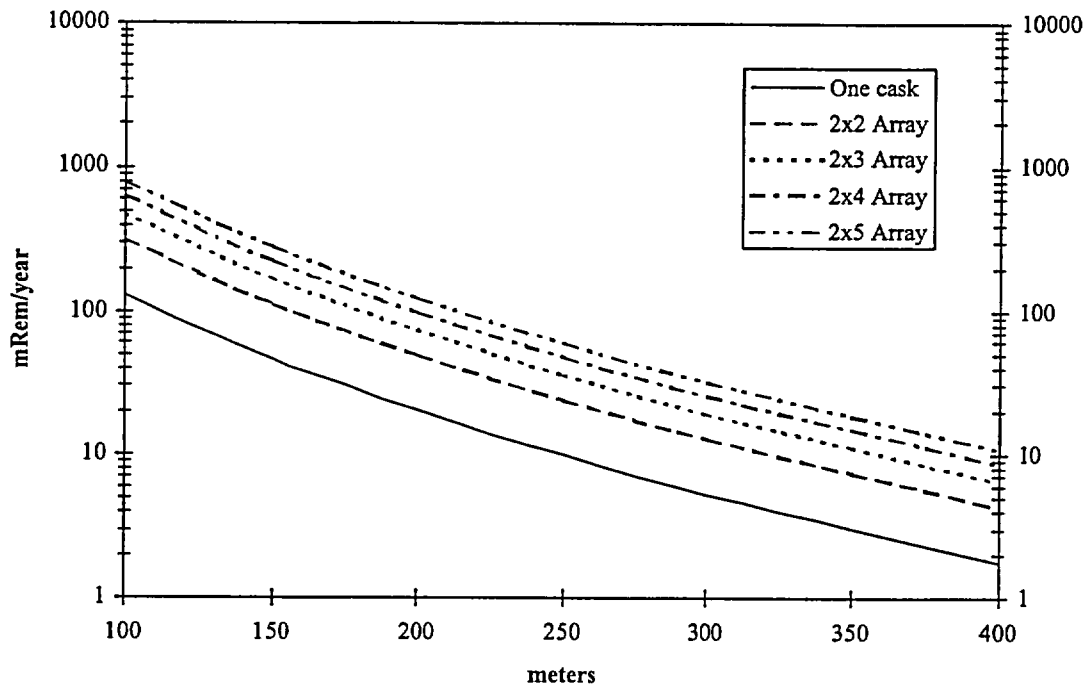


FIGURE 5.1.3; ANNUAL DOSE VERSUS DISTANCE FOR VARIOUS CONFIGURATIONS OF THE MPC-24 FOR 52,500 MWD/MTU AND 5-YEAR COOLING (8760 HOUR OCCUPANCY ASSUMED)

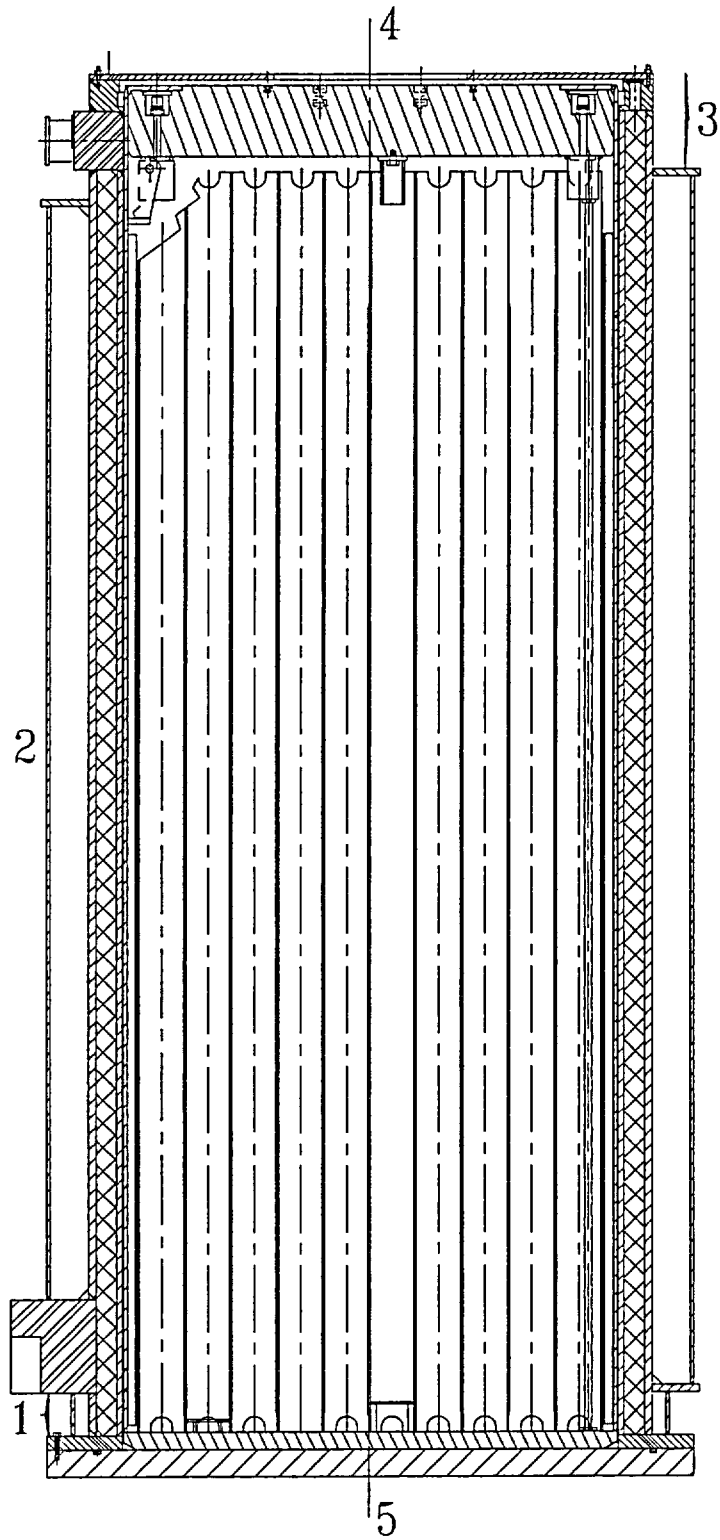


FIGURE 5.1.4; CROSS SECTION ELEVATION VIEW OF 100 TON HI-TRAC TRANSFER CASK (WITH POOL LID) WITH DOSE POINT LOCATIONS

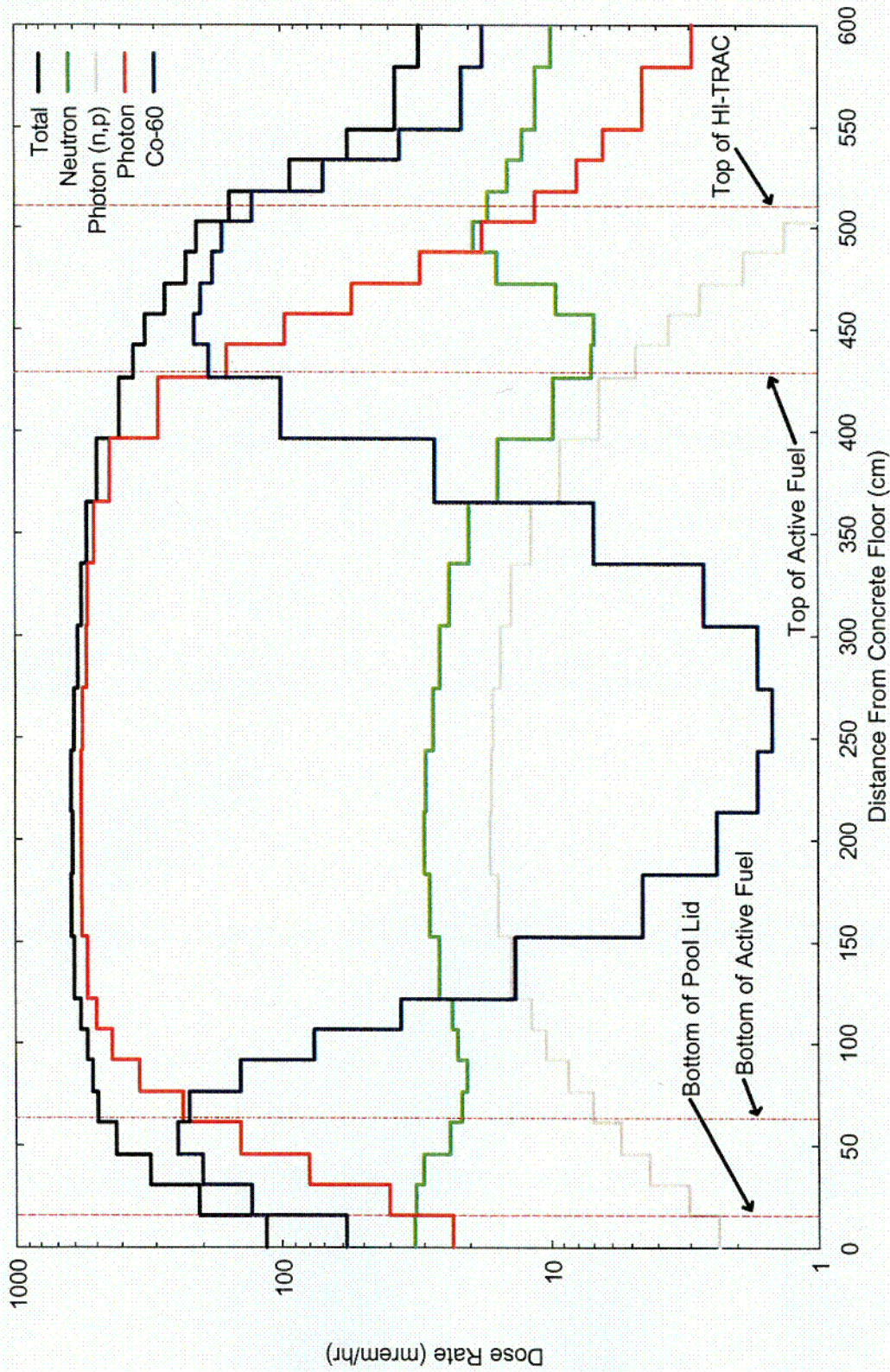


FIGURE 5.1.5; DOSE RATE 1-FOOT FROM THE SIDE OF THE 100-TON HI-TRAC TRANSFER CASK WITH THE MPC-24 FOR 35,000 MWD/MTU AND 5-YEAR COOLING

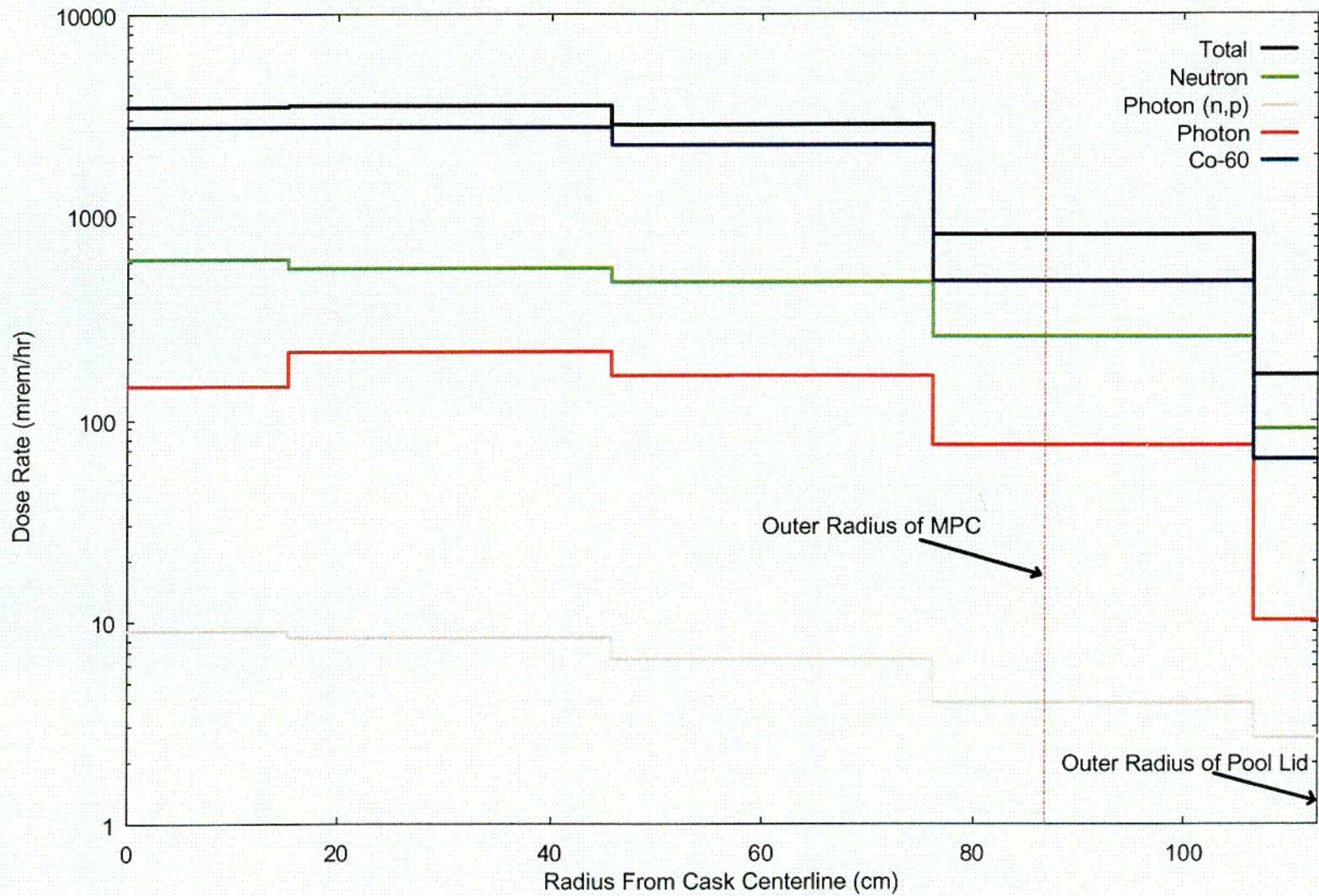


FIGURE 5.1.6; DOSE RATE ON THE SURFACE OF THE POOL LID ON THE 100-TON HI-TRAC TRANSFER CASK WITH THE MPC-24 FOR 35,000 MWD/MTU AND 5-YEAR COOLING

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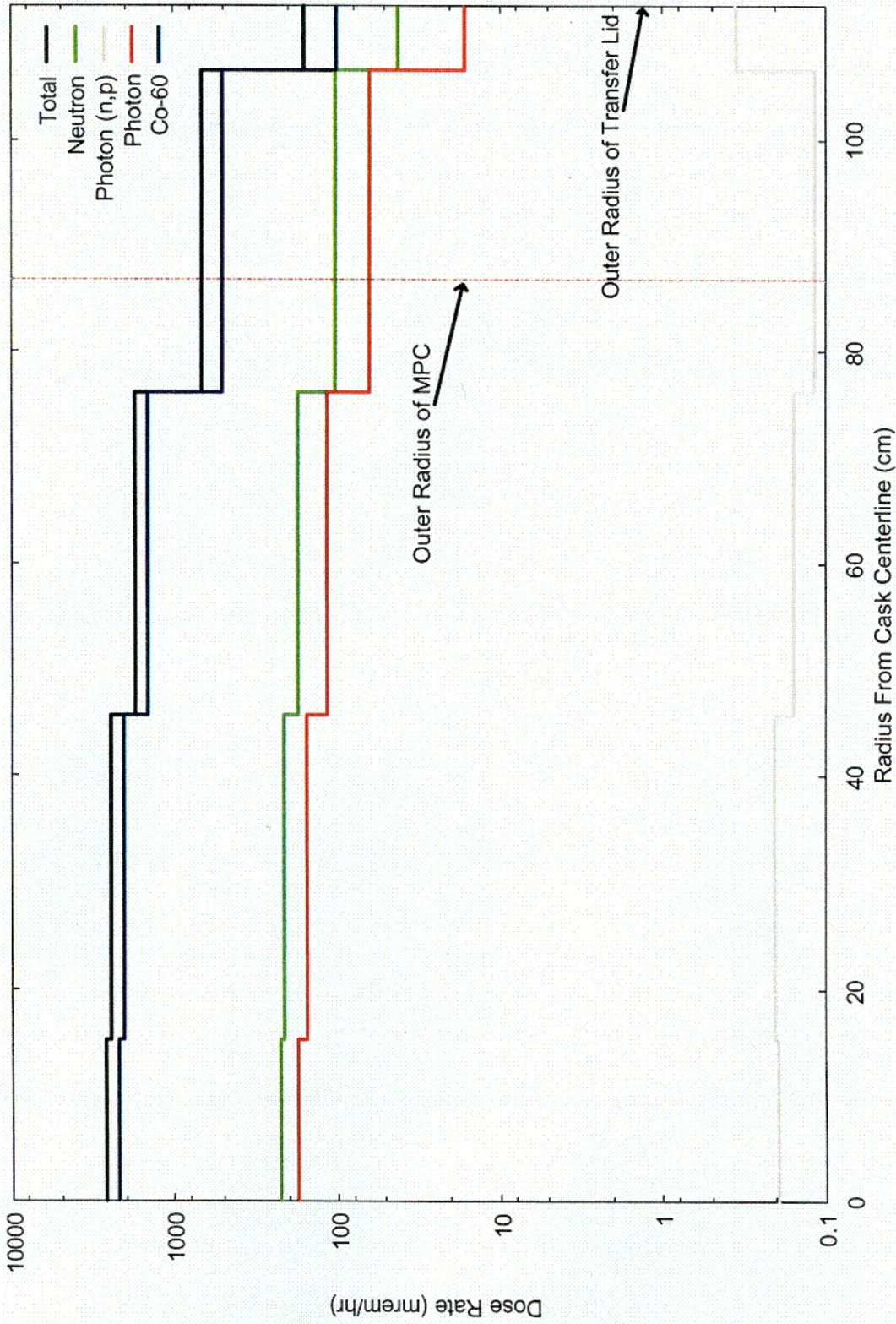


FIGURE 5.1.7; DOSE RATE 1-FOOT FROM THE BOTTOM OF TRANSFER LID ON THE 100-TON HI-TRAC TRANSFER CASK WITH THE MPC-24 FOR 35,000 MWD/MTU AND 5-YEAR COOLING

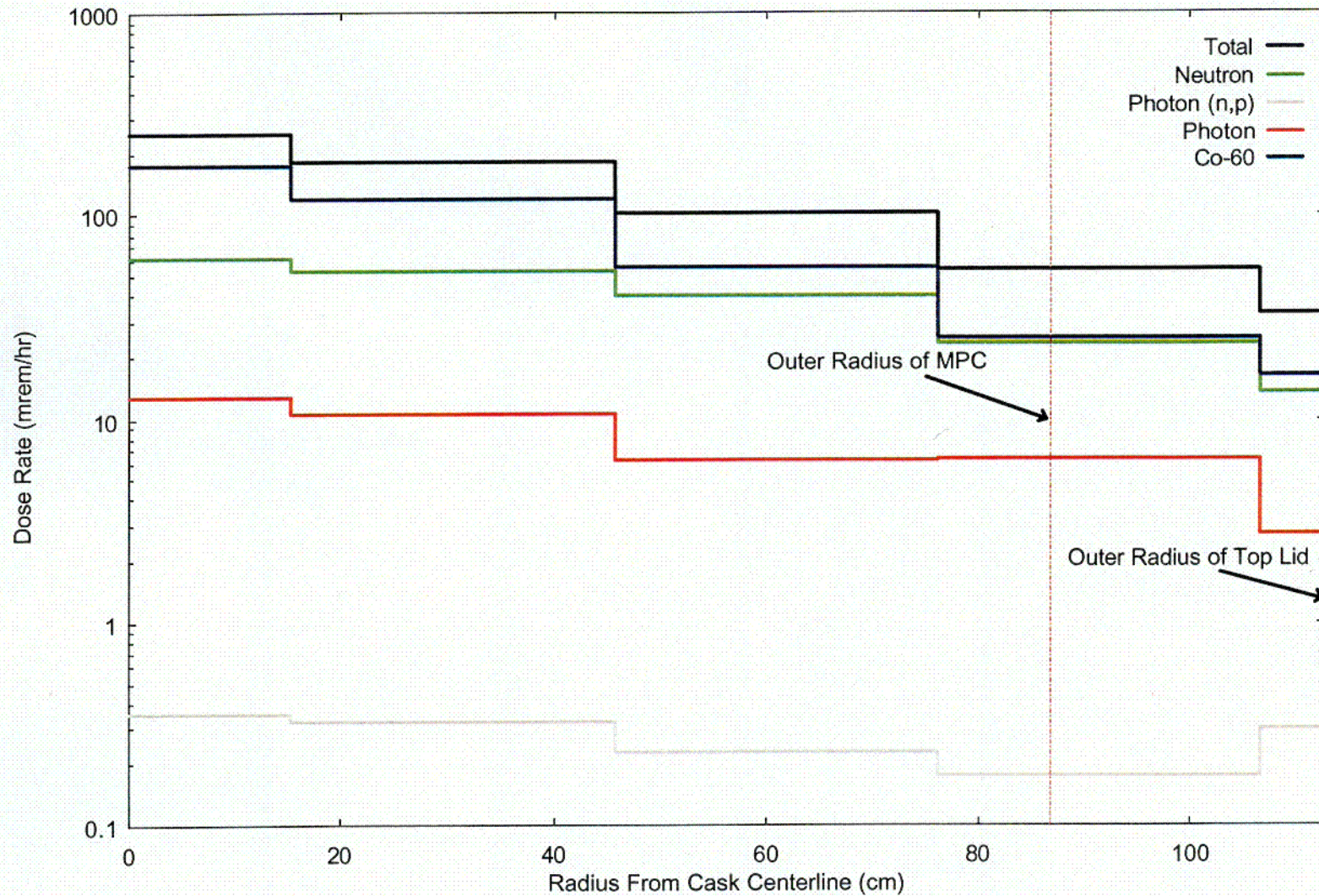


FIGURE 5.1.8; DOSE RATE 1-FOOT FROM THE TOP OF TOP LID ON THE 100-TON HI-TRAC TRANSFER CASK WITH THE MPC-24 FOR 35,000 MWD/MTU AND 5-YEAR COOLING

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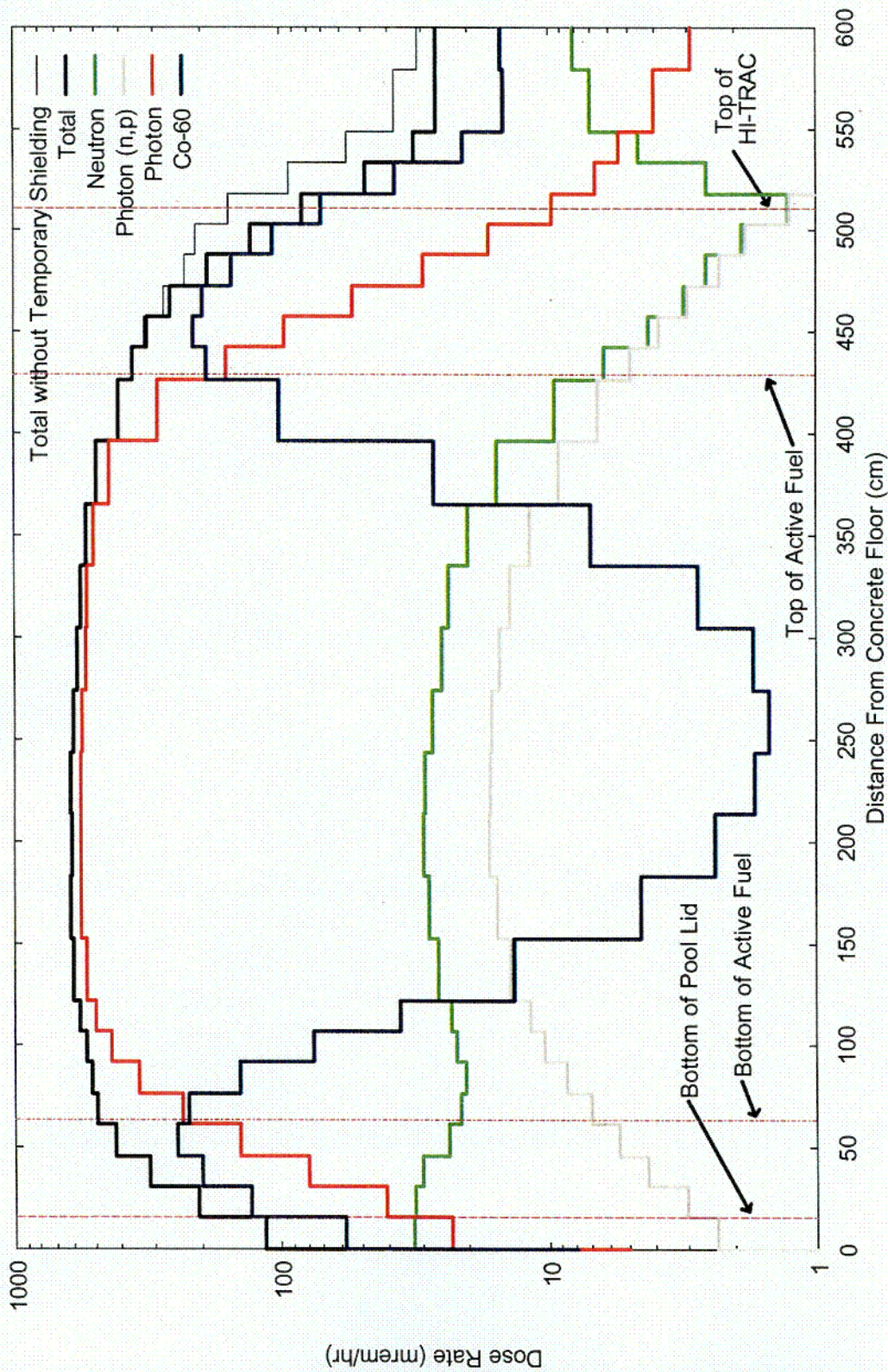


FIGURE 5.1.9; DOSE RATE 1-FOOT FROM THE SIDE OF THE 100-TON HI-TRAC TRANSFER CASK WITH TEMPORARY SHIELDING INSTALLED, WITH THE MPC-24 FOR 35,000 MWD/MTU AND 5-YEAR COOLING (TOTAL DOSE WITHOUT TEMPORARY SHIELDING SHOWN FOR COMPARISON)

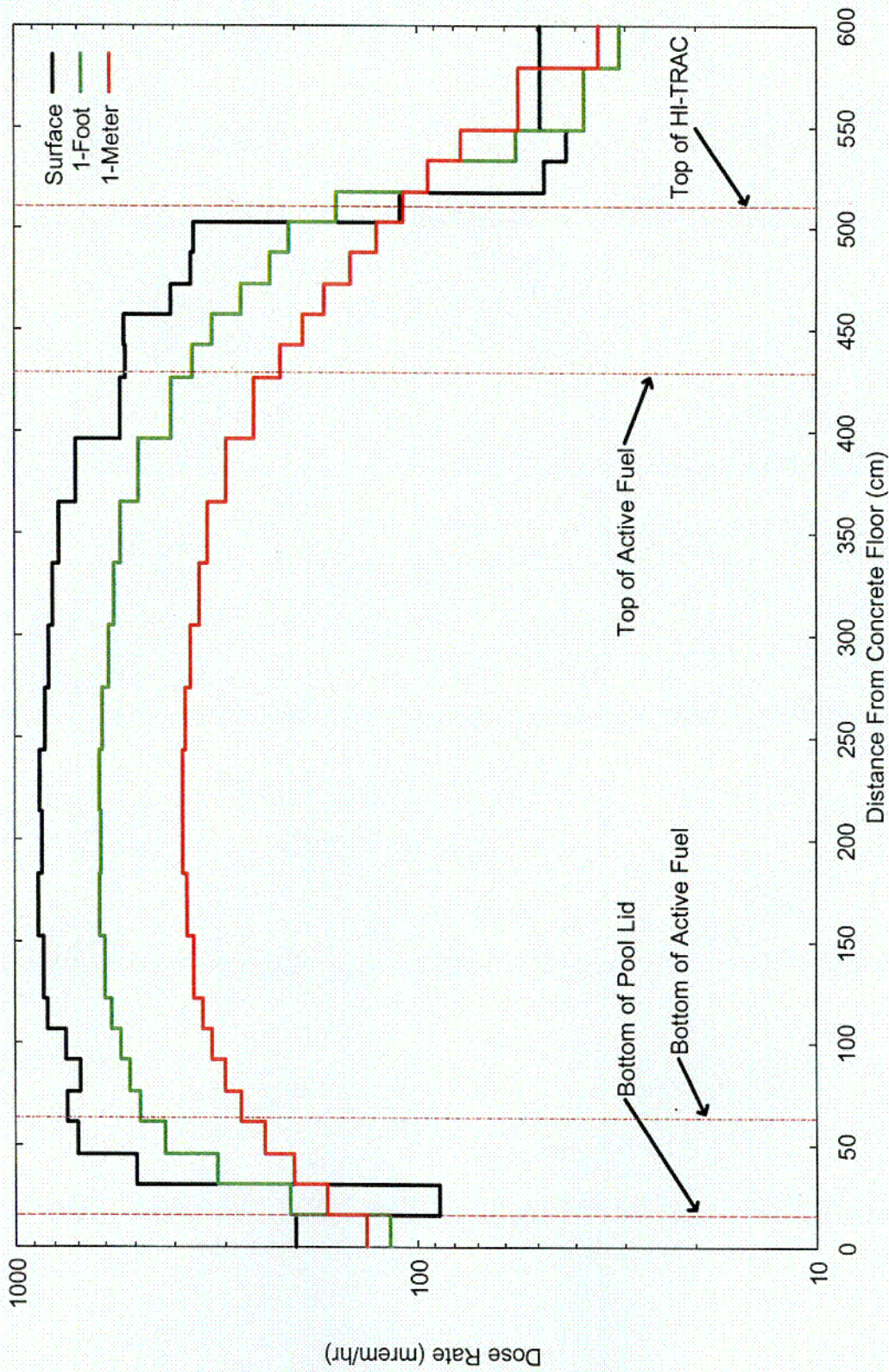


FIGURE 5.1.10; DOSE RATE AT VARIOUS DISTANCES FROM THE SIDE OF THE 100-TON HI-TRAC TRANSFER CASK WITH THE MPC-24 FOR 35,000 MWD/MTU AND 5-YEAR COOLING

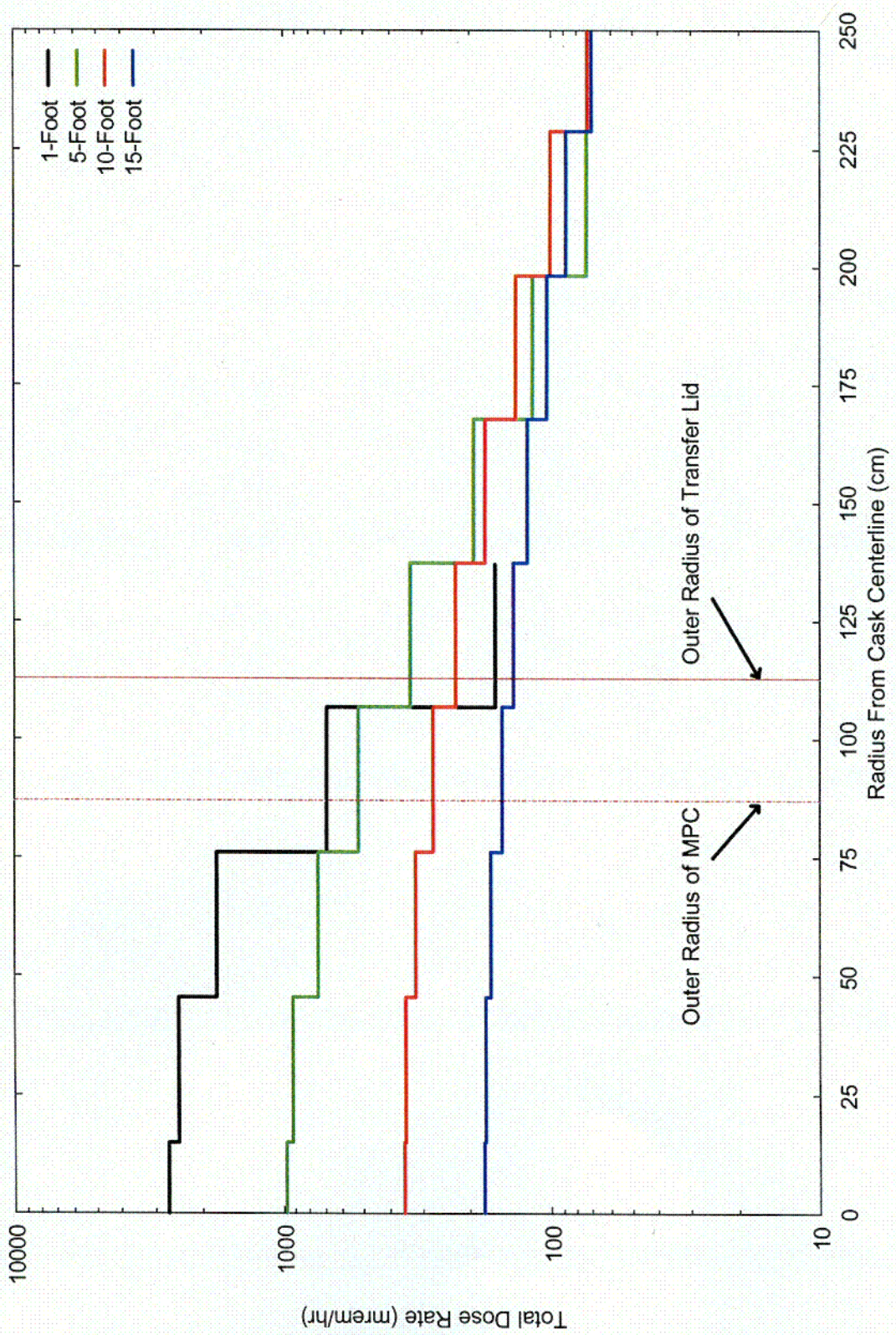


FIGURE 5.1.11; DOSE RATE AT VARIOUS DISTANCES FROM THE BOTTOM OF TRANSFER LID ON THE 100-TON HI-TRAC TRANSFER CASK WITH THE MPC-24 FOR 35,000 MWD/MTU AND 5-YEAR COOLING

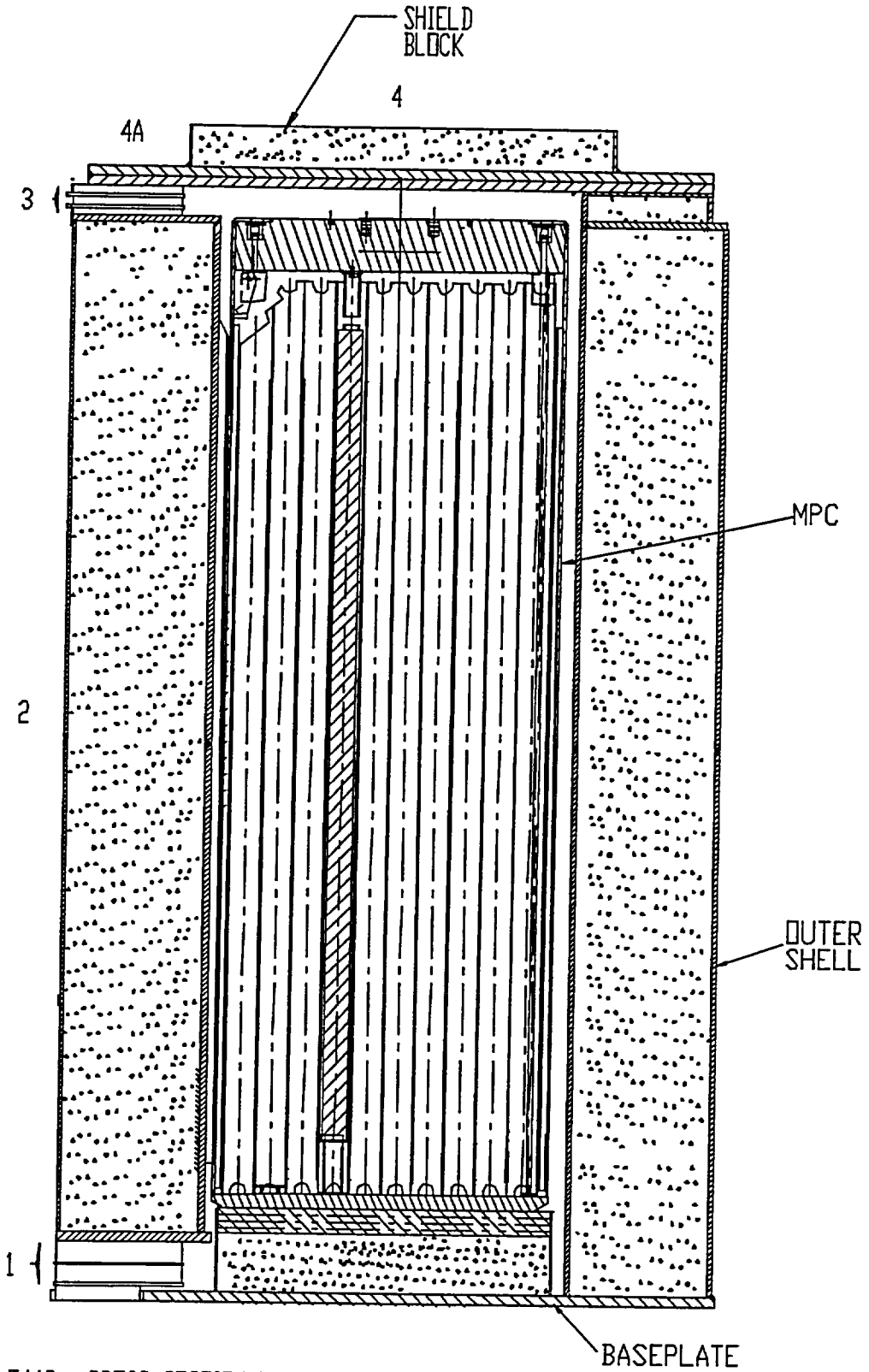


FIGURE 5.1.2; CROSS SECTION ELEVATION VIEW OF THE HI-STORM 100S OVERPACK WITH DOSE POINT LOCATION

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 4.3 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] 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 decay 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 Appendices 5.A and 5.B, respectively. 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 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 zircaloy clad fuel for the source term calculations is provided in Table 5.2.1. The PWR fuel assembly described is the assembly that produces the highest neutron and gamma sources and the highest decay heat load from the following fuel assembly classes listed in Table 2.1.1: B&W 15x15, B&W 17x17, CE 14x14, CE 16x16, WE 14x14, WE 15x15, WE 17x17, St. Lucie, and Ft. Calhoun. The BWR fuel assembly described is the assembly that produces the highest neutron and gamma sources and the highest decay heat load from the following fuel assembly classes listed in Table 2.1.2: GE BWR/2-3, GE BWR/4-6, Humboldt Bay 7x7, and Dresden 1 8x8. Multiple SAS2H and ORIGEN-S calculations were performed to confirm that the B&W 15x15 and the GE 7x7, which have the highest UO_2 mass, bound all other PWR and BWR fuel assemblies, respectively. Section 5.2.5 discusses, in detail, the determination of the design basis fuel assemblies.

The design basis Humboldt Bay and Dresden 1 6x6 fuel assembly is described in Table 5.2.2. The fuel assembly type listed produces the highest total neutron and gamma sources from the fuel assemblies at Dresden 1 and Humboldt Bay. Table 5.2.21 provides a description of the design basis Dresden 1 MOX fuel assembly used in this analysis. The design basis 6x6 and MOX fuel assemblies which are smaller than the GE 7x7, are assumed to have the same hardware characteristics as the GE 7x7. This is conservative because the larger hardware mass of the GE 7x7 results in a larger ^{60}Co activity.

The design basis stainless steel clad fuel assembly for the Indian Point 1, Haddam Neck and San Onofre 1 assembly classes is described in Table 5.2.3. This table also describes the design basis stainless steel clad LaCrosse fuel assembly.

The design basis assemblies mentioned above are the design basis assemblies for both intact and damaged fuel and fuel debris for their respective array classes. Analyses of damaged fuel is presented in Section 5.4.2.

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 Tables 5.2.1, 5.2.2, 5.2.3, and 5.2.21 resulted in conservative source term calculations.

Sections 5.2.1 and 5.2.2 describe the calculation of gamma and neutron source terms for zircaloy clad fuel while Section 5.2.3 discusses the calculation of the gamma and neutron source terms for the stainless steel clad fuel.

5.2.1 Gamma Source

Tables 5.2.4 through 5.2.6 provide the gamma source in MeV/s and photons/s as calculated with SAS2H and ORIGEN-S for the design basis zircaloy clad fuels at varying burnups and cooling times. Tables 5.2.7 and 5.2.22 provides the gamma source in MeV/s and photons/s for the design basis 6x6 and MOX fuel, respectively.

Specific analysis for the HI-STORM 100 System, which includes the HI-STORM storage overpacks and the HI-TRAC transfer casks, was performed to determine the dose contribution from gammas as a function of energy. This analysis considered dose locations external to the 100-ton HI-TRAC transfer cask and the HI-STORM 100 overpack and vents. The results of this analysis have revealed that, due to the magnitude of the gamma source at lower energies, gammas with energies as low as 0.45 MeV must be included in the shielding analysis. The effect of gammas with energies above 3.0 MeV, on the other hand, was found to be insignificant (less than 1% of the total gamma dose at all high dose locations). This is due to the fact that the source of gammas in this range (i.e., above 3.0 MeV) is extremely low (less than 1% of the total source). Therefore, all gammas with energies in the range of 0.45 to 3.0 MeV are included in the shielding calculations. Dose rate contributions from above and below this range were evaluated and found to be negligible. Photons with energies below 0.45 MeV are too weak to penetrate the HI-STORM overpack or HI-TRAC, and photons with energies above 3.0 MeV are too few to contribute significantly to the external dose.

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. Conservatively, the impurity level of ^{59}Co was assumed to be 1000 ppm or 1.0 gm/kg. Therefore, Inconel and stainless steel in the non-fuel regions are both conservatively assumed to have the same 1.0 gm/kg impurity level.

Holtec International has gathered information from utilities and vendors which shows that the 1.0 gm/kg impurity level is very conservative for fuel which has been manufactured since the mid-to-late 1980s after the implementation of an industry wide cobalt reduction program. The typical Cobalt-59 impurity level for fuel since the late 1980s is less than 0.5 gm/kg. Based on this, fuel with a short cooling time, 5 to 9 years, would have a Cobalt-59 impurity level less than 0.5 gm/kg. Therefore, the use of a bounding Cobalt-59 impurity level of 1.0 gm/kg is very conservative, particularly for recently manufactured assemblies. Analysis in Reference [5.2.3] indicates that the cobalt impurity in steel and inconel for fuel manufactured in the 1970s ranged from approximately 0.2 gm/kg to 2.2 gm/kg. However, older fuel manufactured with higher cobalt impurity levels will also have a corresponding longer cooling time and therefore will be bounded by the analysis presented in this chapter. As confirmation of this statement, Appendix D presents a comparison of the dose rates around the 100-ton HI-TRAC and the HI-STORM with the MPC-24 for a short cooling time (5 years) using the 1.0 gm/kg mentioned above and for a long cooling time (9 years) using a higher cobalt impurity level of 4.7 gm/kg for inconel. These results confirm that the dose rates for the longer cooling time with the higher impurity level are essentially equivalent to (within 11%) or bounded by the dose rates for the shorter cooling time with the lower impurity level. Therefore, the analysis in this chapter is conservative.

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 100 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 1 gm/kg (0.1 wt%) 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 are for an 8x8 fuel assembly. These masses are also appropriate for the 7x7 assembly since the masses of the non-fuel hardware from a 7x7 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 are larger than most other fuel assemblies from other manufacturers. This, in combination with the conservative ⁵⁹Co impurity level and the use of conservative flux weighting fractions (discussed below) results in an over-prediction of the non-fuel hardware source that bounds all fuel for which storage is requested.

The masses in Table 5.2.1 were used to calculate a ⁵⁹Co impurity level in the fuel assembly material. The grams of impurity were then used in ORIGEN-S to calculate a ⁶⁰Co 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 5.2.10. These scaling factors were taken from Reference [5.2.3].

Tables 5.2.11 through 5.2.13 provide the ^{60}Co activity utilized in the shielding calculations for the non-fuel regions of the assemblies in the MPC-32, MPC-24, and the MPC-68 for varying burnup and cooling times. The design basis 6x6 and MOX fuel assemblies are conservatively assumed to have the same ^{60}Co source strength as the BWR design basis fuel. This is a conservative assumption as the design basis 6x6 fuel and MOX fuel assemblies are limited to a significantly lower burnup and longer cooling time than the design basis fuel.

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 were chosen for the BWR and PWR design basis fuel assemblies. The enrichments are appropriately varied as a function of burnup. Table 5.2.24 presents the ^{235}U initial enrichments for various burnup ranges from 20,000 - 70,000 MWD/MTU for PWR and BWR zircaloy clad fuel. These enrichments are based on References [5.2.6] and [5.2.7]. Table 8 of reference [5.2.6] presents average enrichments for burnup ranges. The initial enrichments chosen in Table 5.2.24, for burnups up to 50,000 MWD/MTU, are approximately the average enrichments from Table 8 of reference [5.2.6] for the burnup range that is 5,000 MWD/MTU less than the ranges listed in Table 5.2.24. These enrichments are below the enrichments typically required to achieve the burnups that were analyzed. For burnups greater than 50,000 MWD/MTU, the data on historical and projected burnups available in the LWR Quantities Database in reference [5.2.7] was reviewed and conservatively low enrichments were chosen for each burnup range above 50,000 MWD/MTU.

Inherent to this approach of selecting minimum enrichments that bound the vast majority of discharged fuel is the fact that a small number of atypical assemblies will not be bounded. However, these atypical assemblies are very few in number (as evidenced by the referenced discharge data), and thus, it is unlikely that a single cask would contain several of these outlying assemblies. Further, because the approach is based on using minimum enrichments for given burnup ranges, any atypical assemblies that may exist are expected to have enrichments that are

very near to the minimum enrichments used in the analysis. Therefore, the result is an insignificant effect on the calculated dose rates. Consequently, the minimum enrichment values used in the analysis are adequate to bound the fuel authorized by the limits in the CoC for loading in the HI-STORM system. Therefore a minimum enrichment is not specified in the limits in the CoC. Since the enrichment does affect the source term evaluation, it is recommended that the site-specific dose evaluation consider the enrichment for the fuel being stored.

The neutron source calculated for the design basis fuel assemblies for the MPC-24, MPC-32, and MPC-68 and the design basis 6x6 fuel are listed in Tables 5.2.15 through 5.2.18 in neutrons/s for varying burnup and cooling times. Table 5.2.23 provides the neutron source in neutrons/sec for the design basis MOX fuel assembly. ^{244}Cm accounts for approximately 96% of the total number of neutrons produced, with slightly over 2% originating from (α, n) reactions within the UO_2 fuel. The remaining 2% derive from spontaneous fission in various Pu and Cm radionuclides. In addition, any neutrons generated from subcritical multiplication, $(n, 2n)$ or similar reactions are properly accounted for in the MCNP calculation.

5.2.3 Stainless Steel Clad Fuel Source

Table 5.2.3 lists the characteristics of the design basis stainless steel clad fuel. The fuel characteristics listed in this table are the input parameters that were used in the shielding calculations described in this chapter. The active fuel length listed in Table 5.2.3 is actually longer than the true active fuel length of 122 inches for the WE 15x15 and 83 inches for the LaCrosse 10x10. Since the true active fuel length is shorter than the design basis zircaloy clad active fuel length, it would be incorrect to calculate source terms for the stainless steel fuel using the correct fuel length and compare them directly to the zircaloy clad fuel source terms because this does not reflect the potential change in dose rates. As an example, if it is assumed that the source strength for both the stainless steel and zircaloy fuel is 144 neutrons/s and that the active fuel lengths of the stainless steel fuel and zircaloy fuel are 83 inches and 144 inches, respectively; the source strengths per inch of active fuel would be different for the two fuel types, 1.73 neutrons/s/inch and 1 neutron/s/inch for the stainless steel and zircaloy fuel, respectively. The result would be a higher neutron dose rate at the center of the cask with the stainless steel fuel than with the zircaloy clad fuel; a conclusion that would be overlooked by just comparing the source terms. This is an important consideration because the stainless steel clad fuel differs from the zircaloy clad in one important aspect: the stainless steel cladding will contain a significant photon source from Cobalt-60 which will be absent from the zircaloy clad fuel.

In order to eliminate the potential confusion when comparing source terms, the stainless steel clad fuel source terms were calculated with the same active fuel length as the design basis zircaloy clad fuel. Reference [5.2.2] indicates that the Cobalt-59 impurity level in steel is 800 ppm or 0.8 gm/kg. This impurity level was used for the stainless steel cladding in the source term calculations. It is assumed that the end fitting masses of the stainless steel clad fuel are the same as the end fitting masses of the zircaloy clad fuel. Therefore, separate source terms are not provided for the end fittings of the stainless steel fuel.

Tables 5.2.8, 5.2.9, 5.2.19, and 5.2.20 list the gamma and neutron source strengths for the design basis stainless steel clad fuel. It is obvious from these source terms that the neutron source strength for the stainless steel fuel is lower than for the zircaloy fuel. However, this is not true for all photon energy groups. The peak energy group is from 1.0 to 1.5 MeV, which results from the large Cobalt activation in the cladding. Since some of the source strengths are higher for the stainless steel fuel, Section 5.4.4 presents the dose rates at the center of the overpack for the stainless steel fuel. The center dose location is the only location of concern since the end fittings are assumed to be the same mass as the end fittings for the zircaloy clad fuel. In addition, the burnup is lower and the cooling time is longer for the stainless steel fuel compared to the zircaloy clad fuel.

5.2.4 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 100 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 to the inner four fuel storage locations in the MPC-24, MPC-24E, and the MPC-32.

5.2.4.1 BPRAs and TPDs

Burnable poison rod assemblies (BPRAs) (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 than fuel assemblies.

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 do not extend into the active fuel region with the exception of the W 14x14 water displacement guide tube plugs. 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 B&W 15x15 fuel assembly. The mass of material in the regions above the active fuel zone was scaled by the appropriate scaling factors listed in Table 5.2.10 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. For burnups beyond 45,000 MWD/MTU, it was assumed, for the purpose of the calculation, that the burned fuel assembly was replaced with a fresh fuel assembly every 45,000 MWD/MTU. This was achieved in ORIGEN-S by resetting the flux levels and cross sections to the 0 MWD/MTU condition after every 45,000 MWD/MTU.

Since the HI-STORM 100 cask system is designed to store many varieties of PWR fuel, a bounding TPD and BPRA had to be determined for the purposes of the analysis. This was accomplished 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 bounding TPD was determined to be the Westinghouse 17x17 guide tube plug and the bounding BPRA was actually determined by combining the higher masses of the Westinghouse 17x17 and 15x15 BPRAs into a singly hypothetical BPRA. The masses of this TPD and BPRA are listed in Table 5.2.30. As mentioned above, reference [5.2.5] describes the Westinghouse 14x14 water displacement guide tube plug as having a steel portion which extends into the active fuel zone. This particular water displacement guide tube plug was analyzed and determined to be bounded by the design basis TPD and BPRA.

Once the bounding BPRA and TPD were determined, the allowable Co-60 source from the BPRA and TPD were specified: 50 curies Co-60 for each TPD and 831 curies Co-60 for each BPRA. Table 5.2.31 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). An allowable burnup and cooling time, separate from the fuel assemblies, is used for BPRAs and TPDs. These burnup and cooling times assure that the Cobalt-60 activity remains below the allowable levels specified above. It should be noted that at very high burnups, greater than 200,000 MWD/MTU the TPD Co-60 source actually decreases as the burnup continues to increase. This is due to a decrease in the Cobalt-60 production rate as the initial Cobalt-59 impurity is being depleted. Conservatively, a constant cooling time has been specified for burnups from 180,000 to 630,000 MWD/MTU for the TPDs.

Section 5.4.6 discusses the increase in the cask dose rates due to the insertion of BPRAs or TPDs into fuel assemblies.

5.2.4.2 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 B₄C 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 to four CRAs and/or APSRs. These four devices are required to be stored in the inner four locations in the MPC-24, MPC-24E, MPC-24EF, and MPC-32 as outlined in Appendix B to the CoC.

In order to determine the impact on the dose rates around the HI-STORM 100 System, source terms for the CRAs and APSRs were calculated using SAS2H and ORIGEN-S. 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 1 kg of steel, inconel, and AgInCd using the flux calculated for the design basis B&W 15x15 fuel assembly. The total curies of cobalt for the steel and inconel and the 0.3-1.0 MeV source for the AgInCd were calculated as a function of burnup and cooling time to a maximum burnup of 630,000 MWD/MTU. For burnups beyond 45,000 MWD/MTU, it was assumed, for the purpose of the calculation, that the burned fuel assembly was replaced with a fresh fuel assembly every 45,000 MWD/MTU. This was achieved in ORIGEN-S by resetting the flux levels and cross sections to the 0 MWD/MTU condition after every 45,000 MWD/MTU. The sources were then scaled by the appropriate mass using the flux weighting factors for the different regions of the assembly to determine the final source term. Two different configurations were analyzed for both the CRAs and APSRs with an additional third configuration analyzed for the APSRs. The configurations, which are summarized below, are described in Tables 5.2.32 for the CRAs and Table 5.2.33 for the APSR. The masses of the materials listed in these tables were determined

from a review of [5.2.5] with bounding values chosen. The masses listed in Tables 5.2.32 and 5.2.33 do not match exact values from [5.2.5] because the values in the reference were adjusted to the lengths shown in the tables.

Configuration 1: CRA and APSR

This configuration had the lower 15 inches of the CRA and APSR activated at full flux with two regions above the 15 inches activated at a reduced power level. This simulates a CRA or APSR which was operated at 10% insertion. The regions above the 15 inches reflect the upper portion of the fuel assembly.

Configuration 2: CRA and APSR

This configuration represents a fully removed CRA or APSR during normal core operations. The activated portion corresponds to the upper portion of a fuel assembly above the active fuel length with the appropriate flux weighting factors used.

Configuration 3: APSR

This configuration represents a fully inserted gray APSR during normal core operations. The region in full flux was assumed to be the 63 inches of the absorber.

Tables 5.2.34 and 5.2.35 present the source terms that were calculated for the CRAs and APSRs respectively. The only significant source from the activation of inconel or steel is Co-60 and the only significant source from the activation of AgInCd is from 0.3-1.0 MeV. The source terms for CRAs, Table 5.2.34, were calculated for a maximum burnup of 630,000 MWD/MTU and a minimum cooling time of 5 years. Because of the significant source term in APSRs that have seen extensive in-core operations, the source term in Table 5.2.35 was calculated to be a bounding source term for a variable burnup and cooling time as outlined in Appendix B to the CoC. The very larger Cobalt-60 activity in configuration 3 in Table 5.2.35 is due to the assumed Cobalt-59 impurity level of 4.7 gm/kg. If this impurity level were similar to the assumed value for steel, 0.8 gm/kg, this source would decrease by approximately a factor of 5.8.

Section 5.4.6 discusses the effect on dose rate of the insertion of APSRs and CRAs into the inner four fuel assemblies in the MPC-24 or MPC-32.

5.2.5 Choice of Design Basis Assembly

The analysis presented in this chapter was performed to bound the fuel assembly classes listed in Tables 2.1.1 and 2.1.2. In order to perform a bounding analysis, a design basis fuel assembly must be chosen. Therefore, a fuel assembly from each fuel class was analyzed and a comparison of the neutrons/sec, photons/sec, and thermal power (watts) was performed. The fuel assembly that produced the highest source for a specified burnup, cooling time, and enrichment was chosen as the design basis fuel assembly. A separate design basis assembly was chosen for the PWR MPCs (MPC-24 and MPC-32) and the BWR MPCs (MPC-68).

5.2.5.1 PWR Design Basis Assembly

Table 2.1.1 lists the PWR fuel assembly classes that were evaluated to determine the design basis PWR fuel assembly. Within each class, the fuel assembly with the highest UO_2 mass was analyzed. Since the variations of fuel assemblies within a class are very minor (pellet diameter, clad thickness, etc.), it is conservative to choose the assembly with the highest UO_2 mass. For a given class of assemblies, the one with the highest UO_2 mass will produce the highest radiation source because, for a given burnup (MWD/MTU) and enrichment, the highest UO_2 mass will have produced the most energy and therefore the most fission products.

Table 5.2.25 presents the characteristics of the fuel assemblies analyzed to determine the design basis zircaloy clad PWR fuel assembly. The fuel assembly listed for each class is the assembly with the highest UO_2 mass. The St. Lucie and Ft. Calhoun classes are not present in Table 5.2.25. These assemblies are shorter versions of the CE 16x16 and CE 14x14 assembly classes, respectively. Therefore, these assemblies are bounded by the CE 16x16 and CE 14x14 classes and were not explicitly analyzed. Since the Indian Point 1, Haddam Neck, and San Onofre 1 classes are stainless steel clad fuel, these classes were analyzed separately and are discussed below. All fuel assemblies in Table 5.2.25 were analyzed at the same burnup and cooling time. The initial enrichment used in the analysis is consistent with Table 5.2.24. The results of the comparison are provided in Table 5.2.27. These results indicate that the B&W 15x15 fuel assembly has the highest radiation source term of the zircaloy clad fuel assembly classes considered in Table 2.1.1. This fuel assembly also has the highest UO_2 mass (see Table 5.2.25) which confirms that, for a given initial enrichment, burnup, and cooling time, the assembly with the highest UO_2 mass produces the highest radiation source term.

The Haddam Neck and San Onofre 1 classes are shorter stainless steel clad versions of the WE 15x15 and WE 14x14 classes, respectively. Since these assemblies have stainless steel clad, they were analyzed separately as discussed in Section 5.2.3. Based on the results in Table 5.2.27, which show that the WE 15x15 assembly class has a higher source term than the WE 14x14 assembly class, the Haddam Neck, WE 15x15, fuel assembly was analyzed as the bounding PWR stainless steel clad fuel assembly. The Indian Point 1 fuel assembly is a unique 14x14 design with a smaller mass of fuel and clad than the WE14x14. Therefore, it is also bounded by the WE 15x15 stainless steel fuel assembly.

5.2.5.2 BWR Design Basis Assembly

Table 2.1.2 lists the BWR fuel assembly classes that were evaluated to determine the design basis BWR fuel assembly. Since there are minor differences between the array types in the GE BWR/2-3 and GE BWR/4-6 assembly classes, these assembly classes were not considered individually but rather as a single class. Within that class, the array types, 7x7, 8x8, 9x9, and 10x10 were analyzed to determine the bounding BWR fuel assembly. Since the Humboldt Bay 7x7 and Dresden 1 8x8 are smaller versions of the 7x7 and 8x8 assemblies they are bounded by the 7x7 and 8x8 assemblies in the GE BWR/2-3 and GE BWR/4-6 classes. Within each array type, the fuel assembly with the highest UO_2 mass was analyzed. Since the variations of fuel

assemblies within an array type are very minor, it is conservative to choose the assembly with the highest UO_2 mass. For a given array type of assemblies, the one with the highest UO_2 mass will produce the highest radiation source because, for a given burnup (MWD/MTU) and enrichment, it will have produced the most energy and therefore the most fission products. The Humboldt Bay 6x6, Dresden 1 6x6, and LaCrosse assembly classes were not considered in the determination of the bounding fuel assembly. However, these assemblies were analyzed explicitly as discussed below.

Table 5.2.26 presents the characteristics of the fuel assemblies analyzed to determine the design basis zircaloy clad BWR fuel assembly. The fuel assembly listed for each array type is the assembly that has the highest UO_2 mass. All fuel assemblies in Table 5.2.26 were analyzed at the same burnup and cooling time. The initial enrichment used in these analyses is consistent with Table 5.2.24. The results of the comparison are provided in Table 5.2.28. These results indicate that the 7x7 fuel assembly has the highest radiation source term of the zircaloy clad fuel assembly classes considered in Table 2.1.2. This fuel assembly also has the highest UO_2 mass which confirms that, for a given initial enrichment, burnup, and cooling time, the assembly with the highest UO_2 mass produces the highest radiation source term. According to Reference [5.2.6], the last discharge of a 7x7 assembly was in 1985 and the maximum average burnup for a 7x7 during their operation was 29,000 MWD/MTU. This clearly indicates that the existing 7x7 assemblies have an average burnup and minimum cooling time that is well within the burnup and cooling time limits in Appendix B to the CoC. Therefore, the 7x7 assembly has never reached the burnup level analyzed in this chapter. However, in the interest of conservatism the 7x7 was chosen as the bounding fuel assembly array type.

Since the LaCrosse fuel assembly type is a stainless steel clad 10x10 assembly it was analyzed separately. The maximum burnup and minimum cooling time for this assembly are limited to 22,500 MWD/MTU and 10-year cooling as specified in Appendix B to the CoC. This assembly type is discussed further in Section 5.2.3.

The Humboldt Bay 6x6 and Dresden 1 6x6 fuel are older and shorter fuel than the other array types analyzed and therefore are considered separately. The Dresden 1 6x6 was chosen as the design basis fuel assembly for the Humboldt Bay 6x6 and Dresden 1 6x6 fuel assembly classes because it has the higher UO_2 mass. Dresden 1 also contains a few 6x6 MOX fuel assemblies, which were explicitly analyzed as well.

Reference [5.2.6] indicates that the Dresden 1 6x6 fuel assembly has a higher UO_2 mass than the Dresden 1 8x8 or the Humboldt Bay fuel (6x6 and 7x7). Therefore, the Dresden 1 6x6 fuel assembly was also chosen as the bounding assembly for damaged fuel and fuel debris for the Humboldt Bay and Dresden 1 fuel assembly classes.

Since the design basis 6x6 fuel assembly can be intact or damaged, the analysis presented in Section 5.4.2 for the damaged 6x6 fuel assembly also demonstrates the acceptability of storing intact 6x6 fuel assemblies from the Dresden 1 and Humboldt Bay fuel assembly classes.

5.2.5.3 Decay Heat Loads

Section 2.1.6 describes the calculation of the burnup versus cooling time limits in the CoC that are based on a maximum permissible decay heat per assembly. The decay heat values per assembly were calculated using the methodology described in Section 5.2. The design basis fuel assemblies, as described in Table 5.2.1, were used in the calculation of the burnup versus cooling time limits in the CoC. The enrichments used in the calculation of the decay heats were consistent with Table 5.2.24. As demonstrated in Tables 5.2.27 and 5.2.28, the design basis fuel assembly produces a higher decay heat value than the other assembly types considered. This is due to the higher heavy metal mass in the design basis fuel assemblies. Conservatively, Appendix B to the CoC limits the heavy metal mass to a value less than the design basis value utilized in this chapter. This provides additional assurance that the decay heat values are bounding values.

As further demonstration that the decay heat values (calculated using the design basis fuel assemblies) are conservative, a comparison between these calculated decay heats and the decay heats reported in Reference [5.2.7] are presented in Table 5.2.29. This comparison is made for a burnup of 30,000 MWD/MTU and a cooling time of 5 years. The burnup was chosen based on the limited burnup data available in Reference [5.2.7].

The heavy metal mass of the non-design basis fuel assembly classes in Appendix B of the Certificate of Compliance are limited to the masses used in Tables 5.2.25 and 5.2.26. No margin is applied between the allowable mass and the analyzed mass of heavy metal for the non-design basis fuel assemblies. This is acceptable because additional assurance that the decay heat values for the non-design basis fuel assemblies are bounding values is obtained by using the decay heat values for the design basis fuel assemblies to determine the acceptable storage criteria for all fuel assemblies. As mentioned above, Table 5.2.29 demonstrates the level of conservatism in applying the decay heat from the design basis fuel assembly to all fuel assemblies.

As mentioned above, the fuel assembly burnup and cooling times in Appendix B to the CoC were calculated using the decay heat limits which are also stipulated in Appendix B to the CoC. The burnup and cooling times for the non-fuel hardware, in Appendix B to the CoC, were chosen based on the radiation source term calculations discussed previously. The fuel assembly burnup and cooling times were calculated without consideration for the decay heat from BPRAs, TPDs, CRAs, or APSRs. This is acceptable since the user of the HI-STORM 100 system is required to demonstrate compliance with the assembly decay heat limits in Appendix B to the CoC regardless of the heat source (assembly or non-fuel hardware) and the actual decay heat from the non-fuel hardware is expected to be minimal. In addition, the shielding analysis presented in this chapter conservatively calculates the dose rates using both the burnup and cooling times for the fuel assemblies and non-fuel hardware. Therefore, the safety of the HI-STORM 100 system is guaranteed through the bounding analysis in this chapter, represented by the burnup and cooling time limits in the CoC, and the bounding thermal analysis in Chapter 4, represented by the decay heat limits in the CoC.

5.2.6 Thoria Rod Canister

Dresden Unit 1 has a single DFC containing 18 thoria rods which have obtained a relatively low burnup, 16,000 MWD/MTU. These rods were removed from two 8x8 fuel assemblies which contained 9 rods each. The irradiation of thorium produces an isotope which is not commonly found in depleted uranium fuel. Th-232 when irradiated produces U-233. The U-233 can undergo an (n,2n) reaction which produces U-232. The U-232 decays to produce Tl-208 which produces a 2.6 MeV gamma during Beta decay. This results in a significant source in the 2.5-3.0 MeV range which is not commonly present in depleted uranium fuel. Therefore, this single DFC container was analyzed to determine if it was bounded by the current shielding analysis.

A radiation source term was calculated for the 18 thoria rods using SAS2H and ORIGEN-S for a burnup of 16,000 MWD/MTU and a cooling time of 18 years. Table 5.2.36 describes the 8x8 fuel assembly that contains the thoria rods. Table 5.2.37 and 5.2.38 show the gamma and neutron source terms, respectively, that were calculated for the 18 thoria rods in the thoria rod canister. Comparing these source terms to the design basis 6x6 source terms for Dresden Unit 1 fuel in Tables 5.2.7 and 5.2.18 clearly indicates that the design basis source terms bound the thoria rods source terms in all neutron groups and in all gamma groups except the 2.5-3.0 MeV group. As mentioned above, the thoria rods have a significant source in this energy range due to the decay of Tl-208.

Section 5.4.8 provides a further discussion of the thoria rod canister and its acceptability for storage in the HI-STORM 100 System.

5.2.7 Fuel Assembly Neutron Sources

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

Dresden Unit 1 has a few antimony-beryllium neutron sources. These sources have been analyzed in Section 5.4.7 to demonstrate that they are acceptable for storage in the HI-STORM 100 System. Currently these are the only neutron source permitted for storage in the HI-STORM 100 System.

5.2.8 Stainless Steel Channels

The LaCrosse nuclear plant used two types of channels for their BWR assemblies: stainless steel and zircaloy. Since the irradiation of zircaloy does not produce significant activation, there are no restrictions on the storage of these channels and they are not explicitly analyzed in this chapter. The stainless steel channels, however, can produce a significant amount of activation, predominantly from Co-60. LaCrosse has thirty-two stainless steel channels, a few of which, have been in the reactor core for, approximately, the lifetime of the plant. Therefore, the

activation of the stainless steel channels was conservatively calculated to demonstrate that they are acceptable for storage in the HI-STORM 100 system. For conservatism, the number of stainless steel channels in an MPC-68 is being limited to sixteen and Appendix B to the CoC requires that these channels be stored in the inner sixteen locations.

The activation of a single stainless steel channel was calculated by simulating the irradiation of the channels with ORIGEN-S using the flux calculated from the LaCrosse fuel assembly. The mass of the steel channel in the active fuel zone (83 inches) was used in the analysis. For burnups beyond 22,500 MWD/MTU, it was assumed, for the purpose of the calculation, that the burned fuel assembly was replaced with a fresh fuel assembly every 22,500 MWD/MTU. This was achieved in ORIGEN-S by resetting the flux levels and cross sections to the 0 MWD/MTU condition after every 22,500 MWD/MTU.

LaCrosse was commercially operated from November 1969 until it was shutdown in April 1987. Therefore, the shortest cooling time for the assemblies and the channels is 13 years. Assuming the plant operated continually from 11/69 until 4/87, approximately 17.5 years or 6388 days, the accumulated burnup for the channels would be 186,000 MWD/MTU (6388 days times 29.17 MW/MTU from Table 5.2.3). Therefore, the cobalt activity calculated for a single stainless steel channel irradiated for 180,000 MWD/MTU was calculated to be 667 curies of Co-60 for 13 years cooling. This is equivalent to a source of $4.94\text{E}+13$ photons/sec in the energy range of 1.0-1.5 MeV.

In order to demonstrate that sixteen stainless steel channels are acceptable for storage in an MPC-68, a comparison of source terms is performed. Table 5.2.8 indicates that the source term for the LaCrosse design basis fuel assembly in the 1.0-1.5 MeV range is $6.34\text{E}+13$ photons/sec for 10 years cooling, assuming a 144 inch active fuel length. This is equivalent to $4.31\text{E}+15$ photons/sec/cask. At 13 years cooling, the fuel source term in that energy range decreases to $4.31\text{E}+13$ photons/sec which is equivalent to $2.93\text{E}+15$ photons/sec/cask. If the source term from the stainless steel channels is scaled to 144 inches and added to the 13 year fuel source term the result is $4.30\text{E}+15$ photons/sec/cask ($2.93\text{E}+15$ photons/sec/cask + $4.94\text{E}+13$ photons/sec/channel x 144 inch/83 inch x 16 channels/cask). This number is equivalent to the 10 year $4.31\text{E}+15$ photons/sec/cask source calculated from Table 5.2.8 and used in the shielding analysis in this chapter. Therefore, it is concluded that the storage of 16 stainless steel channels in an MPC-68 is acceptable.

Table 5.2.1

DESCRIPTION OF DESIGN BASIS ZIRCALOY CLAD FUEL

	PWR	BWR
Assembly type/class	B&W 15x15	GE 7x7
Active fuel length (in.)	144	144
No. of fuel rods	208	49
Rod pitch (in.)	0.568	0.738
Cladding material	Zircaloy-4	Zircaloy-2
Rod diameter (in.)	0.428	0.570
Cladding thickness (in.)	0.0230	0.0355
Pellet diameter (in.)	0.3742	0.488
Pellet material	UO ₂	UO ₂
Pellet density (gm/cc)	10.412 (95% of theoretical)	10.412 (95% of theoretical)
Enrichment (w/o ²³⁵ U)	3.6	3.2
Burnup (MWD/MTU) [†]	52,500 (MPC-24) 45,000 (MPC-32)	47,500 (MPC-68)
Cooling Time (years) [†]	5 (MPC-24 and 32)	5 (MPC-68)
Specific power (MW/MTU)	40	30
Weight of UO ₂ (kg) ^{††}	562.029	225.177
Weight of U (kg) ^{††}	495.485	198.516

Notes:

1. The B&W 15x15 is the design basis assembly for the following fuel assembly classes listed in Table 2.1.1: B&W 15x15, B&W 17x17, CE 14x14, CE 16x16, WE 14x14, WE 15x15, WE 17x17, St. Lucie, and Ft. Calhoun.
2. The GE 7x7 is the design basis assembly for the following fuel assembly classes listed in Table 2.1.2: GE BWR/2-3, GE BWR/4-6, Humboldt Bay 7x7, and Dresden 1 8x8.

[†] Burnup and cooling time combinations conservatively bound the acceptable burnup and cooling times listed in Appendix B to the CoC.

^{††} Derived from parameters in this table.

Table 5.2.1 (continued)

DESCRIPTION OF DESIGN BASIS FUEL

	PWR	BWR
No. of Water Rods	17	0
Water Rod O.D. (in.)	0.53	N/A
Water Rod Thickness (in.)	0.016	N/A
Lower End Fitting (kg)	8.16 (steel) 1.3 (inconel)	4.8 (steel)
Gas Plenum Springs (kg)	0.48428 (inconel) 0.23748 (steel)	1.1 (steel)
Gas Plenum Spacer (kg)	0.82824	N/A
Expansion Springs (kg)	N/A	0.4 (steel)
Upper End Fitting (kg)	9.28 (steel)	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

DESCRIPTION OF DESIGN BASIS GE 6x6 ZIRCALOY CLAD FUEL

	BWR
Fuel type	GE 6x6
Active fuel length (in.)	110
No. of fuel rods	36
Rod pitch (in.)	0.694
Cladding material	Zircaloy-2
Rod diameter (in.)	0.5645
Cladding thickness (in.)	0.035
Pellet diameter (in.)	0.494
Pellet material	UO ₂
Pellet density (gm/cc)	10.412 (95% of theoretical)
Enrichment (w/o ²³⁵ U)	2.24
Burnup (MWD/MTU)	30,000
Cooling Time (years)	18
Specific power (MW/MTU)	16.5
Weight of UO ₂ (kg) [†]	129.5
Weight of U (kg) [†]	114.2

Notes:

1. The 6x6 is the design basis damaged fuel assembly for the Humboldt Bay (all array types) and the Dresden 1 (all array types) damaged fuel assembly classes. It is also the design basis fuel assembly for the intact Humboldt Bay 6x6 and Dresden 1 6x6 fuel assembly classes.
2. This design basis damaged fuel assembly is also the design basis fuel assembly for fuel debris.

[†] Derived from parameters in this table.

Table 5.2.3

DESCRIPTION OF DESIGN BASIS STAINLESS STEEL CLAD FUEL

	PWR	BWR
Fuel type	WE 15x15	LaCrosse 10x10
Active fuel length (in.)	144	144
No. of fuel rods	204	100
Rod pitch (in.)	0.563	0.565
Cladding material	304 SS	348H SS
Rod diameter (in.)	0.422	0.396
Cladding thickness (in.)	0.0165	0.02
Pellet diameter (in.)	0.3825	0.35
Pellet material	UO ₂	UO ₂
Pellet density (gm/cc)	10.412 (95% of theoretical)	10.412 (95% of theoretical)
Enrichment (w/o ²³⁵ U)	3.5	3.5
Burnup (MWD/MTU) [†]	40,000 (MPC-24 and 32)	22,500 (MPC-68)
Cooling Time (years) [†]	8 (MPC-24), 9 (MPC-32)	10 (MPC-68)
Specific power (MW/MTU)	37.96	29.17
No. of Water Rods	21	0
Water Rod O.D. (in.)	0.546	N/A
Water Rod Thickness (in.)	0.017	N/A

Notes:

1. The WE 15x15 is the design basis assembly for the following fuel assembly classes listed in Table 2.1.1: Indian Point 1, Haddam Neck, and San Onofre 1.
2. The LaCrosse 10x10 is the design basis assembly for the following fuel assembly class listed in Table 2.1.2: LaCrosse.

[†] Burnup and cooling time combinations are equivalent to or conservatively bound the limits in Appendix B to the CoC.

Table 5.2.4

CALCULATED MPC-32 PWR FUEL GAMMA SOURCE PER ASSEMBLY
 FOR DESIGN BASIS ZIRCALOY CLAD FUEL
 FOR VARYING BURNUPS AND COOLING TIMES

Lower Energy (MeV)	Upper Energy (MeV)	32,500 MWD/MTU 5 Year Cooling		45,000 MWD/MTU 5 Year Cooling		45,000 MWD/MTU 10 Year Cooling	
		(MeV/s)	(Photons/s)	(MeV/s)	(Photons/s)	(MeV/s)	(Photons/s)
0.45	0.7	1.47E+15	2.56E+15	2.09E+15	3.63E+15	1.33E+15	2.32E+15
0.7	1.0	4.49E+14	5.28E+14	7.06E+14	8.31E+14	1.62E+14	1.91E+14
1.0	1.5	1.07E+14	8.53E+13	1.62E+14	1.30E+14	6.79E+13	5.43E+13
1.5	2.0	7.51E+12	4.29E+12	9.97E+12	5.70E+12	3.35E+12	1.92E+12
2.0	2.5	6.42E+12	2.86E+12	7.06E+12	3.14E+12	1.34E+11	5.97E+10
2.5	3.0	2.38E+11	8.67E+10	2.89E+11	1.05E+11	1.02E+10	3.71E+09
Total		2.04E+15	3.18E+15	2.97E+15	4.60E+15	1.57E+15	2.57E+15

Table 5.2.5

CALCULATED MPC-24 PWR FUEL GAMMA SOURCE PER ASSEMBLY
 FOR DESIGN BASIS ZIRCALOY CLAD FUEL
 FOR VARYING BURNUPS AND COOLING TIMES

Lower Energy	Upper Energy	42,500 MWD/MTU 5 Year Cooling		52,500 MWD/MTU 5 Year Cooling		57,500 MWD/MTU 12 Year Cooling	
		(MeV/s)	(Photons/s)	(MeV/s)	(Photons/s)	(MeV/s)	(Photons/s)
0.45	0.7	1.97E+15	3.42E+15	2.47E+15	4.29E+15	1.55E+15	2.69E+15
0.7	1.0	6.54E+14	7.70E+14	8.78E+14	1.03E+15	1.36E+14	1.61E+14
1.0	1.5	1.51E+14	1.21E+14	1.99E+14	1.59E+14	7.44E+13	5.95E+13
1.5	2.0	9.51E+12	5.43E+12	1.15E+13	6.56E+12	3.82E+12	2.18E+12
2.0	2.5	6.97E+12	3.10E+12	7.29E+12	3.24E+12	4.16E+10	1.85E+10
2.5	3.0	2.82E+11	1.03E+11	3.17E+11	1.15E+11	4.17E+09	1.52E+09
Total		2.79E+15	4.32E+15	3.56E+15	5.49E+15	1.76E+15	2.91E+15

Table 5.2.6

CALCULATED MPC-68 BWR FUEL GAMMA SOURCE PER ASSEMBLY
 FOR DESIGN BASIS ZIRCALOY CLAD FUEL
 FOR VARYING BURNUPS AND COOLING TIMES

Lower Energy (MeV)	Upper Energy (MeV)	40,000 MWD/MTU 5 Year Cooling		47,500 MWD/MTU 5 Year Cooling		50,000 MWD/MTU 10 Year Cooling	
		(MeV/s)	(Photons/s)	(MeV/s)	(Photons/s)	(MeV/s)	(Photons/s)
0.45	0.7	7.15E+14	1.24E+15	8.58E+14	1.49E+15	5.82E+14	1.01E+15
0.7	1.0	2.25E+14	2.64E+14	2.85E+14	3.36E+14	6.82E+13	8.03E+13
1.0	1.5	5.14E+13	4.11E+13	6.38E+13	5.10E+13	2.82E+13	2.25E+13
1.5	2.0	3.18E+12	1.82E+12	3.69E+12	2.11E+12	1.38E+12	7.90E+11
2.0	2.5	2.19E+12	9.75E+11	2.26E+12	1.00E+12	4.57E+10	2.03E+10
2.5	3.0	9.40E+10	3.42E+10	1.05E+11	3.82E+10	3.72E+09	1.35E+09
Total		9.96E+14	1.55E+15	1.21E+15	1.88E+15	6.79E+14	1.12E+15

Table 5.2.7

CALCULATED MPC-68 BWR FUEL GAMMA SOURCE PER ASSEMBLY
FOR DESIGN BASIS ZIRCALOY CLAD GE 6x6 FUEL

Lower Energy	Upper Energy	30,000 MWD/MTU 18-Year Cooling	
		(MeV/s)	(Photons/s)
4.5e-01	7.0e-01	1.53e+14	2.65e+14
7.0e-01	1.0	3.97e+12	4.67e+12
1.0	1.5	3.67e+12	2.94e+12
1.5	2.0	2.20e+11	1.26e+11
2.0	2.5	1.35e+09	5.99e+08
2.5	3.0	7.30e+07	2.66e+07
Totals		1.61e+14	2.73e+14

Table 5.2.8

CALCULATED BWR FUEL GAMMA SOURCE PER ASSEMBLY
FOR STAINLESS STEEL CLAD FUEL

Lower Energy	Upper Energy	22,500 MWD/MTU 10-Year Cooling	
		(MeV/s)	(Photons/s)
(MeV)	(MeV)		
4.5e-01	7.0e-01	2.72e+14	4.74e+14
7.0e-01	1.0	1.97e+13	2.31e+13
1.0	1.5	7.93e+13	6.34e+13
1.5	2.0	4.52e+11	2.58e+11
2.0	2.5	3.28e+10	1.46e+10
2.5	3.0	1.69e+9	6.14e+8
Totals		3.72e+14	5.61e+14

Note: These source terms were calculated for a 144-inch fuel length. The limits in Appendix B to the CoC are based on the actual 83-inch active fuel length.

Table 5.2.9

CALCULATED PWR FUEL GAMMA SOURCE PER ASSEMBLY
FOR STAINLESS STEEL CLAD FUEL

Lower Energy	Upper Energy	40,000 MWD/MTU 8-Year Cooling		40,000 MWD/MTU 9-Year Cooling	
(MeV)	(MeV)	(MeV/s)	(Photons/s)	(MeV/s)	(Photons/s)
4.5e-01	7.0e-01	1.37e+15	2.38e+15	1.28E+15	2.22E+15
7.0e-01	1.0	2.47e+14	2.91e+14	1.86E+14	2.19E+14
1.0	1.5	4.59e+14	3.67e+14	4.02E+14	3.21E+14
1.5	2.0	3.99e+12	2.28e+12	3.46E+12	1.98E+12
2.0	2.5	5.85e+11	2.60e+11	2.69E+11	1.20E+11
2.5	3.0	3.44e+10	1.25e+10	1.77E+10	6.44E+09
Totals		2.08e+15	3.04e+15	1.87E+15	2.76E+15

Note: These source terms were calculated for a 144-inch fuel length. The limits in Appendix B to the CoC are based on the actual 122-inch active fuel length.

Table 5.2.10

SCALING FACTORS USED IN CALCULATING THE ⁶⁰Co 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 5.2.11

CALCULATED MPC-32 ⁶⁰Co SOURCE PER ASSEMBLY FOR DESIGN BASIS
ZIRCALOY CLAD FUEL
AT DESIGN BASIS BURNUP AND COOLING TIME

Location	32,500 MWD/MTU and 5-Year Cooling (curies)	45,000 MWD/MTU and 5-Year Cooling (curies)	45,000 MWD/MTU and 10-Year Cooling (curies)
Lower End Fitting	139.25	167.06	86.46
Gas Plenum Springs	10.62	12.75	6.60
Gas Plenum Spacer	6.10	7.31	3.79
Expansion Springs	N/A	N/A	N/A
Incore Grid Spacers	360.64	432.67	223.93
Upper End Fitting	68.30	81.94	42.41
Handle	N/A	N/A	N/A

Table 5.2.12

CALCULATED MPC-24 ⁶⁰Co SOURCE PER ASSEMBLY FOR DESIGN BASIS
 ZIRCALOY CLAD FUEL
 AT DESIGN BASIS BURNUP AND COOLING TIME

Location	42,500 MWD/MTU and 5-Year Cooling (curies)	52,500 MWD/MTU and 5-Year Cooling (curies)	57,500 MWD/MTU and - 12 Year Cooling (curies)
Lower End Fitting	163.47	183.33	76.06
Gas Plenum Springs	12.47	13.99	5.80
Gas Plenum Spacer	7.16	8.03	3.33
Expansion Springs	N/A	N/A	N/A
Incore Grid Spacers	423.36	474.81	196.98
Upper End Fitting	80.18	89.92	37.31
Handle	N/A	N/A	N/A

Table 5.2.13

CALCULATED MPC-68 ⁶⁰Co SOURCE PER ASSEMBLY FOR DESIGN BASIS
 ZIRCALOY CLAD FUEL
 AT DESIGN BASIS BURNUP AND COOLING TIME

Location	40,000 MWD/MTU and 5-Year Cooling (curies)	47,500 MWD/MTU and 5-Year Cooling (curies)	50,000 MWD/MTU and - 10 Year Cooling (curies)
Lower End Fitting	63.49	71.35	36.00
Gas Plenum Springs	19.40	21.80	11.00
Gas Plenum Spacer	N/A	N/A	N/A
Expansion Springs	3.53	3.96	2.00
Grid Spacer Springs	29.10	32.70	16.50
Upper End Fitting	17.64	19.82	10.00
Handle	2.20	2.48	1.25

Table 5.2.14

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Table 5.2.15

CALCULATED MPC-32 PWR NEUTRON SOURCE PER ASSEMBLY
 FOR DESIGN BASIS ZIRCALOY CLAD FUEL
 FOR VARYING BURNUPS AND COOLING TIMES

Lower Energy (MeV)	Upper Energy (MeV)	32,500 MWD/MTU 5-Year Cooling (Neutrons/s)	45,000 MWD/MTU 5-Year Cooling (Neutrons/s)	45,000 MWD/MTU 10-Year Cooling (Neutrons/s)
1.0e-01	4.0e-01	6.35E+06	1.63E+07	1.35E+07
4.0e-01	9.0e-01	3.24E+07	8.33E+07	6.89E+07
9.0e-01	1.4	2.98E+07	7.63E+07	6.31E+07
1.4	1.85	2.20E+07	5.62E+07	4.66E+07
1.85	3.0	3.90E+07	9.92E+07	8.25E+07
3.0	6.43	3.52E+07	9.01E+07	7.46E+07
6.43	20.0	3.11E+06	7.98E+06	6.60E+06
Totals		1.68E+08	4.29E+08	3.56E+08

Table 5.2.16

CALCULATED MPC-24 PWR NEUTRON SOURCE PER ASSEMBLY
 FOR DESIGN BASIS ZIRCALOY CLAD FUEL
 FOR VARYING BURNUPS AND COOLING TIMES

Lower Energy (MeV)	Upper Energy (MeV)	42,500 MWD/MTU 5-Year Cooling (Neutrons/s)	52,500 MWD/MTU 5-Year Cooling (Neutrons/s)	57,500 MWD/MTU 12-Year Cooling (Neutrons/s)
1.0e-01	4.0e-01	1.42E+07	2.64E+07	2.52E+07
4.0e-01	9.0e-01	7.26E+07	1.35E+08	1.29E+08
9.0e-01	1.4	6.65E+07	1.24E+08	1.18E+08
1.4	1.85	4.90E+07	9.09E+07	8.69E+07
1.85	3.0	8.66E+07	1.60E+08	1.54E+08
3.0	6.43	7.86E+07	1.46E+08	1.39E+08
6.43	20.0	6.96E+06	1.29E+07	1.24E+07
Totals		3.75E+08	6.95E+08	6.64E+08

Table 5.2.17

CALCULATED MPC-68 BWR NEUTRON SOURCE PER ASSEMBLY
 FOR DESIGN BASIS ZIRCALOY CLAD FUEL
 FOR VARYING BURNUPS AND COOLING TIMES

Lower Energy (MeV)	Upper Energy (MeV)	40,000 MWD/MTU 5-Year Cooling (Neutrons/s)	47,500 MWD/MTU 5-Year Cooling (Neutrons/s)	50,000 MWD/MTU 10-Year Cooling (Neutrons/s)
1.0e-01	4.0e-01	5.03E+06	9.02E+06	7.43E+06
4.0e-01	9.0e-01	2.57E+07	4.61E+07	3.80E+07
9.0e-01	1.4	2.35E+07	4.22E+07	3.48E+07
1.4	1.85	1.73E+07	3.11E+07	2.56E+07
1.85	3.0	3.06E+07	5.47E+07	4.52E+07
3.0	6.43	2.78E+07	4.98E+07	4.11E+07
6.43	20.0	2.46E+06	4.42E+06	3.64E+06
Totals		1.32E+08	2.37E+08	1.96E+08

Table 5.2.18

CALCULATED MPC-68 BWR NEUTRON SOURCE PER ASSEMBLY
FOR DESIGN BASIS ZIRCALOY CLAD GE 6x6 FUEL

Lower Energy (MeV)	Upper Energy (MeV)	30,000 MWD/MTU 18-Year Cooling (Neutrons/s)
1.0e-01	4.0e-01	8.22e+5
4.0e-01	9.0e-01	4.20e+6
9.0e-01	1.4	3.87e+6
1.4	1.85	2.88e+6
1.85	3.0	5.18e+6
3.0	6.43	4.61e+6
6.43	20.0	4.02e+5
Total		2.20e+7

Table 5.2.19

CALCULATED BWR NEUTRON SOURCE PER ASSEMBLY
FOR STAINLESS STEEL CLAD FUEL

Lower Energy (MeV)	Upper Energy (MeV)	22,500 MWD/MTU 10-Year Cooling (Neutrons/s)
1.0e-01	4.0e-01	2.23e+5
4.0e-01	9.0e-01	1.14e+6
9.0e-01	1.4	1.07e+6
1.4	1.85	8.20e+5
1.85	3.0	1.56e+6
3.0	6.43	1.30e+6
6.43	20.0	1.08e+5
Total		6.22e+6

Note: These source terms were calculated for a 144-inch fuel length. The limits in Appendix B to the CoC are based on the actual 83-inch active fuel length.

Table 5.2.20

CALCULATED PWR NEUTRON SOURCE PER ASSEMBLY
FOR STAINLESS STEEL CLAD FUEL

Lower Energy (MeV)	Upper Energy (MeV)	40,000 MWD/MTU 8-Year Cooling (Neutrons/s)	40,000 MWD/MTU 9-Year Cooling (Neutrons/s)
1.0e-01	4.0e-01	1.04e+7	1.01E+07
4.0e-01	9.0e-01	5.33e+7	5.14E+07
9.0e-01	1.4	4.89e+7	4.71E+07
1.4	1.85	3.61e+7	3.48E+07
1.85	3.0	6.41e+7	6.18E+07
3.0	6.43	5.79e+7	5.58E+07
6.43	20.0	5.11e+6	4.92E+06
Totals		2.76e+8	2.66E+08

Note: These source terms were calculated for a 144-inch fuel length. The limits in Appendix B to the CoC are based on the actual 122-inch active fuel length.

Table 5.2.21

DESCRIPTION OF DESIGN BASIS ZIRCALOY CLAD MIXED OXIDE FUEL

	BWR
Fuel type	GE 6x6
Active fuel length (in.)	110
No. of fuel rods	36
Rod pitch (in.)	0.696
Cladding material	Zircaloy-2
Rod diameter (in.)	0.5645
Cladding thickness (in.)	0.036
Pellet diameter (in.)	0.482
Pellet material	UO ₂ and PuUO ₂
No. of UO ₂ Rods	27
No. of PuUO ₂ rods	9
Pellet density (gm/cc)	10.412 (95% of theoretical)
Enrichment (w/o ²³⁵ U) [†]	2.24 (UO ₂ rods) 0.711 (PuUO ₂ rods)
Burnup (MWD/MTU)	30,000
Cooling Time (years)	18
Specific power (MW/MTU)	16.5
Weight of UO ₂ ,PuUO ₂ (kg) ^{††}	123.3
Weight of U,Pu (kg) ^{††}	108.7

[†] See Table 5.3.3 for detailed composition of PuUO₂ rods.

^{††} Derived from parameters in this table

Table 5.2.22

CALCULATED MPC-68 BWR FUEL GAMMA SOURCE PER ASSEMBLY
FOR DESIGN BASIS ZIRCALOY CLAD MIXED OXIDE FUEL

Lower Energy	Upper Energy	30,000 MWD/MTU 18-Year Cooling	
		(MeV/s)	(Photons/s)
4.5e-01	7.0e-01	1.45e+14	2.52e+14
7.0e-01	1.0	3.87e+12	4.56e+12
1.0	1.5	3.72e+12	2.98e+12
1.5	2.0	2.18e+11	1.25e+11
2.0	2.5	1.17e+9	5.22e+8
2.5	3.0	9.25e+7	3.36e+7
Totals		1.53e+14	2.60e+14

Table 5.2.23

CALCULATED MPC-68 BWR NEUTRON SOURCE PER ASSEMBLY
FOR DESIGN BASIS ZIRCALOY CLAD MIXED OXIDE FUEL

Lower Energy (MeV)	Upper Energy (MeV)	30,000 MWD/MTU 18-Year Cooling (Neutrons/s)
1.0e-01	4.0e-01	1.24e+6
4.0e-01	9.0e-01	6.36e+6
9.0e-01	1.4	5.88e+6
1.4	1.85	4.43e+6
1.85	3.0	8.12e+6
3.0	6.43	7.06e+6
6.43	20.0	6.07e+5
Totals		3.37e+7

Table 5.2.24

INITIAL ENRICHMENTS USED IN THE SOURCE TERM CALCULATIONS

Burnup Range (MWD/MTU)	Initial Enrichment (wt. % ²³⁵ U)
BWR Fuel	
20,000-25,000	2.1
25,000-30,000	2.4
30,000-35,000	2.6
35,000-40,000	2.9
40,000-45,000	3.0
45,000-50,000	3.2
50,000-55,000	3.6
55,000-60,000	4.0
60,000-65,000	4.4
PWR Fuel	
20,000-25,000	2.3
25,000-30,000	2.6
30,000-35,000	2.9
35,000-40,000	3.2
40,000-45,000	3.4
45,000-50,000	3.6
50,000-55,000	3.9
55,000-60,000	4.2
60,000-65,000	4.5
65,000-70,000	4.8

Note: The burnup ranges do not overlap. Therefore, 20,000-25,000 MWD/MTU means 20,000-24,999.9 MWD/MTU, etc.

Table 5.2.25

DESCRIPTION OF EVALUATED ZIRCALOY CLAD PWR FUEL

Assembly class	WE 14x14	WE 15x15	WE 17x17	CE 14x14	CE 16x16	B&W 15x15	B&W 17x17
Active fuel length (in.)	144	144	144	144	150	144	144
No. of fuel rods	179	204	264	176	236	208	264
Rod pitch (in.)	0.556	0.563	0.496	0.580	0.5063	0.568	0.502
Cladding material	Zr-4	Zr-4	Zr-4	Zr-4	Zr-4	Zr-4	Zr-4
Rod diameter (in.)	0.422	0.422	0.374	0.440	0.382	0.428	0.377
Cladding thickness (in.)	0.0243	0.0245	0.0225	0.0280	0.0250	0.0230	0.0220
Pellet diameter (in.)	0.3659	0.366	0.3225	0.377	0.3255	0.3742	0.3252
Pellet material	UO ₂	UO ₂	UO ₂	UO ₂	UO ₂	UO ₂	UO ₂
Pellet density (gm/cc) (95% of theoretical)	10.412	10.412	10.412	10.412	10.412	10.412	10.412
Enrichment (wt.% ²³⁵ U)	3.4	3.4	3.4	3.4	3.4	3.4	3.4
Burnup (MWD/MTU)	40,000	40,000	40,000	40,000	40,000	40,000	40,000
Cooling time (years)	5	5	5	5	5	5	5
Specific power (MW/MTU)	40	40	40	40	40	40	40
Weight of UO ₂ (kg) [†]	462.451	527.327	529.848	482.706	502.609	562.029	538.757
Weight of U (kg) [†]	407.697	464.891	467.114	425.554	443.100	495.485	474.968
No. of Guide Tubes	17	21	25	5	5	17	25
Guide Tube O.D. (in.)	0.539	0.546	0.474	1.115	0.98	0.53	0.564
Guide Tube Thickness (in.)	0.0170	0.0170	0.0160	0.0400	0.0400	0.0160	0.0175

[†] Derived from parameters in this table.

Table 5.2.26

DESCRIPTION OF EVALUATED ZIRCALOY CLAD BWR FUEL

Array Type	7×7	8×8	9×9	10×10
Active fuel length (in.)	144	144	144	144
No. of fuel rods	49	64	74	92
Rod pitch (in.)	0.738	0.642	0.566	0.510
Cladding material	Zr-2	Zr-2	Zr-2	Zr-2
Rod diameter (in.)	0.570	0.484	0.440	0.404
Cladding thickness (in.)	0.0355	0.02725	0.0280	0.0260
Pellet diameter (in.)	0.488	0.4195	0.376	0.345
Pellet material	UO ₂	UO ₂	UO ₂	UO ₂
Pellet density (gm/cc) (% of theoretical)	10.412 (95%)	10.412 (95%)	10.5216 (96%)	10.5216 (96%)
Enrichment (wt.% ²³⁵ U)	3.0	3.0	3.0	3.0
Burnup (MWD/MTU)	40,000	40,000	40,000	40,000
Cooling time (years)	5	5	5	5
Specific power (MW/MTU)	30	30	30	30
Weight of UO ₂ (kg) [†]	225.177	217.336	204.006	213.531
Weight of U (kg) [†]	198.516	191.603	179.852	188.249
No. of Water Rods	0	0	2	2
Water Rod O.D. (in.)	n/a	n/a	0.980	0.980
Water Rod Thickness (in.)	n/a	n/a	0.0300	0.0300

[†] Derived from parameters in this table.

Table 5.2.27

COMPARISON OF SOURCE TERMS FOR ZIRCALOY CLAD PWR FUEL
3.4 wt.% ²³⁵U - 40,000 MWD/MTU - 5 years cooling

Assembly class	WE 14x14	WE 15x15	WE 17x17	CE 14x14	CE 16x16	B&W 15x15	B&W 17x17
Neutrons/sec	2.29e+8 / 2.31e+8	2.63e+8 / 2.65e+8	2.62e+8	2.31e+8	2.34e+8	2.94e+8	2.64e+8
Photons/sec (0.45-3.0 MeV)	3.28e+15 / 3.33e+15	3.74e+15 / 3.79e+15	3.76e+15	3.39e+15	3.54e+15	4.01e+15	3.82e+15
Thermal power (watts)	926.6 / 936.8	1056 / 1068	1062	956.6	995.7	1137	1077

Note:

The WE 14x14 and WE 15x15 have both zircaloy and stainless steel guide tubes. The first value presented is for the assembly with zircaloy guide tubes and the second value is for the assembly with stainless steel guide tubes.

Table 5.2.28

COMPARISON OF SOURCE TERMS FOR ZIRCALOY CLAD BWR FUEL
 3.0 wt.% ²³⁵U - 40,000 MWD/MTU - 5 years cooling

Assembly class	7×7	8×8	9×9	10×10
Neutrons/sec	1.33e+8	1.22e+8	1.13e+8	1.24e+8
Photons/sec (0.45-3.0 MeV)	1.55e+15	1.49e+15	1.40e+15	1.47e+15
Thermal power (watts)	435.5	417.3	389.4	411.5

Table 5.2.29

COMPARISON OF CALCULATED DECAY HEATS FOR DESIGN BASIS FUEL
AND VALUES REPORTED IN THE
DOE CHARACTERISTICS DATABASE[†] FOR
30,000 MWD/MTU AND 5-YEAR COOLING

Fuel Assembly Class	Decay Heat from the DOE Database (watts/assembly)	Decay Heat from Design Basis Fuel (watts/assembly)
PWR Fuel		
B&W 15x15	752.0	827.5
B&W 17x17	732.9	827.5
CE 16x16	653.7	827.5
CE 14x14	601.3	827.5
WE 17x17	742.5	827.5
WE 15x15	762.2	827.5
WE 14x14	649.6	827.5
BWR Fuel		
7x7	310.9	315.7
8x8	296.6	315.7
9x9	275.0	315.7

Notes:

1. The PWR and BWR design basis fuels are the B&W 15x15 and the GE 7x7, respectively.
2. The decay heat values from the database include contributions from in-core material (e.g. spacer grids).
3. Information on the 10x10 was not available in the DOE database. However, based on the results in Table 5.2.28, the actual decay heat values from the 10x10 would be very similar to the values shown above for the 8x8.

[†] Reference [5.2.7].

Table 5.2.30

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 5.2.31

DESIGN BASIS COBALT-60 ACTIVITIES FOR BURNABLE POISON ROD
ASSEMBLIES AND THIMBLE PLUG DEVICES

Region	BPRA	TPD
Upper End Fitting (curies Co-60)	30.4	25.21
Gas Plenum Spacer (curies Co-60)	4.6	9.04
Gas Plenum Springs (curies Co-60)	8.2	15.75
In-core (curies Co-60)	787.8	N/A

Table 5.2.32

DESCRIPTION OF DESIGN BASIS CONTROL ROD ASSEMBLY
CONFIGURATIONS FOR SOURCE TERM CALCULATIONS

Axial Dimensions Relative to Bottom of Active Fuel			Flux Weighting Factor	Mass of cladding (kg Inconel)	Mass of absorber (kg AgInCd)
Start (in)	Finish (in)	Length (in)			
Configuration 1 - 10% Inserted					
0.0	15.0	15.0	1.0	1.32	7.27
15.0	18.8125	3.8125	0.2	0.34	1.85
18.8125	28.25	9.4375	0.1	0.83	4.57
Configuration 2 - Fully Removed					
0.0	3.8125	3.8125	0.2	0.34	1.85
3.8125	13.25	9.4375	0.1	0.83	4.57

Table 5.2.33

DESCRIPTION OF DESIGN BASIS AXIAL POWER SHAPING ROD
CONFIGURATIONS FOR SOURCE TERM CALCULATIONS

Axial Dimensions Relative to Bottom of Active Fuel			Flux Weighting Factor	Mass of cladding (kg Steel)	Mass of absorber (kg Inconel)
Start (in)	Finish (in)	Length (in)			
Configuration 1 - 10% Inserted					
0.0	15.0	15.0	1.0	1.26	5.93
15.0	18.8125	3.8125	0.2	0.32	1.51
18.8125	28.25	9.4375	0.1	0.79	3.73
Configuration 2 - Fully Removed					
0.0	3.8125	3.8125	0.2	0.32	1.51
3.8125	13.25	9.4375	0.1	0.79	3.73
Configuration 3 - Fully Inserted					
0.0	63.0	63.0	1.0	5.29	24.89
63.0	66.8125	3.8125	0.2	0.32	1.51
66.8125	76.25	9.4375	0.1	0.79	3.73

Table 5.2.34

DESIGN BASIS SOURCE TERMS FOR CONTROL ROD
ASSEMBLY CONFIGURATIONS

Axial Dimensions Relative to Bottom of Active Fuel			Photons/sec from AgInCd			Curies Co-60 from Inconel
Start (in)	Finish (in)	Length (in)	0.3-0.45 MeV	0.45-0.7 MeV	0.7-1.0 MeV	
Configuration 1 - 10% Inserted						
0.0	15.0	15.0	1.91e+14	1.78e+14	1.42e+14	1111.38
15.0	18.8125	3.8125	9.71e+12	9.05e+12	7.20e+12	56.50
18.8125	28.25	9.4375	1.20e+13	1.12e+13	8.92e+12	69.92
Configuration 2 - Fully Removed						
0.0	3.8125	3.8125	9.71e+12	9.05e+12	7.20e+12	56.50
3.8125	13.25	9.4375	1.20e+13	1.12e+13	8.92e+12	69.92

Table 5.2.35

DESIGN BASIS SOURCE TERMS FROM AXIAL POWER
SHAPING ROD CONFIGURATIONS

Axial Dimensions Relative to Bottom of Active Fuel			Curies of Co-60
Start (in)	Finish (in)	Length (in)	
Configuration 1 - 10% Inserted			
0.0	15.0	15.0	2682.57
15.0	18.8125	3.8125	136.36
18.8125	28.25	9.4375	168.78
Configuration 2 - Fully Removed			
0.0	3.8125	3.8125	136.36
3.8125	13.25	9.4375	168.78
Configuration 3 - Fully Inserted			
0.0	63.0	63.0	11266.80
63.0	66.8125	3.8125	136.36
66.8125	76.25	9.4375	168.78

Table 5.2.36

DESCRIPTION OF FUEL ASSEMBLY USED TO ANNALYZE
THORIA RODS IN THE THORIA ROD CANISTER

	BWR
Fuel type	8x8
Active fuel length (in.)	110.5
No. of UO ₂ fuel rods	55
No. of UO ₂ /ThO ₂ fuel rods	9
Rod pitch (in.)	0.523
Cladding material	zircaloy
Rod diameter (in.)	0.412
Cladding thickness (in.)	0.025
Pellet diameter (in.)	0.358
Pellet material	98.2% ThO ₂ and 1.8% UO ₂ for UO ₂ /ThO ₂ rods
Pellet density (gm/cc)	10.412
Enrichment (w/o ²³⁵ U)	93.5 in UO ₂ for UO ₂ /ThO ₂ rods and 1.8 for UO ₂ rods
Burnup (MWD/MTIHM)	16,000
Cooling Time (years)	18
Specific power (MW/MTIHM)	16.5
Weight of ThO ₂ and UO ₂ (kg) [†]	121.46
Weight of U (kg) [†]	92.29
Weight of Th (kg) [†]	14.74

[†] Derived from parameters in this table.

Table 5.2.37

CALCULATED FUEL GAMMA SOURCE FOR THORIA ROD
CANISTER CONTAINING EIGHTEEN THORIA RODS

Lower Energy (MeV)	Upper Energy (MeV)	16,000 MWD/MTIHM 18-Year Cooling	
		(MeV/s)	(Photons/s)
4.5e-01	7.0e-01	3.07e+13	5.34e+13
7.0e-01	1.0	5.79e+11	6.81e+11
1.0	1.5	3.79e+11	3.03e+11
1.5	2.0	4.25e+10	2.43e+10
2.0	2.5	4.16e+8	1.85e+8
2.5	3.0	2.31e+11	8.39e+10
Totals		1.23e+12	1.09e+12

Table 5.2.38

CALCULATED FUEL NEUTRON SOURCE FOR THORIA ROD
CANISTER CONTAINING EIGHTEEN THORIA RODS

Lower Energy (MeV)	Upper Energy (MeV)	16,000 MWD/MTIHM 18-Year Cooling (Neutrons/s)
1.0e-01	4.0e-01	5.65e+2
4.0e-01	9.0e-01	3.19e+3
9.0e-01	1.4	6.79e+3
1.4	1.85	1.05e+4
1.85	3.0	3.68e+4
3.0	6.43	1.41e+4
6.43	20.0	1.60e+2
Totals		7.21e+4

5.3 MODEL SPECIFICATIONS

The shielding analysis of the HI-STORM 100 System was performed with MCNP-4A [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 100 System, including the HI-TRAC transfer casks, in the shielding analysis. A sample input file for MCNP is provided in Appendix 5.C.

As discussed in Section 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. Section 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. Section 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 100 System, including the HI-TRAC transfer casks. 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 through 5.3.6 show cross sectional views of the HI-STORM 100 overpack and MPC as it was modeled in MCNP for each of the MPCs. Figures 5.3.1 through 5.3.3 were created with the MCNP two-dimensional plotter 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. Figure 5.3.7 shows a cross sectional view of the 100-ton HI-TRAC with the MPC-24 inside as it was modeled in MCNP. Since the fins and pocket trunnions were modeled explicitly, neutron streaming through these components is accounted for in the calculations of the dose adjacent to the overpack and 1 meter dose. In Section 5.4.1, the dose effect of localized streaming through these compartments is analyzed.

Figure 5.3.10 shows a cross sectional view of the HI-STORM 100 overpack with the as-modeled thickness of the various materials. These dimensions are the same for the HI-STORM 100S overpack. Figures 5.3.11 and 5.3.18 are axial representations of the HI-STORM 100 and HI-STORM 100S overpacks, respectively, with the various as-modeled dimensions indicated.

Figures 5.3.12 and 5.3.13 show axial cross-sectional views of the 100- and 125-ton HI-TRAC transfer casks, respectively, with the as-modeled dimensions and materials specified. Figures 5.3.14, 5.3.15, and 5.3.20 show fully labeled radial cross-sectional views of the HI-TRAC 100, 125, and 125D transfer casks, respectively. Finally, Figures 5.3.16 and 5.3.17 show fully labeled

diagrams of the transfer lids for the HI-TRAC 100 and 125 transfer casks. Since lead plate may be used instead of poured lead in the pool and transfer lids, there exists the possibility of a gap between the lead plate and the surrounding steel walls. This gap was accounted for in the analysis as depicted on Figures 5.3.16 and 5.3.17. The gap was not modeled in the pool lid since the gap will only exist on the outer edges of the pool lid and the highest dose rate is in the center. (All results presented in this chapter were calculated with the gap with the exception of the results presented in Figures 5.1.6, 5.1.7, and 5.1.11 which did not include the gap.) The HI-TRAC 125D does not utilize the transfer lid, rather it utilizes the pool lid in conjunction with the mating device. Therefore the dose rates reported for the pool lid in this chapter are applicable to both the HI-TRAC 125 and 125D, while the dose rates reported for the transfer lid are applicable only to the HI-TRAC 125. Consistent with the analysis of the transfer lid in which only the portion of the lid directly below the MPC was modeled, the structure of the mating device which surrounds the pool lid was not modeled.

Since the HI-TRAC 125D has fewer radial ribs, the dose rate at the midplane of the HI-TRAC 125D is higher than the dose rate at the midplane of the HI-TRAC 125. The HI-TRAC 125D has steel ribs in the lower water jacket while the HI-TRAC 125 does not. These additional ribs in the lower water jacket reduce the dose rate in the vicinity of the pool lid for the HI-TRAC 125D compared to the HI-TRAC 125. Since the dose rates at the midplane of the HI-TRAC 125D are higher than the HI-TRAC 125, the results on the radial surface are only presented for the HI-TRAC 125D in this chapter.

To reduce the gamma dose around the inlet and outlet vents, stainless steel cross plates, designated gamma shield cross plates[†] (see Figures 5.3.11 and 5.3.18), have been installed inside all vents. The steel in these plates effectively attenuates the fuel and ⁶⁰Co gammas that dominated the dose at these locations prior to their installation. Figure 5.3.19 shows two designs for the gamma shield cross plates to be used in the inlet and outlet vents. The designs in the top portion of the figure are mandatory for use in the HI-STORM 100 and 100S overpacks during normal storage operations and were assumed to be in place in the shielding analysis. The designs in the bottom portion of the figure may be used instead of the mandatory designs in the HI-STORM 100S overpack to further reduce the radiation dose rates at the vents. These optional gamma shield cross plates could further reduce the dose rate at the vent openings by as much as a factor of two.

Calculations were performed to determine the acceptability of homogenizing the fuel assembly versus explicit modeling. Based on these calculations it was concluded that it was acceptable to homogenize the fuel assembly without loss of accuracy. The width of the PWR and BWR homogenized fuel assembly is equal to 15 times the pitch and 7 times the pitch, respectively. Homogenization resulted in a noticeable decrease in run time.

[†] This design embodiment, formally referred to as "Duct Photon Attenuator," has been disclosed as an invention by Holtec International for consideration by the US Patent Office for issuance of a patent under U.S. law.

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

1. The basket material in the top and bottom 0.9 inches where the MPC basket flow holes are located is not modeled. The length of the basket not modeled (0.9 inches) was determined by calculating the equivalent area removed by the flow holes. This method of approximation is conservative because no material for the basket shielding is provided in the 0.9-inch area at the top and bottom of the MPC basket.
2. The upper and lower fuel spacers are not modeled, as the fuel spacers are not needed on all fuel assembly types. However, most PWR fuel assemblies will have upper and lower fuel spacers. The fuel spacer length for the design basis fuel assembly type determines the positioning of the fuel assembly for the shielding analysis, but the fuel spacer materials are not modeled. This is conservative since it removes steel that would provide a small amount of additional shielding.
3. For the MPC-32, MPC-24, and MPC-68, the MPC basket supports are not modeled. This is conservative since it removes steel that would provide a small increase in shielding. The optional aluminum heat conduction elements are also conservatively not modeled.
4. The MPC-24 basket is fabricated from 5/16 inch thick cell plates. It is conservatively assumed for modeling purposes that the structural portion of the MPC-24 basket is uniformly fabricated from 9/32 inch thick steel. The Boral and sheathing are modeled explicitly. This is conservative since it removes steel that would provide a small amount of additional shielding.
5. In the modeling of the BWR fuel assemblies, the zircaloy flow channels were not represented. This was done because it cannot be guaranteed that all BWR fuel assemblies will have an associated flow channel when placed in the MPC. The flow channel does not contribute to the source, but does provide some small amount of shielding. However, no credit is taken for this additional shielding.
6. In the MPC-24, conservatively, all Boral panels on the periphery were modeled with a reduced width of 5 inches compared to 6.25 inches or 7.5 inches.

During this project several design changes occurred that affected the drawings, but did not significantly affect the MCNP models of the HI-STORM 100 and HI-TRAC. Therefore, the models do not exactly represent the drawings. The discrepancies between models and drawings are listed and discussed here.

MPC Modeling Discrepancies

1. In the MPCs, there is a sump in the baseplate to enhance draining of the MPC. This localized reduction in the thickness of the baseplate was not modeled. Since there is significant shielding and distance in both the HI-TRAC and the HI-STORM outside the MPC baseplate, this localized reduction in shielding will not affect the calculated dose rates outside the HI-TRAC or the HI-STORM.
2. The design configuration of the MPC-24 has been enhanced for criticality purposes. The general location of the 24 assemblies remains basically the same, therefore the shielding analysis continues to use the superseded configuration. Since the new MPC-24 configuration and the configuration of the MPC-24E are almost identical, the analysis of the earlier MPC-24 configuration is valid for the MPC-24E as well. Figure 5.3.21 shows the superseded and current configuration for the MPC-24 for comparison.
3. The sheathing thickness on the new MPC-24 configuration was reduced from 0.06 inches to 0.0235 inches. However, the model still uses 0.06 inches. This discrepancy is compensated for by the use of 9/32 inch cell walls and 5 inch boral on the periphery as described above. MCNP calculations were performed with the new MPC-24 configuration in the 100-ton HI-TRAC for comparison to the superseded configuration. These results indicate that on the side of the overpack, the dose rates decrease by approximately 12% on the surface. These results demonstrate that using the superseded MPC-24 design is conservative.

HI-TRAC Modeling Discrepancies

1. The pocket trunnion on the HI-TRAC 125 was modeled as penetrating the lead. This is conservative for gamma dose rates as it reduces effective shielding thickness. The HI-TRAC 125D does not use pocket trunnions.
2. The lifting blocks in the top lid of the 125-ton HI-TRACs were not modeled. Holtite-A was modeled instead. This is a small, localized item and will not impact the dose rates.
3. The door side plates that are in the middle of the transfer lid of the HI-TRAC 125 are not modeled. This is acceptable because the dose location calculated on the bottom of the transfer lid is in the center.
4. The outside diameter of the Holtite-A portion of the top lid of the 125-ton HI-TRACs was modeled as 4 inches larger than it is due to a design enhancement. This is acceptable because the peak dose rates on the top lid occur on the inner portions of the lid.

HI-STORM Modeling Discrepancies

1. The steel channels in the cavity between the MPC and overpack were not modeled. This is conservative since it removes steel that would provide a small amount of additional shielding.
2. The bolt anchor blocks were not explicitly modeled. Concrete was used instead. These are small, localized items and will not impact the dose rates.
3. In the HI-STORM 100S model, the exit vents were modeled as being inline with the inlet vents. In practice, they are rotated 45 degrees and positioned above the short radial plates. Therefore, this modeling change has the exit vents positioned above the full length radial plates. This modeling change has minimal impact on the dose rates at the exit vents.
4. The short radial plates in the HI-STORM 100S overpack were modeled in MCNP even though they are optional.
5. The pedestal baseplate, which is steel with holes for pouring concrete, in the HI-STORM overpacks was modeled as concrete rather than steel. This is acceptable because this piece of steel is positioned at the bottom of the pedestal below 5 inches of steel and a minimum of 11.5 inches of concrete and therefore will have no impact on the dose rates at the bottom vent.
6. Minor penetrations in the body of the overpack (e.g. holes for grounding straps) are not modeled as these are small localized effects which will not affect the off-site dose rates.
7. In June 2001, the inner shield shell of the HI-STORM 100 overpack was removed and the concrete density in the body of the overpack (not the pedestal of lid) was increased to compensate. Appendix 5.E presents a comparison of the dose rates calculated for a HI-STORM 100 overpack with and without the inner shield shell. The MPC-24 was used in this comparison. The results indicate that there is very little difference in the calculated dose rates when the inner shield shell is removed and the concrete density is increased. Therefore, all HI-STORM 100 analysis presented in the main portion of this chapter includes the inner shield shell.
8. The drawings in Section 1.5 indicate that the HI-STORM 100S has a variable height. This is achieved by adjusting the height of the body of the overpack. The pedestal height is not adjusted. Conservatively, all calculations in this chapter used the shorter height for the HI-STORM 100S.

9. In February 2002, the top plate on the HI-STORM 100 overpack was modified to be two pieces in a shear ring arrangement. The total thickness of the top plate was not changed. However, there is approximately a 0.5 inch gap between the two pieces of the top plate. This gap was not modeled in MCNP since it will result in a small increase in the dose rate on the overpack lid in an area where the dose rate is greatly reduced compared to other locations on the lid.

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 5.3.8 and 5.3.9 graphically depict the location of the PWR and BWR fuel assemblies within the HI-STORM 100 System. The axial locations of the Boral, 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. The gamma shield cross plates located in the inlet and outlet vents, which effectively reduce the gamma dose in these locations, are modeled explicitly.

The MCNP model of the HI-TRAC transfer cask describes the lifting trunnions, pocket trunnions, and the opening in the HI-TRAC top lid. The fins through the HI-TRAC water jacket are also modeled. Streaming considerations through these trunnions and fins are discussed in Section 5.4.1.

The design of the HI-STORM 100 System, as described in the drawings in Chapter 1, has eliminated all other possible streaming paths. Therefore, the MCNP model does not represent any additional streaming paths. A brief justification of this assumption is provided for each penetration.

- The lifting trunnions will remain installed in the HI-TRAC transfer cask.
- The pocket trunnions of the HI-TRAC are modeled as solid blocks of steel. No credit is taken for any part of the pocket trunnion that extends beyond the water jacket.
- The threaded holes in the MPC lid are plugged with solid plugs during storage and, therefore, do not create a void in the MPC lid.
- The drain and vent ports in the MPC lid are designed to eliminate streaming paths. The holes in the vent and drain port cover plates are filled with a set screw and plug weld. The steel lost in the MPC lid at the port location is replaced with a block of steel

approximately 6 inches thick located directly below the port opening and attached to the underside of the lid. This design feature is shown on the drawings in Chapter 1. The MCNP model did not explicitly represent this arrangement but, rather, modeled the MPC lid as a solid plate.

5.3.2 Regional Densities

Composition and densities of the various materials used in the HI-STORM 100 System and HI-TRAC shielding analyses are given in Tables 5.3.2 and 5.3.3. All of the materials and their actual geometries are represented in the MCNP model.

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 is equipped with a water jacket providing radial neutron shielding. Demineralized 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 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.

Since the HI-STORM 100S and the newer configuration of the HI-STORM 100 do not have the inner shield shell present, the minimum density of the concrete in the body (not the lid or pedestal) of the overpack has been increased slightly to compensate for the change in shielding relative to the HI-STORM 100 overpack with the inner shield shell. Table 5.3.2 shows the concrete composition and densities that were used for the HI-STORM 100 and HI-STORM 100S overpacks. Since the density of concrete is increased by altering the aggregate that is used, the composition of the slightly denser concrete was calculated by keeping the same mass of water as the 2.35 gm/cc composition and increasing all other components by the same ratio.

Sections 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 Section 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	7.375	7.375	SS304	SS304
Space	7.375	8.375	1.0	zircaloy	void
Fuel	8.375	152.375	144	fuel & zircaloy	fuel
Gas Plenum Springs	152.375	156.1875	3.8125	SS304 & zircaloy	SS304
Gas Plenum Spacer	156.1875	160.5625	4.375	SS304 & zircaloy	SS304
Upper End Fitting	160.5625	165.625	5.0625	SS304	SS304
BWR					
Lower End Fitting	0.0	7.385	7.385	SS304	SS304
Fuel	7.385	151.385	144	fuel & zircaloy	fuel
Space	151.385	157.385	6	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 lower fuel spacer 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 100 SYSTEM

Component	Density (g/cm ³)	Elements	Mass Fraction (%)
Uranium Oxide	10.412	²³⁵ U	2.9971(BWR) 3.2615(PWR)
		²³⁸ U	85.1529(BWR) 84.8885(PWR)
		O	11.85
Boral [†]	2.644	¹⁰ B	4.4226 (MPC-68 and MPC-32 in HI-STORM & HI-TRAC; MPC-24 in HI-STORM) 4.367 (MPC-24 in HI-TRAC)
		¹¹ B	20.1474 (MPC-68 and MPC-32 in HI-STORM & HI-TRAC; MPC-24 in HI-STORM) 19.893 (MPC-24 in HI-TRAC)
		Al	68.61 (MPC-68 and MPC-32 in HI-STORM & HI-TRAC; MPC-24 in HI-STORM) 69.01 (MPC-24 in HI-TRAC)
		C	6.82 (MPC-68 and MPC-32 in HI-STORM & HI-TRAC; MPC-24 in HI-STORM) 6.73 (MPC-24 in HI-TRAC)
SS304	7.92	Cr	19
		Mn	2
		Fe	69.5
		Ni	9.5
Carbon Steel	7.82	C	0.5
		Fe	99.5
Zircaloy	6.55	Zr	100

[†] All B-10 loadings in the Boral 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 100 SYSTEM

Component	Density (g/cm ³)	Elements	Mass Fraction (%)
Neutron Shield Holtite-A	1.61	C	27.66039
		H	5.92
		Al	21.285
		N	1.98
		O	42.372
		¹⁰ B	0.14087
		¹¹ B	0.64174
BWR Fuel Region Mixture	4.29251	²³⁵ U	2.4966
		²³⁸ U	70.9315
		O	9.8709
		Zr	16.4046
		N	8.35E-05
		Cr	0.0167
		Fe	0.0209
		Sn	0.2505
PWR Fuel Region Mixture	3.869939	²³⁵ U	2.7652
		²³⁸ U	71.9715
		O	10.0469
		Zr	14.9015
		Cr	0.0198
		Fe	0.0365
		Sn	0.2587

Table 5.3.2 (continued)

COMPOSITION OF THE MATERIALS IN THE HI-STORM 100 SYSTEM

Component	Density (g/cm ³)	Elements	Mass Fraction (%)
Lower End Fitting (PWR)	1.0783	SS304	100
Gas Plenum Springs (PWR)	0.1591	SS304	100
Gas Plenum Spacer (PWR)	0.1591	SS304	100
Upper End Fitting (PWR)	1.5410	SS304	100
Lower End Fitting (BWR)	1.4862	SS304	100
Gas Plenum Springs (BWR)	0.2653	SS304	100
Expansion Springs (BWR)	0.6775	SS304	100
Upper End Fitting (BWR)	1.3692	SS304	100
Handle (BWR)	0.2572	SS304	100
Lead	11.3	Pb	99.9
		Cu	0.08
		Ag	0.02
Water	0.9140 (water jacket)	H	11.2
	0.9619 (inside MPC)	O	88.8

Table 5.3.2 (continued)

COMPOSITION OF THE MATERIALS IN THE HI-STORM 100 SYSTEM

Component	Density (g/cm ³)	Elements	Mass Fraction (%)
Concrete Lid and pedestal of the HI-STORM 100 and 100S and the body of the 100 when the inner shield shell is present	2.35	H	0.6
		O	50.0
		Si	31.5
		Al	4.8
		Na	1.7
		Ca	8.3
		Fe	1.2
		K	1.9
Concrete HI-STORM 100S body and HI-STORM 100 body when the inner shield shell is not present	2.48	H	0.569
		O	49.884
		Si	31.594
		Al	4.814
		Na	1.705
		Ca	8.325
		Fe	1.204
		K	1.905

Table 5.3.3

COMPOSITION OF THE FUEL PELLETS IN THE MIXED OXIDE FUEL ASSEMBLIES

Component	Density (g/cm ³)	Elements	Mass Fraction (%)
Mixed Oxide Pellets	10.412	²³⁸ U	85.498
		²³⁵ U	0.612
		²³⁸ Pu	0.421
		²³⁹ Pu	1.455
		²⁴⁰ Pu	0.034
		²⁴¹ Pu	0.123
		²⁴² Pu	0.007
		O	11.85
Uranium Oxide Pellets	10.412	²³⁸ U	86.175
		²³⁵ U	1.975
		O	11.85

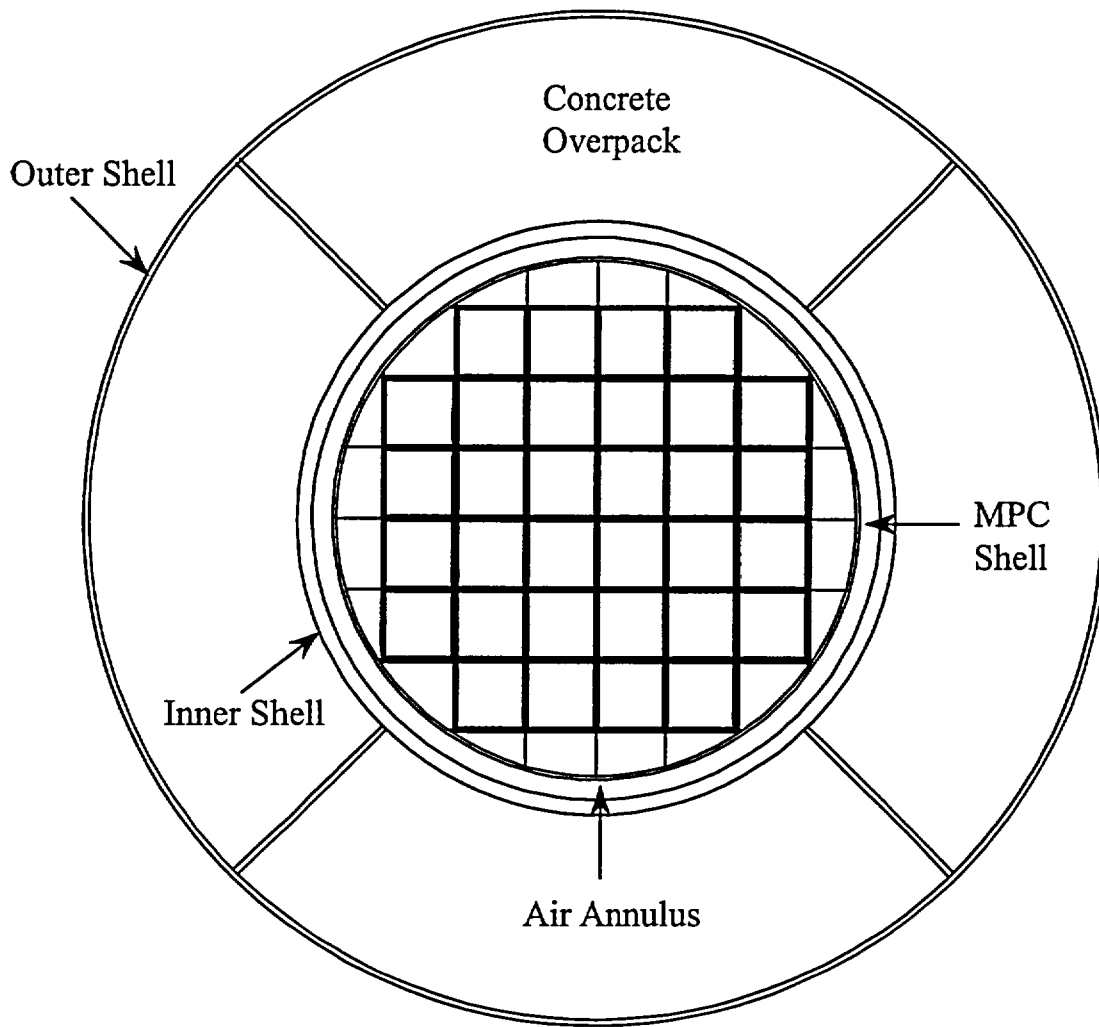


FIGURE 5.3.1; HI-STORM 100 OVERPACK WITH MPC-32 CROSS SECTIONAL VIEW AS MODELLED IN MCNP[†]

[†] This figure is drawn to scale using the MCNP plotter.

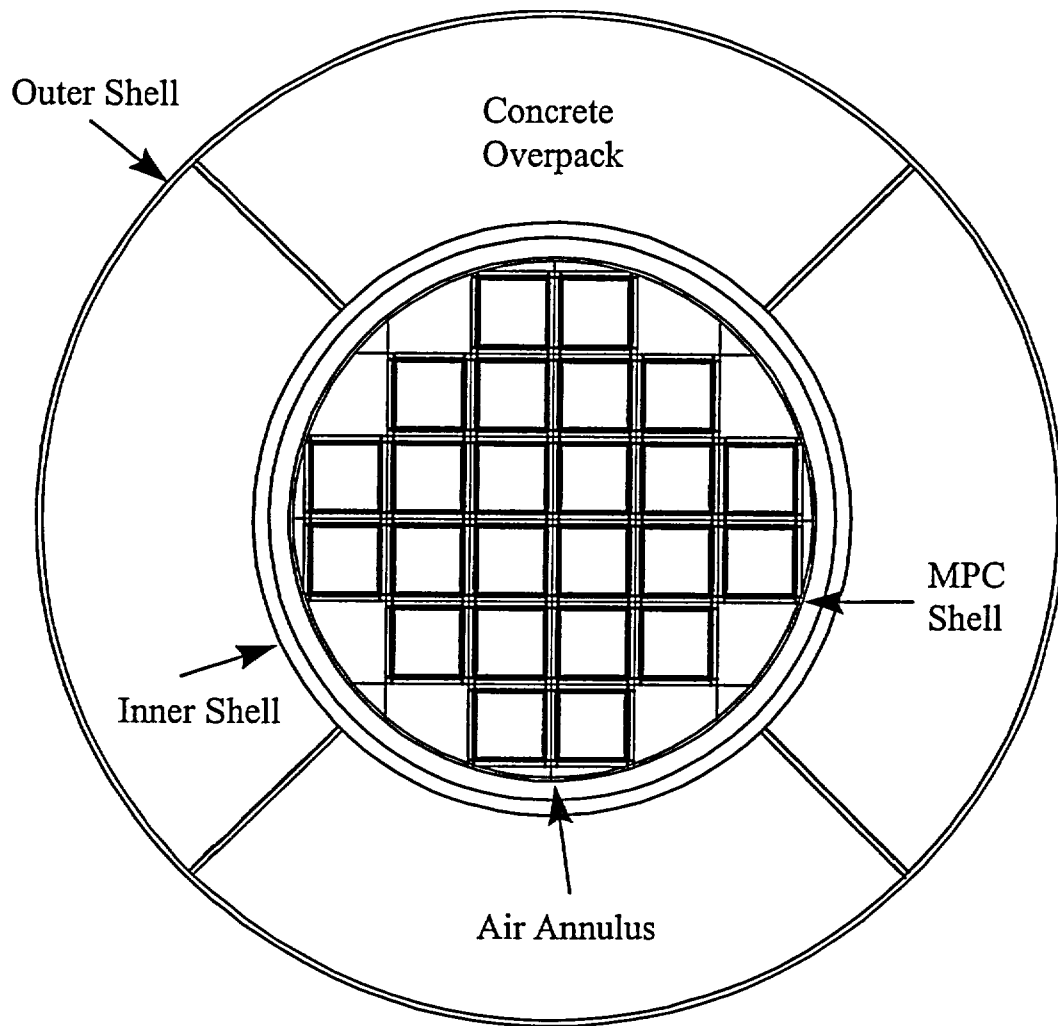


FIGURE 5.3.2; HI-STORM 100 OVERPACK WITH MPC-24 CROSS SECTIONAL VIEW AS MODELLED IN MCNP[†]

[†] This figure is drawn to scale using the MCNP plotter.

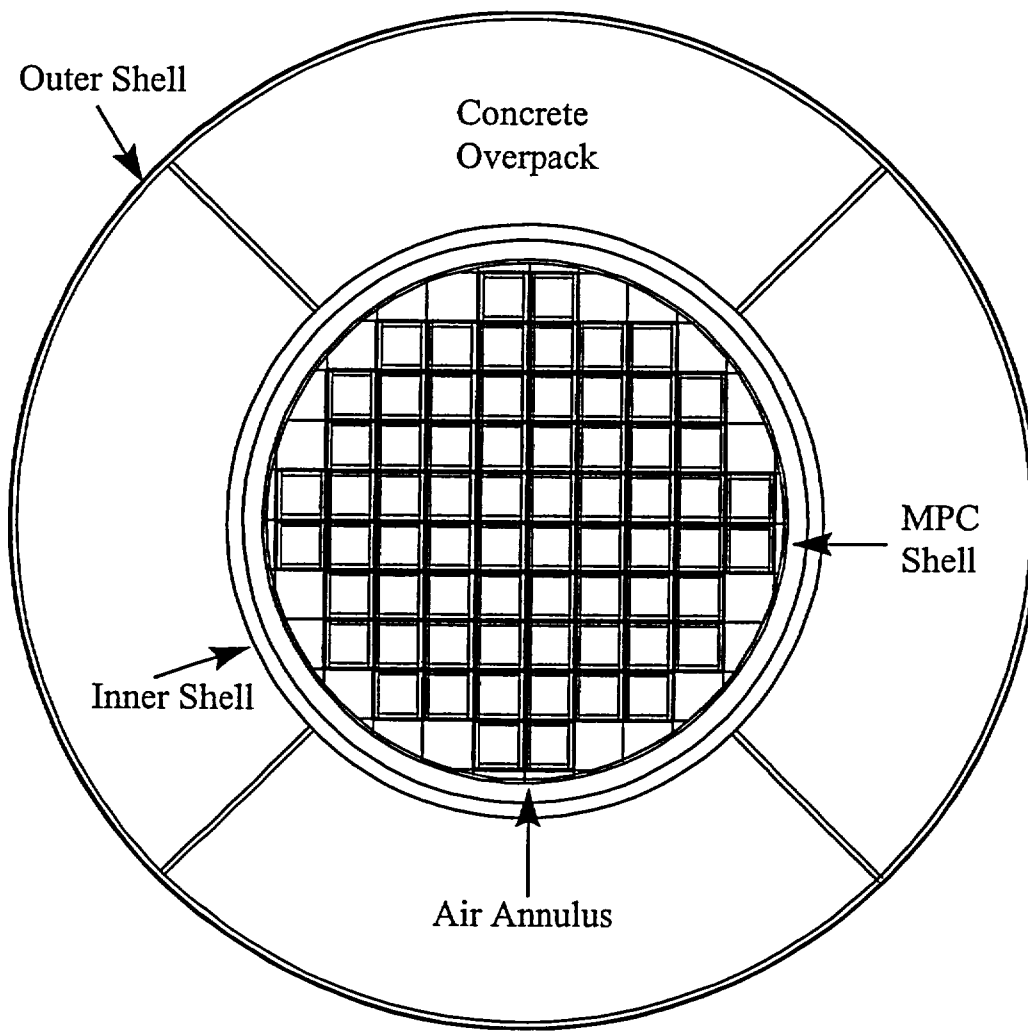


FIGURE 5.3.3; HI-STORM 100 OVERPACK WITH MPC-68 CROSS SECTIONAL VIEW AS MODELLED IN MCNP[†]

[†] This figure is drawn to scale using the MCNP plotter.

**FIGURE WITHHELD AS SENSITIVE
UNCLASSIFIED INFORMATION**

**FIGURE 5.3.4; CROSS SECTIONAL VIEW OF AN MPC-32 BASKET CELL
AS MODELED IN MCNP**

FIGURE WITHHELD AS SENSITIVE
UNCLASSIFIED INFORMATION

FIGURE 5.3.5; CROSS SECTIONAL VIEW OF AN MPC-24 BASKET CELL AS MODELED
IN MCNP

FIGURE WITHHELD AS SENSITIVE
UNCLASSIFIED INFORMATION

FIGURE 5.3.6; CROSS SECTIONAL VIEW OF AN MPC-68 BASKET CELL AS MODELED
IN MCNP

100 TON HI-TRAC

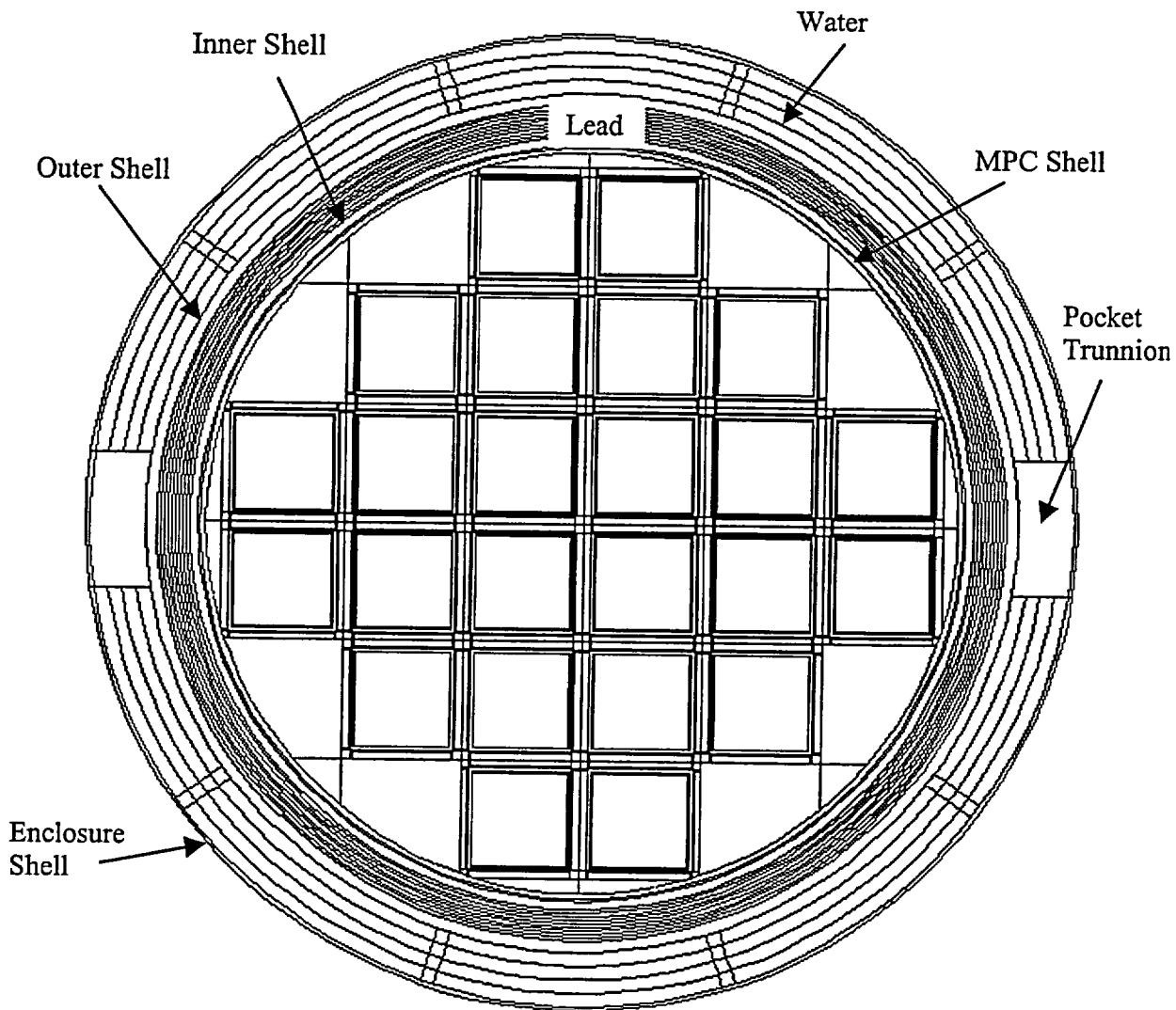


FIGURE 5.3.7; HI-TRAC OVERPACK WITH MPC-24 CROSS SECTIONAL VIEW AS MODELED IN MCNP[†]

[†] This figure is drawn to scale using the MCNP plotter.

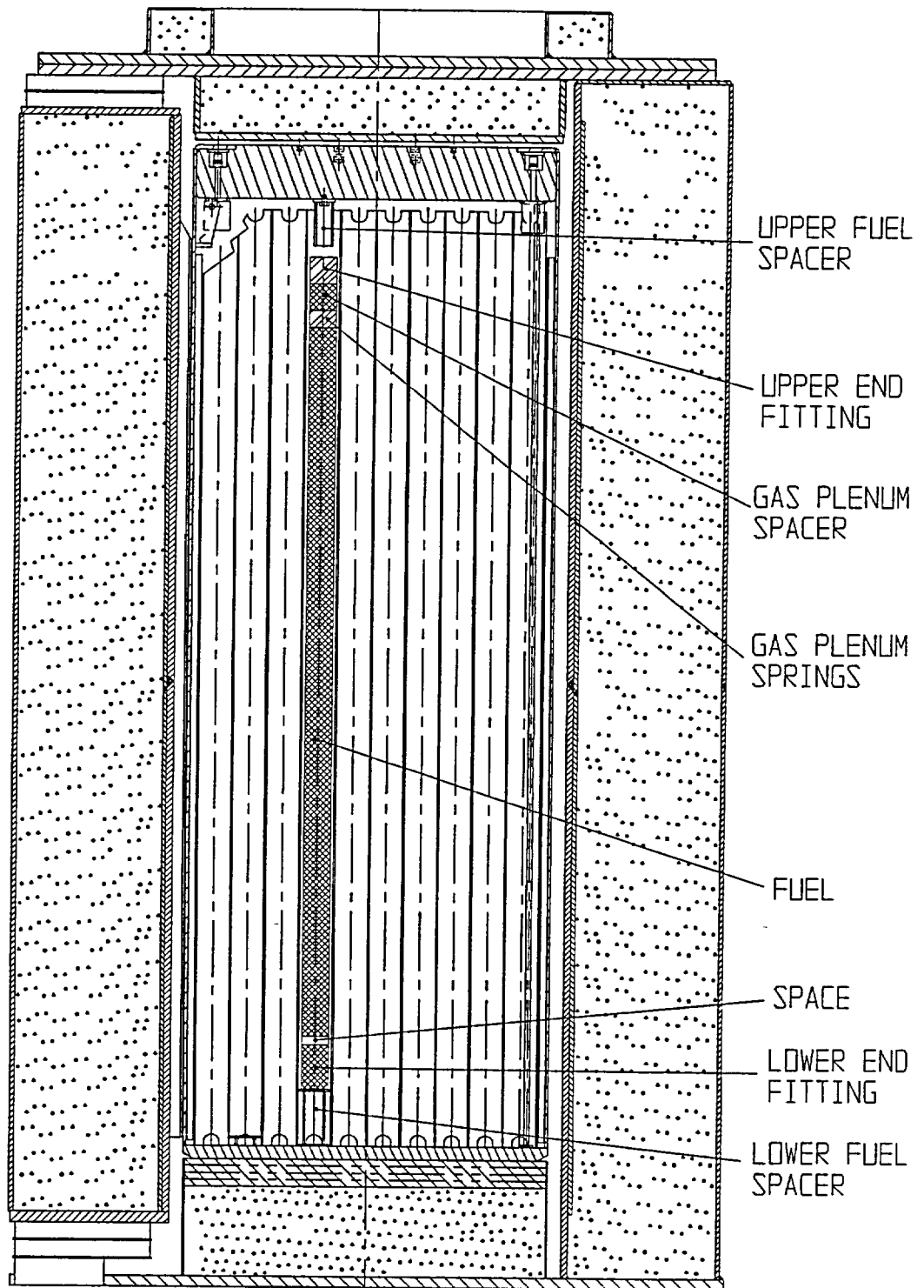


FIGURE 5.3.8; AXIAL LOCATION OF PWR DESIGN BASIS FUEL IN THE HI-STORM OVERPACK

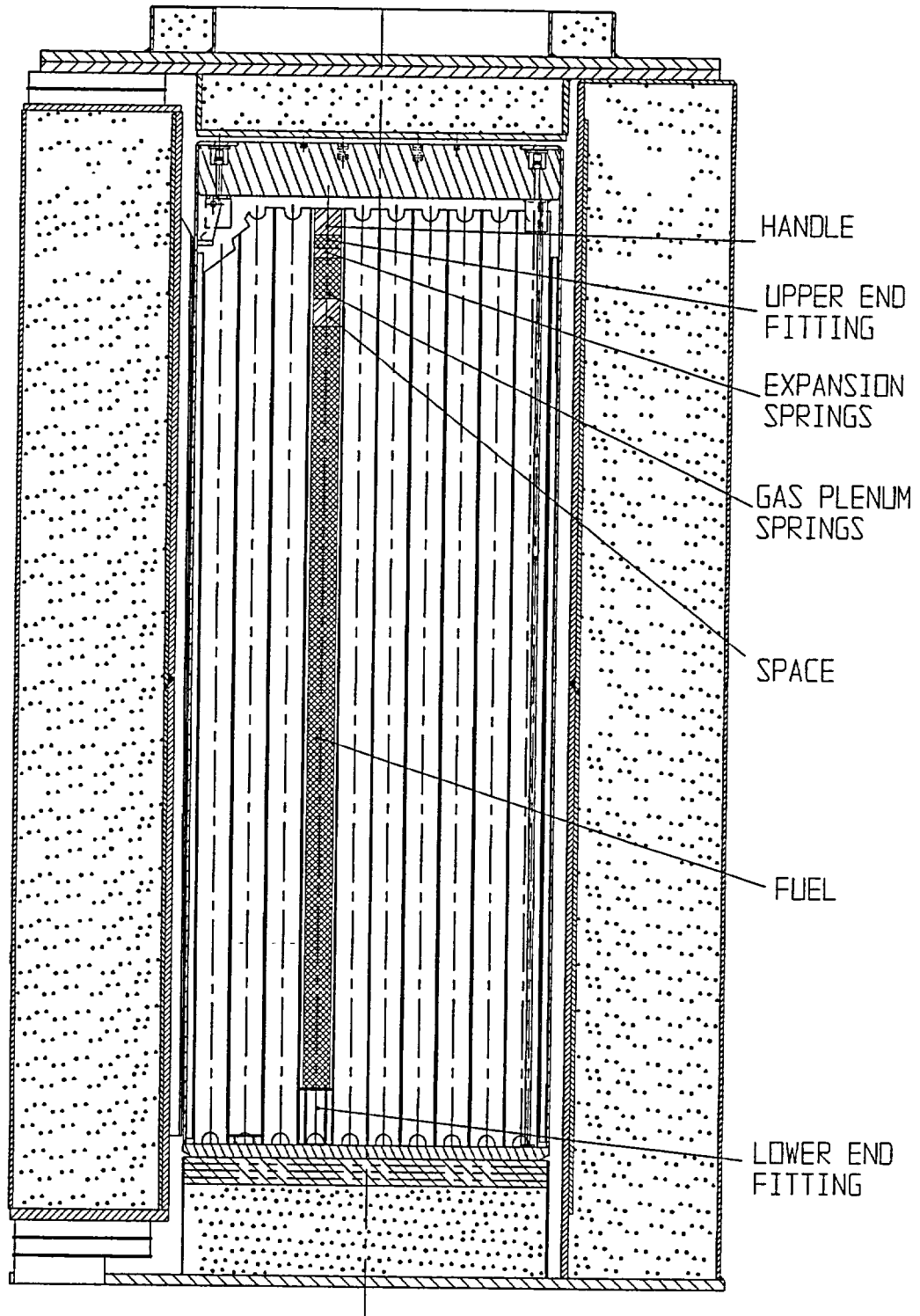


FIGURE 5.3.9; AXIAL LOCATION OF BWR DESIGN BASIS FUEL IN THE HI-STORM OVERPACK

FIGURE WITHHELD AS SENSITIVE
UNCLASSIFIED INFORMATION

FIGURE 5.3.10; CROSS SECTION OF HI-STORM 100 OVERPACK

FIGURE WITHHELD AS SENSITIVE
UNCLASSIFIED INFORMATION

FIGURE 5.3.11; HI-STORM 100 OVERPACK CROSS SECTIONAL ELEVATION VIEW

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FIGURE WITHHELD AS SENSITIVE
UNCLASSIFIED INFORMATION

FIGURE 5.3.12; 100-TON HI-TRAC TRANSFER CASK WITH POOL LID CROSS
SECTIONAL ELEVATION VIEW (AS MODELED)

FIGURE WITHHELD AS SENSITIVE
UNCLASSIFIED INFORMATION

FIGURE 5.3.13; 125-TON HI-TRAC TRANSFER CASK WITH POOL LID CROSS
SECTIONAL ELEVATION VIEW (AS MODELED)

FIGURE WITHHELD AS SENSITIVE
UNCLASSIFIED INFORMATION

FIGURE 5.3.14; HI-TRAC 100 TRANSFER CASK CROSS SECTIONAL VIEW
(AS MODELED)

FIGURE WITHHELD AS SENSITIVE
UNCLASSIFIED INFORMATION

FIGURE 5.3.15; HI-TRAC 125 TRANSFER CASK CROSS SECTIONAL VIEW
(AS MODELED)

FIGURE WITHHELD AS SENSITIVE
UNCLASSIFIED INFORMATION

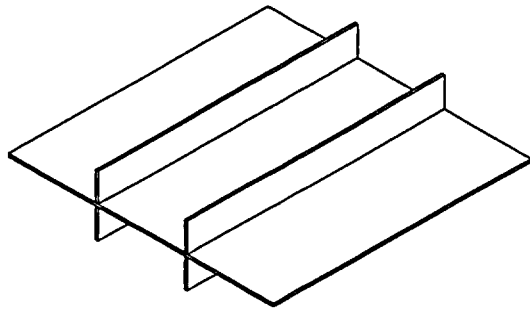
FIGURE 5.3.16; 100-TON HI-TRAC TRANSFER LID (AS MODELED)

FIGURE WITHHELD AS SENSITIVE
UNCLASSIFIED INFORMATION

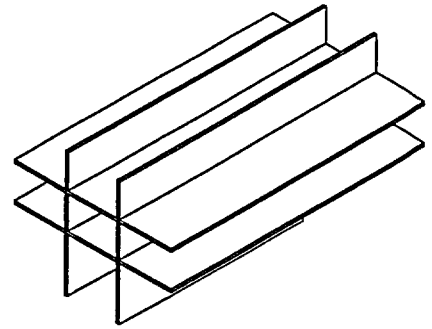
FIGURE 5.3.17; 125-TON HI-TRAC TRANSFER LID (AS MODELED)

**FIGURE WITHHELD AS SENSITIVE
UNCLASSIFIED INFORMATION**

FIGURE 5.3.18; HI-STORM 100S OVERPACK CROSS SECTIONAL ELEVATION VIEW

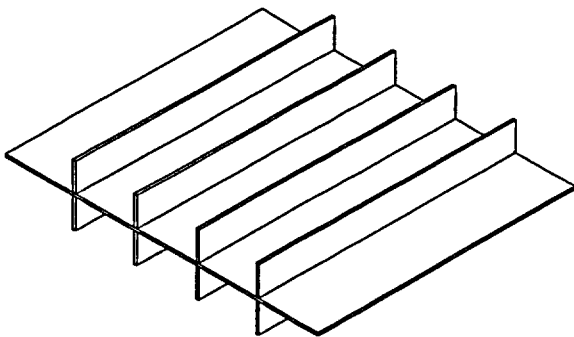


OUTLET VENT

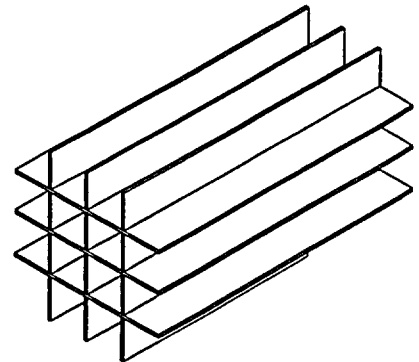


INLET VENT

MANDATORY GAMMA SHIELD CROSS PLATES FOR HI-STORM 100
AND HI-STORM 100S



OUTLET VENT



INLET VENT

OPTIONAL GAMMA SHIELD CROSS PLATES FOR HI-STORM 100S
THAT MAY BE USED INSTEAD OF THE MANDATORY DEVICES.

FIGURE 5.3.19: GAMMA SHIELD CROSS PLATE CONFIGURATION OF
HI-STORM 100 AND HI-STORM 100S

FIGURE WITHHELD AS SENSITIVE
UNCLASSIFIED INFORMATION

FIGURE 5.3.20; HI-TRAC 125D TRANSFER CASK CROSS SECTIONAL VIEW
(AS MODELED)

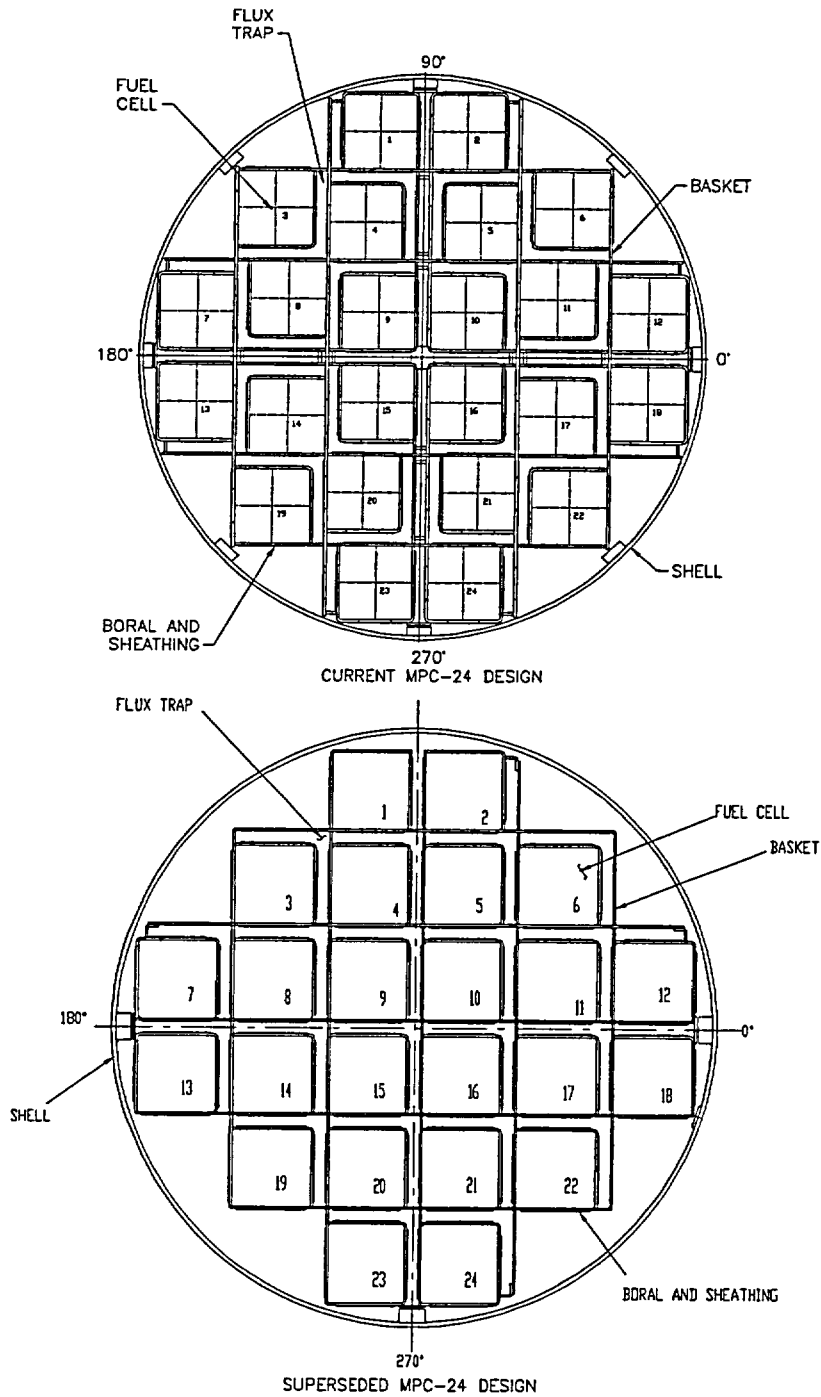


FIGURE 5.3.21; CROSS SECTIONAL VIEWS OF THE CURRENT MPC-24 DESIGN AND THE SUPERSEDED MPC-24 WHICH IS USED IN THE MCNP MODELS.