CHAPTER 5: SHIELDING EVALUATION

5.0 INTRODUCTION

The shielding analysis of the HI-STAR 100 System is presented in this chapter. The HI-STAR 100 System is designed to accommodate different MPCs within one standard HI-STAR 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 *and MPC-68F* (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. 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 or MPC-68F.*

In addition to housing intact PWR and BWR fuel assemblies, the HI-STAR 100 System is designed to transport damaged BWR fuel assemblies and BWR fuel debris. Damaged fuel assemblies and fuel debris are defined in Subsection 1.2.3. Both damaged BWR fuel assemblies and BWR fuel debris are required to be loaded into Damaged Fuel Containers (DFCs)~~ prior to being loaded into the MPC~~. DFCs containing fuel debris must be stored in the MPC-68F. DFCs containing damaged fuel assemblies may be stored in either the MPC-68 or the MPC-68F. Only the fuel assemblies in the Dresden 1 and Humboldt Bay fuel assembly classes identified in Table 1.2.9 are authorized as contents for transport in the HI-STAR 100 system as either *BWR* damaged fuel or fuel debris.

The MPC-68 and MPC-68F are also capable of transporting 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.

Slightly modified version of the MPC-24E and MPC-24EF are being used for the transportation of Trojan nuclear power plant spent nuclear fuel, non-fuel hardware, and damaged fuel and fuel debris as described in Subsection 1.2.3. These MPCs are referred to as the Trojan MPC-24E and Trojan MPC-24EF. The Trojan MPC-24E/EF was explicitly analyzed in this chapter for the inclusion of the Trojan non-fuel hardware, damaged fuel, and Antimony-Beryllium and Californium neutron sources.

This chapter contains the following information:

- A description of the shielding features of the HI-STAR 100 System.
- 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-STAR 100 System.

- Analyses for ~~each of~~ the HI-STAR 100 Systems content conditions to show that the 10CFR71.47 radiation limits are met during normal conditions of transport and that the 10CFR71.51 dose rate limit is not exceeded following hypothetical accident conditions.
- Analyses which demonstrate that the storage of damaged fuel in the HI-STAR 100 System is bounded by the BWR intact fuel analysis during normal and hypothetical accident conditions.
- *Analyses for the Trojan Nuclear Power Plant spent fuel contents, including damaged fuel and fuel debris, and non-fuel hardware.*

5.1 DISCUSSION AND RESULTS

The principal sources of radiation in the HI-STAR 100 System are:

- Gamma radiation originating from the following sources

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

- 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 *and Trojan* ~~antimony-beryllium~~ neutron sources

Shielding from gamma radiation is provided by the steel structure of the MPC and overpack. In order for the neutron shielding to be effective, the neutrons must be thermalized and then absorbed in a material of high neutron cross section. In the HI-STAR 100 System design, a neutron shielding material, Holtite-A, is used to thermalize the neutrons. Boron carbide, dispersed in the neutron shield, 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] from Los Alamos National Laboratory. 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] from Oak Ridge National Laboratory. *The source terms for the Trojan specific inventory were calculated with the SAS2H and ORIGEN-S sequences from the SCALE 4.4 system [5.1.4, 5.1.5] as described in the Trojan FSAR [5.1.6].* 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 intact zircaloy clad fuels used in calculating the dose rates presented in this chapter are the B&W 15x15 (with zircaloy and non-zircaloy incore spacers) and the GE 7x7, for PWR and BWR fuel types, respectively. The design basis intact 6x6, damaged, and mixed oxide (MOX) fuel assemblies are the GE 6x6. Table 1.2.13 specifies the acceptable intact zircaloy clad fuel characteristics for transport. Table 1.2.14 specifies the acceptable damaged and MOX zircaloy clad fuel characteristics for transport.

The design bases intact stainless steel clad fuels are the WE 15x15 and the AC 10x10, for PWR and BWR fuel types, respectively. Table 1.2.19 specifies the acceptable fuel characteristics of stainless steel clad fuel for transport.

The Trojan spent fuel contents were analyzed separately, as discussed in later sections, and therefore are not covered by the design basis fuel assemblies mentioned above.

~~Table 1.2.20~~*Appendix A to the Certificate of Compliance (CoC)* specifies, in tabular form, the minimum enrichment, burnup and cooling time combinations for spent nuclear fuel that were analyzed for transport in the MPC-24, *MPC-32,* and MPC-68. Each combination provides a dose rate equal to or below the maximum values reported in this section. ~~This~~ *The* table*s in the CoC* represents the fuel assembly acceptance criteria.

The burnup, cooling time, and minimum enrichment combinations specified in Appendix A to the Certificate of Compliance (CoC) were calculated by specifically analyzing each combination and verifying that the calculated dose rates were less than the regulatory limits. Results are not presented in this chapter for each burnup, cooling time, and minimum enrichment combination analyzed. Rather, the results for the combination that produced the highest dose rate for a specific regulatory acceptance criteria in a specific MPC are presented. The results for the 2 meter location during normal conditions are presented in this section as well as the results for the hypothetical accident condition. Results for the other dose locations can be found in Section 5.4. Different burnup, cooling time, and minimum enrichment combinations may be presented for the different locations based on the calculated results.
~~Table 1.2.20 was developed in a two stage process. First, the burnup and cooling time combinations that produced assembly decay heat rates equal to the thermal limits specified in Figure 1.2.12 were calculated. Second, the dose rates at the various locations were calculated for these burnup and cooling time combinations and compared to the regulatory limits. In some cases, the burnup, for a specified cooling time, had to be reduced to meet the dose rate limits. Therefore, Table 1.2.20 is based on both the maximum permissible decay heat per assembly and the regulatory dose rate limits. The burnup and cooling time combinations analyzed in this chapter are equivalent to or bound the acceptable burnup and cooling time combinations in Table 1.2.20. The dose rates from the burnup and cooling time combination which provided the highest dose rates at the midplane of the cask for each location (surface and 2 meter – normal condition, and 1 meter – hypothetical accident condition) are reported in this section. As a result, the burnup and cooling time combinations reported in this section may be different between locations. Dose rates for each combination calculated are listed in Section 5.4.~~

Unless otherwise stated, all dose rates reported in this chapter are average surface dose rates. The effect of radiation peaking due to azimuthal variations in the fuel loading pattern and the steel radial channels is specifically addressed in Subsection 5.4.1.

5.1.1 Normal Operations

The 10CFR71.47 external radiation requirements during normal transport operations for an exclusive use shipment are:

1. 200 mrem/hr (2 mSv/hr) on the external surface of the package, *unless the following conditions are met, in which case the limit is 1000 mrem/hr (10 mSv/hr).*
   i. *The shipment is made in a closed transport vehicle;*
   ii. *The package is secured within the vehicle so that its position remains fixed during transportation; and*
   iii. *There are no loading and unloading operations between the beginning and end of the transportation.*

2. 200 mrem/hr (2 mSv/hr) at any point on the outer surface of the vehicle, including the top and underside of the vehicle; or in the case of a flat-bed style vehicle, at any point on the vertical planes projected from the outer edges of the vehicle, on the upper surface of the load or enclosure, if used, and on the lower external surface of the vehicle.

3. 10 mrem/hr (0.1 mSv/hr) at any point 2 meters (80 in) from the outer lateral surfaces of the vehicle (excluding the top and underside of the vehicle); or in the case of a flat-bed style vehicle, at any point 2 meters (6.6 feet) from the vertical planes projected by the outer edges of the vehicle (excluding the top and underside of the vehicle).

4. 2 mrem/h (0.02 mSv/hr) in any normally occupied space, except that this provision does not apply to private carriers, if exposed personnel under their control wear radiation dosimetry devices in conformance with 10CFR20.1502.

The Standard Review Plan for Transportation Packages of Spent Nuclear Fuel, NUREG-1617 [5.2.1] states that "Personnel barriers and similar devices that are attached to the conveyance, rather than the package, can, however, qualify the vehicle as a closed vehicle (NUREG/CR-5569A and NUREG/CR-5569B) as defined in 49 CFR 173.403."

When the HI-STAR is transported, a personnel barrier will be placed over the HI-STAR as depicted in Figure 1.2.8. This personnel barrier spans the distance between the impact limiters. The outer radial location of the personnel barrier is equal to the outer radial surface of the impact limiters and the personnel barrier is attached to the saddle on the rail car rather than the HI-STAR overpack. Therefore, the personnel barrier acts as an enclosure for the main body of the HI-STAR overpack. As a result, the 1000 mrem/hr limit for the enclosed package is applicable for the outer radial surface of the overpack in the region between the impact limiters. Since the impact limiters are not enclosed, the surface of the impact limiters is required to meet the lower 200 mrem/hr limit for the package.~~The external surface of the HI-STAR 100 System during normal transportation is defined as the outer surface of the impact limiters and the outer radial surface of the overpack in the region between the impact limiters.~~

The HI-STAR 100 System will be transported on either a flat-bed rail car, heavy haul vehicle, or a barge. The smallest width of a transport vehicle is equivalent to the width of the impact limiters. Therefore, the vertical planes projected by the outer side edges of the transport vehicle are equivalent to the outer edge of the impact limiters. The minimum length of any transport vehicle will be 12 feet longer than the length of the overpack, with impact limiters attached. The HI-STAR 100 System will be conservatively positioned a minimum of 6 feet from either end of the transport vehicle. Therefore, the vertical planes projected from the outer edge of the ends of the vehicle will be taken as the end of the top impact limiter and 6 feet from the end of the bottom impact limiter.

Figure 5.1.1 shows the HI-STAR 100 System during normal transport conditions. The impact limiters *and personnel barrier* are outlined on the figure and various dose point locations are shown on the surface of the *enclosure (personnel barrier) and the* HI-STAR 100 System. The dose values reported at the locations shown on Figure 5.1.1 are averaged over a region that is approximately 1 foot in width. Each of the dose locations ~~on the surface of the HI-STAR 100 System~~*in Figure 5.1.1 (with the exception of 2a and 3a)* has a corresponding location at 2 meters from the surface of the transport vehicle as defined above.

Tables 5.1.1 through 5.1.~~3~~*5* provide the maximum dose rates ~~on the surface of the system~~*at two meters from the transport vehicle* during normal transport conditions for the MPC-24, *MPC-32,* and MPC-68 with design basis intact zircaloy clad fuel. ~~Tables 5.1.4 through 5.1.6 list the maximum dose rates two meters from the edge of the transport vehicle during normal conditions.~~ Section 5.4 provides a detailed list of dose rates at several cask locations for ~~all~~ *other* burnup and cooling times ~~analyzed~~*combinations.*

Subsections 5.2.1 and 5.2.2 list the gamma and neutron sources for the design basis zircaloy clad intact, zircaloy clad damaged and MOX fuel assemblies. Since the source strengths of the damaged and MOX fuel are significantly smaller in all energy groups than the intact design basis fuel source strengths, the damaged and MOX fuel dose rates for normal conditions are bounded by the MPC-68 analysis with design basis intact fuel. Therefore, no explicit analysis of the MPC-68 with either damaged or MOX fuel for normal conditions is required to demonstrate that the MPC-68 with damaged fuel or MOX fuel will meet the normal condition regulatory requirements.

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

Subsection 5.4.5 demonstrates that the Dresden Unit 1 fuel assemblies containing antimony-beryllium neutron sources are bounded by the shielding analysis presented in this section.

Subsections 5.4.7 and 5.4.8 present the results for the Trojan contents in the MPC-24E/EF and demonstrate that these contents are acceptable for transportation.

Subsection 5.2.3 lists the gamma and neutron sources for the design basis intact stainless steel clad fuels. The dose rates from these fuels are provided in Subsection 5.4.4.

Tables 5.1.~~4~~*1* through 5.1.~~6~~*5* show that the dose rate at Dose Location #5 (the top of the HI-STAR 100 System, see Figure 5.1.1) at 2 meters from the edge of the transport vehicle is less than 2 mrem/hr. It is, therefore, recommended that the HI-STAR 100 System be positioned such that the top impact limiter is facing the normally occupied space. If this is the orientation, radiation dosimetry will not be required as long as the normally occupied space is a minimum of 2 meters from the impact limiter on the top of the HI-STAR 100 System. If a different orientation is chosen for the HI-STAR 100 System, the dose rate in the normally occupied space will have to be evaluated against the dose requirement for the normally occupied space to determine if radiation dosimetry is required.

The analyses summarized in this section demonstrate the HI-STAR 100 System's compliance with the 10CFR71.47 limits.

5.1.2 Hypothetical Accident Conditions

The 10CFR71.51 external radiation dose limit for design basis accidents is:

- The external radiation dose rate shall not exceed 1 rem/hr (10 mSv/hr) at 1 m (40 in.) from the external surface of the package.

The hypothetical accident conditions of transport have two bounding consequences which affect the shielding materials. They are the damage to the neutron shield as a result of the design basis fire and damage of the impact limiters as a result of the 30 foot drop. In a conservative fashion, the dose analysis assumes that as a result of the fire, the neutron shield is completely destroyed and replaced by a void. Additionally, the impact limiters are assumed to have been lost. These are highly conservative assumptions since some portion of the neutron shield would be expected to remain after the fire as the neutron shield material is fire retardant, and the impact limiters have been shown by 1/4-scale testing to remain attached following impact~~(see Appendix 2.H)~~.

Throughout the hypothetical accident condition the axial location of the fuel will remain fixed within the MPC because of the fuel spacers or by the MPC lid and baseplate if spacers are not used. Chapter 2 provides an analysis to show that the fuel spacers do not fail under all normal and hypothetical accident conditions. Chapter 2 also shows that the inner shell, intermediate shell, radial channels, and outer enclosure shell of the overpack remain unaltered throughout the hypothetical accident conditions. Localized damage of the overpack outer enclosure shell could be experienced during the pin puncture. However, the localized deformations will have only a negligible impact on the dose rate at 1 meter from the surface.

Figure 5.1.2 shows the HI-STAR 100 System after the postulated accident. The various dose point locations at 1 meter from the HI-STAR 100 System are shown on the figure. Tables ~~5.1.8~~*5.1.6 through 5.1.10*~~and 5.1.9~~ provide the maximum dose rates at 1 meter for the accident conditions.

The consequences of the hypothetical accident conditions for the MPC-68F storing either damaged or MOX (which can also be considered damaged) fuel differ slightly from those with intact fuel. For this accident condition, it is conservatively assumed that during a drop accident the damaged fuel collapses and the pellets rest in the bottom of the damaged fuel container. The analysis presented in Subsections 5.4.2 and 5.4.3 demonstrate that the damaged fuel in the post-accident condition has lower source terms (both gamma and neutron) per inch than the intact BWR design basis fuel. Therefore, the damaged fuel post-accident dose rates are bounded by the BWR intact fuel post-accident dose rates.

Subsections 5.4.7 and 5.4.8 present the results for the Trojan contents in the MPC-24E/EF and demonstrate that these contents are acceptable for transportation.

Analyses summarized in this section demonstrate the HI-STAR 100 System's compliance with the 10CFR71.51 radiation dose limit.

Table 5.1.1

DOSE RATES ~~ON THE SURFACE OF~~*AT TWO METERS FROM* THE HI-STAR 100 SYSTEM FOR NORMAL CONDITIONS
MPC-24 WITH DESIGN BASIS ZIRCALOY CLAD FUEL WITH ZIRCALOY INCORE SPACERS
AT WORST CASE BURNUP AND COOLING TIME
~~44~~*34*,500 MWD/MTU AND ~~15~~*9*-YEAR COOLING

Dose Point† Location	Fuel Gammas†† (mrem/hr)	Gammas from Incore Spacers (mrem/hr)	^{60}Co Gammas (mrem/hr)	Neutrons (mrem/hr)	Totals (mrem/hr)
1	*2.64*	*0.00*	*3.38*	*2.12*	*8.14*
2	*5.81*	*0.00*	*1.59*	*2.32*	*9.72*
3	*2.24*	*0.00*	*3.40*	*2.05*	*7.68*
4	*1.83*	*0.00*	*3.45*	*2.19*	*7.47*
5	*0.02*	*0.00*	*0.01*	*0.26*	*0.29*
6	*0.35*	*0.00*	*8.18*	*0.75*	*9.28*
10CFR71.47 Limit					*10.00*

† Refer to Figure 5.1.1.

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

Table 5.1.2

DOSE RATES ~~ON THE SURFACE OF~~*AT TWO METERS FROM* THE HI-STAR 100 SYSTEM FOR NORMAL CONDITIONS
MPC-24 WITH DESIGN BASIS ZIRCALOY CLAD FUEL WITH NON-ZIRCALOY INCORE SPACERS
AT WORST CASE BURNUP AND COOLING TIME
~~37~~*24*,500 MWD/MTU AND ~~15~~*10*-YEAR COOLING

Dose Point† Location	**Fuel Gammas†† (mrem/hr)**	**Gammas from Incore Spacers (mrem/hr)**	**^{60}Co Gammas (mrem/hr)**	**Neutrons (mrem/hr)**	**Totals (mrem/hr)**
1	*1.35*	*1.89*	*2.45*	*0.74*	*6.42*
2	*3.24*	*4.58*	*0.79*	*0.79*	*9.40*
3	*1.14*	*1.57*	*2.46*	*0.72*	*5.89*
4	*0.93*	*1.28*	*2.49*	*0.77*	*5.46*
5	*0.01*†††	-	*0.00*	*0.09*	*0.10*
6	*0.38*†††	-	*5.91*	*0.26*	*6.55*
10CFR71.47 Limit					*10.00*

† Refer to Figure 5.1.1.

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

††† Gammas from incore spacers are included with fuel gammas.

Table 5.1.3

DOSE RATES ~~ON THE SURFACE OF~~*AT TWO METERS FROM* THE HI-STAR 100 SYSTEM FOR NORMAL CONDITIONS
MPC-68 WITH DESIGN BASIS ZIRCALOY CLAD FUEL AT
WORST CASE BURNUP AND COOLING TIME
24,500 MWD/MTU AND 8-YEAR COOLING

Dose Point† Location	**Fuel Gammas†† (mrem/hr)**	**Gammas from Incore Spacers (mrem/hr)**	**^{60}Co Gammas (mrem/hr)**	**Neutrons (mrem/hr)**	**Totals (mrem/hr)**
1	*2.21*	*0.53*	*2.88*	*0.95*	*6.58*
2	*4.99*	*1.19*	*0.78*	*1.06*	*8.03*
3	*1.44*	*0.35*	*3.89*	*0.64*	*6.31*
4	*1.10*	*0.26*	*4.12*	*0.61*	*6.09*
5	*0.01*†††	-	*0.01*	*0.07*	*0.08*
6	*0.22*†††	-	*7.85*	*0.31*	*8.38*
10CFR71.47 Limit					*10.00*

† Refer to Figure 5.1.1.

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

††† Gammas from incore spacers are included with fuel gammas.

Table 5.1.4

DOSE RATES AT TWO METERS *FROM THE HI-STAR 100 SYSTEM* FOR NORMAL CONDITIONS
MPC-~~24~~ *32* WITH DESIGN BASIS ZIRCALOY CLAD FUEL WITH ZIRCALOY INCORE SPACERS
AT WORST CASE BURNUP AND COOLING TIME
~~44~~*44*,500 MWD/MTU AND ~~7~~*18*-YEAR COOLING

Dose Point† Location	**Fuel Gammas†† (mrem/hr)**	**Gammas from Incore Spacers (mrem/hr)**	**^{60}Co Gammas (mrem/hr)**	**Neutrons (mrem/hr)**	**Totals (mrem/hr)**
1	*2.08*	*0.00*	*1.63*	*4.47*	*8.18*
2	*4.04*	*0.00*	*0.82*	*4.80*	*9.66*
3	*1.77*	*0.00*	*1.70*	*5.63*	*9.10*
4	*1.45*	*0.00*	*1.71*	*5.61*	*8.76*
5	*0.06*	*0.00*	*0.00*	*0.79*	*0.85*
6	*0.29*	*0.00*	*4.29*	*1.88*	*6.46*
10CFR71.47 Limit					10.00

† Refer to Figure 5.1.1.

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

Table 5.1.5

DOSE RATES AT TWO METERS *FROM THE HI-STAR 100 SYSTEM* FOR NORMAL CONDITIONS
MPC-~~24~~*32* WITH DESIGN BASIS ZIRCALOY CLAD FUEL WITH NON-ZIRCALOY INCORE SPACERS
AT WORST CASE BURNUP AND COOLING TIME
~~24~~*42*,500 MWD/MTU AND ~~10~~*20*-YEAR COOLING

Dose Point† Location	Fuel Gammas†† (mrem/hr)	Gammas from Incore Spacers (mrem/hr)	^{60}Co Gammas (mrem/hr)	Neutrons (mrem/hr)	Totals (mrem/hr)
1	*1.67*	*0.80*	*1.20*	*3.46*	*7.13*
2	*3.76*	*1.95*	*0.38*	*3.39*	*9.48*
3	*1.42*	*0.67*	*1.25*	*4.36*	*7.70*
4	*1.16*	*0.54*	*1.26*	*4.34*	*7.30*
5	*0.05*†††	-	*0.00*	*0.61*	*0.66*
6	*0.32*†††	-	*3.16*	*1.45*	*4.93*
10CFR71.47 Limit					10.00

† Refer to Figure 5.1.1.

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

††† Gammas from incore spacers are included with fuel gammas.

Table 5.1.6

DOSE RATES AT ~~TWO~~ *ONE* METERS FOR ~~NORMAL~~ *ACCIDENT* CONDITIONS
MPC-~~68~~*24* WITH DESIGN BASIS ZIRCALOY CLAD FUEL *WITH ZIRCALOY INCORE SPACERS*
AT WORST CASE BURNUP AND COOLING TIME
~~29~~*34*,500 MWD/MTU AND 9-YEAR COOLING

Dose Point† **Location**	**Fuel Gammas**†† **(mrem/hr)**	**^{60}Co Gammas (mrem/hr)**	**Neutrons (mrem/hr)**	**Totals (mrem/hr)**
1	*6.57*	*33.70*	*88.35*	*128.62*
2	*35.24*	*1.13*	*281.33*	*317.70*
3	*4.71*	*20.60*	*63.14*	*88.45*
4	*2.60*	*15.82*	*46.27*	*64.69*
5	*0.05*	*0.21*	*10.50*	*10.76*
6	*17.13*	*540.08*	*73.21*	*630.41*
10CFR71.51 Limit				1000.00

† Refer to Figure 5.1.2.

†† Gammas generated by neutron capture and gammas from incore spacers are included with fuel gammas.

Table 5.1.7

DOSE RATES AT ONE METER FOR ACCIDENT CONDITIONS
MPC-24 WITH DESIGN BASIS ZIRCALOY CLAD FUEL WITH *NON*-ZIRCALOY INCORE SPACERS
AT WORST CASE BURNUP AND COOLING TIME
~~29~~*24*,500 MWD/MTU AND ~~8~~*10*-YEAR COOLING

Dose Point† Location	**Fuel Gammas†† (mrem/hr)**	**^{60}Co Gammas (mrem/hr)**	**Neutrons (mrem/hr)**	**Totals (mrem/hr)**
1	*8.39*	*24.35*	*30.88*	*63.61*
2	*45.37*	*0.82*	*98.30*	*144.48*
3	*6.15*	*14.89*	*22.07*	*43.10*
4	*3.33*	*11.43*	*16.17*	*30.93*
5	*0.03*	*0.15*	*3.67*	*3.85*
6	*22.48*	*390.32*	*25.60*	*438.40*
10CFR71.51 Limit				1000.00

† Refer to Figure 5.1.2.

†† Gammas generated by neutron capture and gammas from incore spacers are included with fuel gammas.

Table 5.1.8

DOSE RATES AT ONE METER FOR ACCIDENT CONDITIONS
MPC-~~24~~ *68* WITH DESIGN BASIS ZIRCALOY CLAD FUEL ~~WITH NON-ZIRCALOY INCORE SPACERS~~
AT WORST CASE BURNUP AND COOLING TIME
~~24~~*44*,500 MWD/MTU AND ~~10~~*19*-YEAR COOLING

Dose Point† Location	Fuel Gammas†† (mrem/hr)	^{60}Co Gammas (mrem/hr)	Neutrons (mrem/hr)	Totals (mrem/hr)
1	*4.47*	*10.20*	*183.86*	*198.52*
2	*21.94*	*0.22*	*600.71*	*622.86*
3	*1.81*	*6.93*	*93.91*	*102.65*
4	*1.02*	*6.18*	*67.17*	*74.36*
5	*0.04*	*0.07*	*10.92*	*11.03*
6	*5.77*	*171.13*	*126.59*	*303.49*
10CFR71.51 Limit				1000.00

† Refer to Figure 5.1.2.

†† Gammas generated by neutron capture and gammas from incore spacers are included with fuel gammas.

Table 5.1.9

DOSE RATES AT ONE METER FOR ACCIDENT CONDITIONS
MPC-~~68~~*32* WITH DESIGN BASIS ZIRCALOY CLAD FUEL *WITH ZIRCALOY INCORE SPACERS*
AT WORST CASE BURNUP AND COOLING TIME
~~24~~*34*,500 MWD/MTU AND ~~8~~*12*-YEAR COOLING

Dose Point† Location	Fuel Gammas†† (mrem/hr)	^{60}Co Gammas (mrem/hr)	Neutrons (mrem/hr)	Totals (mrem/hr)
1	*4.09*	*23.48*	*72.56*	*100.13*
2	*20.43*	*0.66*	*211.40*	*232.48*
3	*2.78*	*14.02*	*63.52*	*80.31*
4	*1.66*	*11.30*	*46.29*	*59.25*
5	*0.08*	*0.20*	*20.06*	*20.33*
6	*15.36*	*525.74*	*107.25*	*648.35*
10CFR71.51 Limit				1000.00

† Refer to Figure 5.1.2.

†† Gammas generated by neutron capture and gammas from incore spacers are included with fuel gammas.

Table 5.1.10

DOSE RATES AT ONE METER FOR ACCIDENT CONDITIONS
MPC-32 WITH DESIGN BASIS ZIRCALOY CLAD FUEL WITH NON-ZIRCALOY INCORE SPACERS
AT WORST CASE BURNUP AND COOLING TIME
24,500 MWD/MTU AND 12-YEAR COOLING

Dose Point[†] Location	Fuel Gammas[††] (mrem/hr)	^{60}Co Gammas (mrem/hr)	Neutrons (mrem/hr)	Totals (mrem/hr)
1	6.04	19.37	26.39	51.81
2	30.14	0.55	76.88	107.57
3	4.10	11.57	23.10	38.77
4	2.45	9.33	16.84	28.61
5	0.04	0.16	7.30	7.50
6	22.92	433.86	39.00	495.78
10CFR71.51 Limit				1000.00

† Refer to Figure 5.1.2.

†† Gammas generated by neutron capture and gammas from incore spacers are included with fuel gammas.

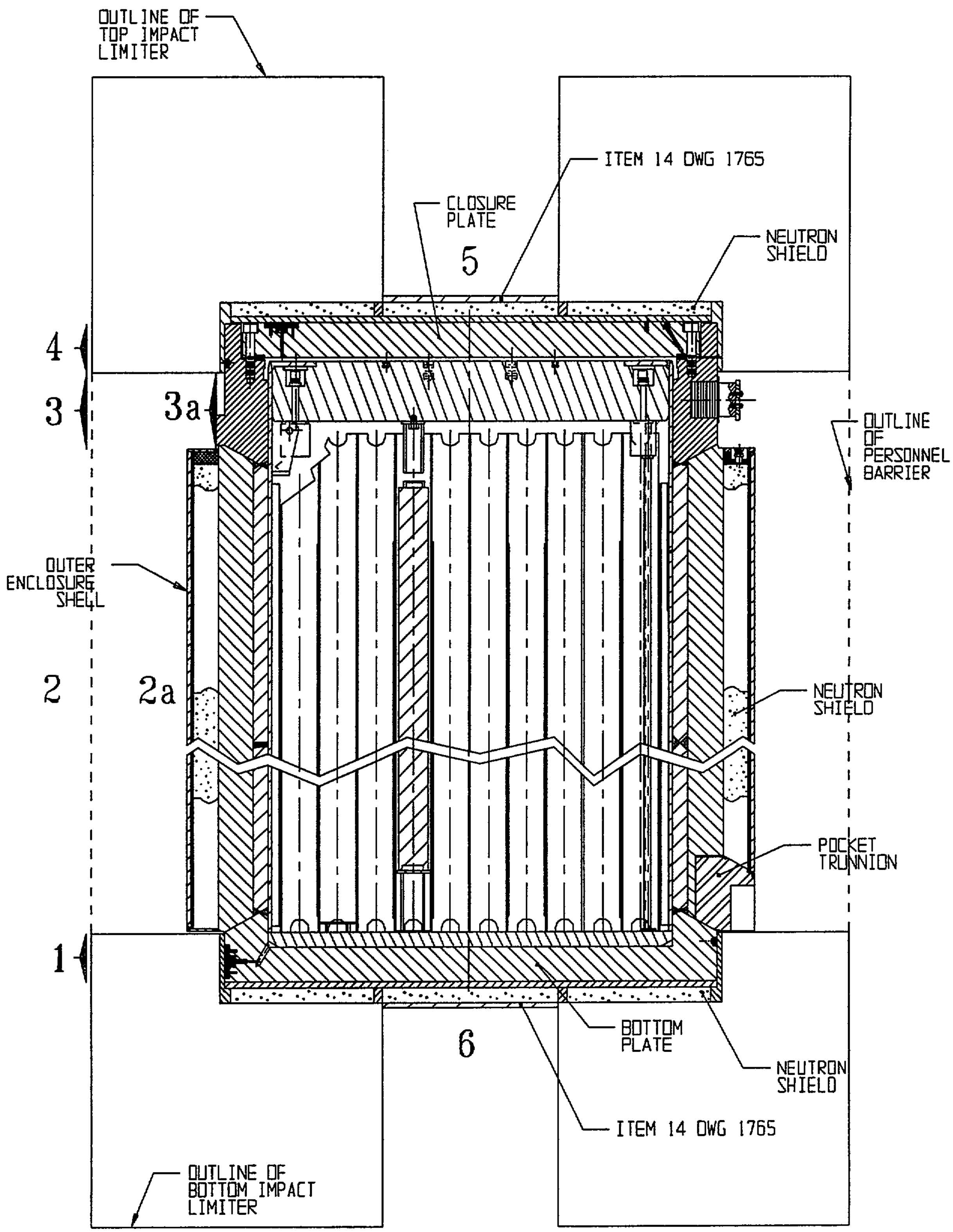

FIGURE 5.1.1; CROSS SECTION ELEVATION VIEW OF THE HI-STAR 100 SYSTEM WITH DOSE POINT LOCATIONS DURING NORMAL CONDITIONS

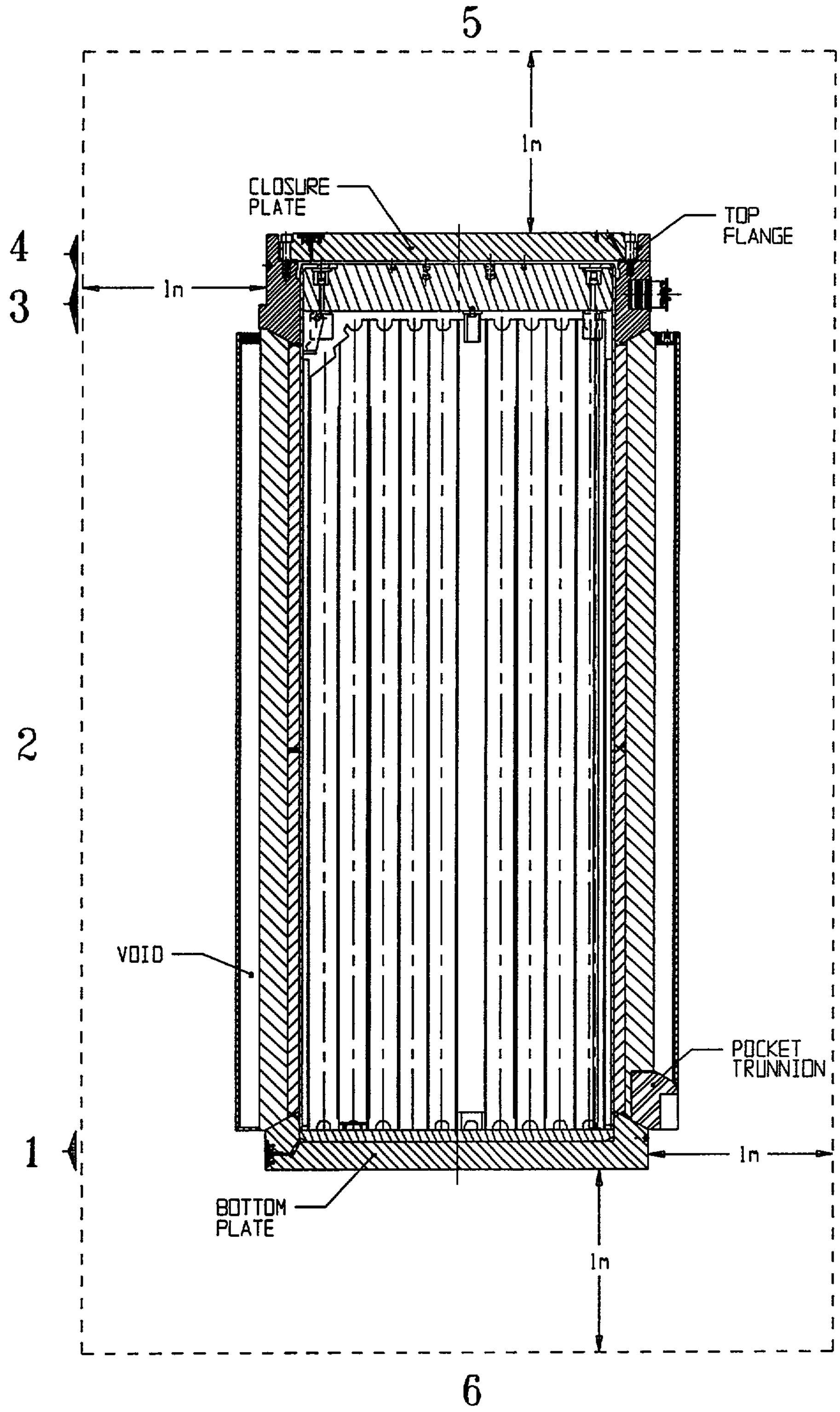

FIGURE 5.1.2; CROSS SECTION ELEVATION VIEW OF THE HI-STAR 100 SYSTEM WITH DOSE POINT LOCATIONS DURING ACCIDENT CONDITIONS

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]. *The source terms for the Trojan specific inventory were calculated with the SAS2H and ORIGEN-S sequences from the SCALE 4.4 system [5.1.4, 5.1.5] as described in the Trojan FSAR [5.1.6].* 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 assembly above and below the active fuel region. The third source is from (n,γ) reactions described below.

A description of the design basis intact 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 1.2.8: 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 1.2.9: 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. Subsection 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, which is also the design basis damaged fuel assembly for the Humboldt Bay and Dresden 1 damaged fuel or fuel debris, is described in Table 5.2.2. The design basis damaged fuel assembly is also the design basis fuel assembly for fuel debris. The fuel assembly type listed produces the highest total neutron and gamma sources from the fuel assemblies at Dresden 1 and Humboldt Bay. Table 5.2.15 provides a description of the design basis Dresden 1 MOX fuel assembly used in this analysis. The design basis 6x6, damaged, 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 5.2.18. This table also describes the design basis stainless steel clad LaCrosse fuel assembly.

Since the MPC-24E being used for Trojan fuel is slightly different than the standard MPC-24E, the Trojan contents were specifically analyzed and are not covered by the design basis PWR fuel assembly described above. The design basis Trojan WE 17x17 fuel assembly is described in Table 5.2.32 and was taken from the site specific Trojan FSAR analysis [5.1.6].

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, 5.2.15, ~~and~~ 5.2.18, *and 5.2.32* resulted in conservative source term calculations.

Subsections 5.2.1 and 5.2.2 describe the calculation of the gamma and neutron source terms for zircaloy clad fuel while Subsection 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 5.2.3 through 5.2.6 *and 5.2.33* provide the gamma source in MeV/s and photons/s as calculated with SAS2H and ORIGEN-S for the design bases intact fuels for the MPC-24, *MPC-32,* MPC-68, ~~and~~ the design basis damaged fuel, *and the Trojan fuel.* Table 5.2.16 provides the gamma source in MeV/s and photons/s for the design basis MOX fuel. NUREG-1617 [5.2.1] states that "In general, only gammas from approximately 0.8 MeV-2.5 MeV will contribute significantly to the external radiation levels." However, specific analysis for the HI-STAR 100 system has revealed that, due to the magnitude of the gamma source in the energy range just below 0.8 MeV, 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). 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. Photons with energies below 0.45 MeV are too weak to penetrate the steel of the overpack, and photons with energies above 3.0 MeV are too few to contribute significantly to the external dose. As discussed earlier, the MPC-24, *MPC-32,* and the MPC-68 are analyzed for transportation of spent nuclear fuel with varying minimum enrichments, burnup levels and cooling times. This section provides the radiation source for each of the burnup levels and cooling times evaluated.

The primary source of activity in the non-fuel regions of an assembly arise from the activation of ^{59}Co to ^{60}Co. The primary source of ^{59}Co in a fuel element is the steel and inconel structural material. The zircaloy in these regions is neglected since it does not have a significant ^{59}Co impurity level. Reference [5.2.3] indicates that the ^{59}Co impurity level in steel is 800 ppm or 0.8 gm/kg and in inconel is approximately 4700 ppm or 4.7 gm/kg. In the early to mid 1980s, the fuel vendors reduced the ^{59}Co impurity level in both inconel and steel to less than 500 ppm or 0.5 gm/kg. Prior to that, the impurity level in inconel in fuel assemblies was typically less than 1200 ppm or 1.2 gm/kg. Nevertheless, a conservative ^{59}Co impurity level of 1.0 gm/kg was used for the stainless steel end fittings and a highly conservative impurity level of 4.7 gm/kg was used for the inconel.

PWR fuel assemblies are currently manufactured with zircaloy incore grid spacers (the plenum spacer and the lower spacer are still inconel in some cases). However, earlier assemblies were manufactured with inconel incore grid spacers. Since the mass of the spacers is significant and since the cobalt impurity level assumed for inconel is very conservative, the Cobalt-60 activity from the incore spacers contributes significantly to the external dose rate. As a result, separate burnup and cooling times were developed for PWR assemblies that utilize zircaloy and non-zircaloy incore spacers. Since steel has a lower cobalt impurity level than inconel, any zircaloy clad PWR assemblies with stainless steel grid spacers are bounded by the analysis performed in this chapter utilizing inconel grid spacers. The BWR assembly grid spacers are zircaloy, however, some assembly designs have inconel springs in conjunction with the grid spacers. The gamma source for the BWR fuel assembly includes the activation of these springs associated with the grid spacers.

The non-fuel data listed in Table 5.2.1 was taken from References [5.2.3], [5.2.4], and [5.2.5]~~.~~ *while the non-fuel data listed in Table 5.2.32 was taken from References [5.2.5] and [5.2.8].* 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. These masses are larger than most other fuel assemblies from other manufacturers. This, in combination with the conservative ^{59}Co impurity level, results in a conservative estimate of the ^{60}Co activity.

The masses in Table 5.2.1 *and 5.2.32* were used to calculate a ^{59}Co impurity level in the fuel material. The grams of impurity were then used in ORIGEN-S to calculate a ^{60}Co activity level for the desired burnup and decay time. The methodology used to determine the activation level was developed from Reference [5.2.2] 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 5.2.7. These scaling factors were taken from Reference [5.2.2]. *In the case of the Trojan fuel, the higher value of 0.2 was used for both the gas plenum springs and spacer consistent with the Trojan FSAR [5.1.6].*

Tables 5.2.8 through 5.2.10 *and 5.2.34* provide the ^{60}Co activity utilized in the shielding calculations in the non-fuel regions of the assemblies for the MPC-24, *MPC-32,* ~~and~~ MPC-68, *and Trojan fuel.* The design basis damaged and MOX fuel assemblies are conservatively assumed to have the same ^{60}Co source strength as the BWR intact design basis fuel. This is a conservative assumption as the design basis damaged fuel and MOX fuel are limited to a significantly lower burnup and longer cooling time than the intact design basis zircaloy clad 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 as a function of burnup and cooling time. Conservatively, the minimum enrichments used to develop the source terms and dose rates presented in this chapter are specified in ~~Table 1.2.20~~*Appendix A to the CoC* as fuel assembly acceptance criteria. The minimum enrichments are also listed in Table 5.2.23 for convenience.

The enrichment analyzed for the Trojan fuel was 3.09 wt% ^{235}U. This is not the absolute minimum enrichment for Trojan fuel but is the lowest enrichment for fuel achieving the analyzed burnup. The minimum enrichment specified in Appendix A to the CoC is an absolute minimum enrichment that bounds all Trojan fuel.

The neutron source calculated for the design basis intact fuel assemblies for the MPC-24, *MPC-32,* ~~and~~ MPC-68, *Trojan fuel,* and the design basis damaged fuel are listed in Tables 5.2.11 through 5.2.14 *and 5.2.35* in neutrons/s. Table 5.2.17 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 5.2.18 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 the table is actually longer than the true active fuel length of 122 inches for the W15x15 and 83 inches for the A/C 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 actual fuel length and compare them directly to the source terms from the zircaloy clad fuel with a longer active fuel length.

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 [5.2.3] indicates that the Cobalt-59 impurity level in steel is 800 ppm or 0.8 gm/kg and in inconel is approximately 4700 ppm or 4.7 gm/kg. In the early to mid 1980s, the fuel vendors reduced the Cobalt-59 impurity level in both inconel and steel to less than 500 ppm or 0.5 gm/kg. Prior to that, the impurity level in inconel in fuel assemblies was typically less than 1200 ppm or 1.2 gm/kg. Nevertheless, a conservative Cobalt-59 impurity level of 0.8 gm/kg was used for the stainless steel cladding and a highly conservative impurity level of 4.7 gm/kg was used for the inconel incore spacers. It is assumed that the end fitting masses of the stainless steel clad fuel are the same as the end fittings masses of the zircaloy clad fuel. Therefore, separate source terms are not provided for the end fittings of the stainless steel fuel.

Tables 5.2.19 through 5.2.22 list the neutron and gamma source strengths for the design basis stainless steel clad fuel. The gamma source strengths include the contribution from the cobalt activation in the incore spacers. Subsection 5.4.4 presents the dose rates around the HI-STAR 100 for the normal and hypothetical accident conditions for the stainless steel fuel. In the calculation of these dose rates the length of the active fuel was conservatively assumed to be 144 inches. In addition, the fuel assembly configuration used in the MCNP calculations was identical to the configuration used for the design basis fuel assemblies as described in Table 5.3.1.

5.2.4 ~~Control Components~~*Non-fuel Hardware*

Generic ~~Ccontrol components~~*PWR non-fuel hardware is* ~~are~~ not permitted for transport in the HI-STAR 100 system. *However, certain non-fuel hardware from the Trojan Nuclear plant has been analyzed and is approved for transportation. These components include rod cluster control assemblies (RCCAs), burnable poison rod assemblies (BPRAs) and thimble plug devices (TPDs). The methodology for analyzing the non-fuel hardware authorized for transportation is described below and has been previously approved in the HI-STORM 100 FSAR [5.2.9].*

5.2.4.1 BPRAs and TPDs

Burnable poison rod assemblies (BPRA) and thimble plug devices (TPD) 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 may 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. 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 Trojan 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 Trojan 17x17 fuel assembly. The mass of material in the regions above the active fuel zone was scaled by the appropriate scaling factors listed in Table 5.2.7 in order to account for the reduced flux levels above the fuel assembly. The total curies of cobalt were calculated for the Trojan TPDs and BPRAs for the actual burnups and cooling times (the BPRAs were only used in the first cycle whereas the TPDs were used in all but the last cycle). The accumulated burnup and cooling time for the BPRAs and TPDs are 15,998 MWD/MTU and 24 years cooling and 118,674 MWD/MTU and 11 years cooling, respectively. Since the operating history of the shutdown Trojan reactor is well known the actual cycle lengths and conservatively short downtimes between cycles were used in the calculation of the source terms. In the ORIGEN-S calculations it was assumed that the burned fuel assembly was replaced with a fresh fuel assembly after every cycle. This was achieved in ORIGEN-S by resetting the flux levels and cross sections to the 0 MWD/MTU condition after every cycle.

Currently only the Trojan non-fuel hardware is permitted for transportation in the HI-STAR 100 System. The masses of the Trojan TPD and BPRA are listed in Table 5.2.36. This information was taken from references [5.2.5] and [5.2.7] and is the same information used in the Trojan FSAR [5.1.6].

Table 5.2.37 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 cooling time, separate from the fuel assemblies, of 24 years and 11 years is used for the Trojan BPRAs and TPDs, respectively.

Subsection 5.4.7 discusses the analysis of cask dose rates from Trojan fuel including the effect of the insertion of BPRAs or TPDs into Trojan fuel assemblies.

5.2.4.2 RCCAs

Rod cluster control assemblies (RCCAs) 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 RCCAs are utilized vary from plant to plant. Some utilities maintain the RCCAs 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 RCCAs 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 RCCAs. In all cases, however, only the lower portion of the RCCAs will be significantly activated. Therefore, when the RCCAs are stored with the PWR fuel assembly, the activated portion of the RCCAs will be in the lower portion of the cask. RCCAs are fabricated of various materials. The cladding is typically stainless steel, although inconel has been used. The absorber can be a single material or a combination of materials. AgInCd is possibly the most common absorber although B_4C in aluminum is used, and hafnium has also been used. AgInCd produces a noticeable source term in the 0.3-1.0 MeV range due to the activation of Ag. The Trojan RCCAs, the only RCCAs currently authorized for transport, were made of AgInCd clad in stainless steel.*

In order to determine the impact on the dose rates around the HI-STAR 100 System, source terms for the Trojan RCCAs 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 Trojan W 17x17 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 for a single burnup, 125,515 MWD/MTU, and cooling time, 9 years, corresponding to the lifetime operation of the Trojan reactor. Since the operating history of the shutdown Trojan reactor is well known the actual cycle lengths and conservatively short downtimes between cycles were used in the calculation of the source terms. In the ORIGEN-S calculations it was assumed that the burned fuel assembly was replaced with a fresh fuel assembly after every cycle. This was achieved in ORIGEN-S by resetting the flux levels and cross sections to the 0 MWD/MTU condition after every cycle. 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. Since the Trojan reactor normally operated with all RCCA rods fully withdrawn only one configuration was analyzed for the RCCAs. The configuration, which is summarized below, is described in Table 5.2.38 for the RCCAs. The masses of the materials listed in these tables were determined from reference [5.2.5]. The masses listed in Table 5.2.38 do not match exact values from [5.2.5] because the values in the reference were adjusted to the lengths shown in the tables.

<u>RCCA Configuration</u>
This configuration represents a fully removed RCCA 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.

Table 5.2.38 presents the source terms that were calculated for the Trojan RCCAs. 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.

Subsection 5.4.7 discusses the analysis of cask dose rates from Trojan fuel including the effect of the insertion of RCCAs into Trojan fuel assemblies.

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 1.2.8 and 1.2.9. 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 which 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 baskets (*MPC-24 *and MPC-32)* and the *BWR basket (*MPC-68*)*.

5.2.5.1 PWR Design Basis Assembly

Table 1.2.8 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 5.2.24 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 5.2.24. 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 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 5.2.24 were analyzed at the same burnup and cooling time. The results of the comparison are provided in Table 5.2.26. 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 1.2.8. This fuel assembly also has the highest UO_2 mass (see Table 5.2.24) 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 Subsection 5.2.3. Based on the results in Table 5.2.26, 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.

5.2.5.2 BWR Design Basis Assembly

Table 1.2.9 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 5.2.25 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 5.2.25 were analyzed at the same burnup and cooling time. The results of the comparison are provided in Table 5.2.27. These results indicate that the 7x7 fuel assembly has the highest radiation source term of the zircaloy clad fuel assembly classes considered in Table 1.2.9. 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 Table 1.2.20. 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 times for this assembly are limited to 22,500 MWD/MTU and 15-year cooling as specified in Table 1.2.19. This assembly type is discussed further in Subsection 5.2.3.

The Humboldt Bay 6x6 and Dresden 1 6x6 fuel are older and shorter 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 damaged fuel assembly and the design basis intact 6x6 fuel assembly are identical, the analysis presented in Subsection 5.4.2 for the damaged fuel assembly also demonstrates the acceptability of transporting intact 6x6 fuel assemblies from the Dresden 1 and Humboldt Bay fuel assembly classes.

5.2.5.3 Decay Heat Loads

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.~~ As demonstrated in Tables 5.2.26 and 5.2.27, 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, Tables 1.2.10 and 1.2.11 limit the heavy metal mass of the design basis fuel assembly classes to a value less than the design basis value utilized in this chapter. This provides additional assurance that the ~~decay heat values~~*radiation source terms* 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 5.2.28. 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 Tables 1.2.10 and 1.2.11 are limited to the masses used in Tables 5.2.24 and 5.2.25. 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~~*radiation source terms* ~~-~~for the non-design basis fuel assemblies are bounding values is obtained by using the ~~decay heat values~~ *radiation source terms* for the design basis fuel assemblies in determining the acceptable loading criteria for all fuel assemblies.

~~As mentioned above, Table 5.2.28 demonstrates the level of conservatism in applying the decay heat from the design basis fuel assembly to all fuel assemblies.~~

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 5.2.29 describes the 8x8 fuel assembly that contains the thoria rods. Table 5.2.30 and 5.2.31 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 5.2.6 and 5.2.14 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.

Subsection 5.4.6 provides a further discussion of the thoria rod ~~cansiter~~*canister* and its ~~acceptablity~~*acceptability* for transport in the HI-STAR 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 Subsection 5.4.5 to demonstrate that they are acceptable for transport in the HI-STAR 100 System.

Trojan Nuclear Power has a couple of primary (californium) neutron sources and a few secondary (antimony-beryllium) neutron sources. The sources are basically BPRAs with the source material placed in a few of the rods instead of burnable absorber. In the case of the californium source, a single rod contained the source while the remaining 19 rods were burnable absorber. The secondary sources used 4 rods for the antimony-beryllium source and the remaining rods were burnable poison or were not used. The californium neutron sources were used in the startup of the reactor and have significantly decayed since their use. Therefore, the source of neutrons from the californium sources is negligible and they are considered to be acceptable for transport without further analysis. The antimony-beryllium sources have been analyzed in Subsection 5.4.8 to demonstrate that they are acceptable for transport.

Currently these are the only neutron source permitted for transport in the HI-STAR 100 System.

5.2.8 Trojan Non-Fuel Bearing Components, Damaged Fuel and Fuel Debris

Trojan Nuclear Power has failed fuel cans containing fuel process can capsules and fuel debris. The fuel process can capsules contain only a limited amount of fuel in the form of fuel debris (metal fragments). The source term from the fuel process can capsules is therefore bounded by the source from a fuel assembly.

The fuel assemblies classified as fuel debris consist of a few assemblies with each containing a maximum of 17 rods. Therefore, even in a collapsed state which might exist after a transport accident, this fuel debris is bounded by an intact fuel assembly and therefore is not explicitly considered in the analysis in this chapter. There are also a couple of fuel assemblies classified as damaged fuel because of missing rods. These assemblies are also bounded by an intact assembly and during the transport accident it is expected that these damaged assemblies would react the same as intact assemblies. Therefore, the Trojan damaged fuel assemblies were not explicitly considered in the analysis in this chapter.

Trojan fuel assembly hardware, non-fuel bearing components and one fuel skeleton will also be transported. These components are made of stainless steel, zircaloy and inconel. The source term from these additional components were not explicitly considered but are bounded by intact fuel assemblies. Therefore, the source term from these components were not explicitly considered.

Table 5.2.1

DESCRIPTION OF DESIGN BASIS INTACT ZIRCALOY CLAD FUEL

	PWR	BWR
Assembly type/class	B&W 15×15	GE 7×7
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)	See ~~Table 1.2.20~~*Appendix A to the CoC*	See *Appendix A to the CoC*~~Table 1.2.20~~
Burnup (MWD/MTU)	See *Appendix A to the CoC*~~Table 1.2.20~~	See *Appendix A to the CoC*~~Table 1.2.20~~
Cooling Time (years)	See *Appendix A to the CoC*~~Table 1.2.20~~	See *Appendix A to the CoC*~~Table 1.2.20~~
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 1.2.8: 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 1.2.9: GE BWR/2-3, GE BWR/4-6, Humboldt Bay 7x7, and Dresden 1 8x8.

† Derived from parameters in this table.

Table 5.2.1 (continued)

DESCRIPTION OF DESIGN BASIS INTACT ZIRCALOY CLAD FUEL

	PWR	BWR
No. of Water Rods/Guide Tubes	17	0
Water Rod O.D. (in.)	0.53	N/A
Water Rod Thickness (in.)	0.0160	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)	0.55572 (inconel) 0.27252 (steel)	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)

† This mass of inconel was used for fuel assemblies with non-zircaloy grid spacers. For fuel assemblies with zircaloy grid spacers the mass was 0.0. However, the mass of the inconel and steel in the other assembly components are identical for assemblies with zircaloy and non-zircaloy incore grid spacers.

Table 5.2.2

DESCRIPTION OF DESIGN BASIS DAMAGED 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)	1.8
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
Incore spacers (kg inconel)	1.07

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

CALCULATED MPC-24 *and MPC-32* PWR FUEL GAMMA SOURCE PER ASSEMBLY FOR DESIGN BASIS ZIRCALOY CLAD FUEL WITH NON-ZIRCALOY INCORE SPACERS FOR VARYING BURNUPS AND COOLING TIMES

Lower Energy	**Upper Energy**	**24,500 MWD/MTU 10 Year Cooling**		**~~29~~*24*,500 MWD/MTU 12 Year Cooling**		**~~34~~*42*,500 MWD/MTU ~~14~~ *20* Year Cooling**	
(MeV)	(MeV)	(MeV/s)	(Photons/s)	(MeV/s)	(Photons/s)	(MeV/s)	(Photons/s)
0.45	0.7	7.23E+14	1.26E+15	*6.67E+14*	*1.16E+15*	*9.13E+14*	*1.59E+15*
0.7	1.0	6.84E+13	8.05E+13	*4.19E+13*	*4.93E+13*	*2.46E+13*	*2.89E+13*
1.0	1.5	2.88E+13	2.30E+13	*2.29E+13*	*1.83E+13*	*2.46E+13*	*1.97E+13*
1.5	2.0	1.49E+12	8.54E+11	*1.23E+12*	*7.05E+11*	*1.46E+12*	*8.36E+11*
2.0	2.5	1.02E+11	4.51E+10	*2.57E+10*	*1.14E+10*	*8.07E+09*	*3.59E+09*
2.5	3.0	6.77E+09	2.46E+09	*1.87E+09*	*6.78E+08*	*6.89E+08*	*2.51E+08*
Totals		8.22E+14	1.36E+15	*7.33E+14*	*1.23E+15*	*9.63E+14*	*1.64E+15*
~~Lower Energy~~	**~~Upper Energy~~**	**~~37,500 MWD/MTU 15 Year Cooling~~**					
~~(MeV)~~	~~(MeV)~~	~~(MeV/s)~~	~~(Photons/s)~~				
~~0.45~~	~~0.7~~						
~~0.7~~	~~1.0~~						
~~1.0~~	~~1.5~~						
~~1.5~~	~~2.0~~						
~~2.0~~	~~2.5~~						
~~2.5~~	~~3.0~~						
~~Totals~~							

Table 5.2.4

CALCULATED MPC-24 *and MPC-32* PWR FUEL GAMMA SOURCE PER ASSEMBLY FOR DESIGN BASIS ZIRCALOY CLAD FUEL WITH ZIRCALOY INCORE SPACERS FOR VARYING BURNUPS AND COOLING TIMES

Lower Energy	**Upper Energy**	**~~24~~*34*,500 MWD/MTU ~~7~~ *9* Year Cooling**		**~~29~~*34*,500 MWD/MTU ~~8~~ *12* Year Cooling**		**~~34~~*44*,500 MWD/MTU ~~10~~ *18* Year Cooling**	
(MeV)	(MeV)	(MeV/s)	(Photons/s)	(MeV/s)	(Photons/s)	(MeV/s)	(Photons/s)
0.45	0.7	*1.08E+15*	*1.88E+15*	*9.38E+14*	*1.63E+15*	*1.00E+15*	*1.75E+15*
0.7	1.0	*1.48E+14*	*1.74E+14*	*6.91E+13*	*8.13E+13*	*3.32E+13*	*3.90E+13*
1.0	1.5	*5.44E+13*	*4.35E+13*	*3.78E+13*	*3.03E+13*	*3.07E+13*	*2.46E+13*
1.5	2.0	*2.70E+12*	*1.54E+12*	*1.99E+12*	*1.14E+12*	*1.78E+12*	*1.02E+12*
2.0	2.5	*2.57E+11*	*1.14E+11*	*3.18E+10*	*1.41E+10*	*9.01E+09*	*4.00E+09*
2.5	3.0	*1.67E+10*	*6.09E+09*	*2.53E+09*	*9.20E+08*	*8.14E+08*	*2.96E+08*
Totals		*1.29E+15*	*2.10E+15*	*1.05E+15*	*1.74E+15*	*1.07E+15*	*1.81E+15*

~~**Lower Energy**~~	~~**Upper Energy**~~	~~**39,500 MWD/MTU 12 Year Cooling**~~		~~**44,500 MWD/MTU 15 Year Cooling**~~	
~~(MeV)~~	~~(MeV)~~	~~(MeV/s)~~	~~(Photons/s)~~	~~(MeV/s)~~	~~(Photons/s)~~
~~0.45~~	~~0.7~~	~~1.07E+15~~	~~1.87E+15~~	~~1.09E+15~~	~~1.90E+15~~
~~0.7~~	~~1.0~~	~~8.29E+13~~	~~9.76E+13~~	~~5.27E+13~~	~~6.20E+13~~
~~1.0~~	~~1.5~~	~~4.53E+13~~	~~3.62E+13~~	~~3.97E+13~~	~~3.18E+13~~
~~1.5~~	~~2.0~~	~~2.37E+12~~	~~1.35E+12~~	~~2.20E+12~~	~~1.26E+12~~
~~2.0~~	~~2.5~~	~~3.43E+10~~	~~1.52E+10~~	~~1.22E+10~~	~~5.41E+09~~
~~2.5~~	~~3.0~~	~~2.85E+09~~	~~1.04E+09~~	~~1.10E+09~~	~~4.00E+08~~
~~Totals~~		~~1.20E+15~~	~~2.00E+15~~	~~1.18E+15~~	~~1.99E+15~~

Table 5.2.5
CALCULATED MPC-68 BWR FUEL GAMMA SOURCE PER ASSEMBLY FOR DESIGN BASIS ZIRCALOY CLAD FUEL FOR VARYING BURNUPS AND COOLING TIMES

Lower Energy	**Upper Energy**	**24,500 MWD/MTU 8 Year Cooling**		**~~29~~*39*,500 MWD/MTU ~~9~~ *14* Year Cooling**		**~~34~~*44*,500 MWD/MTU ~~12~~ *19* Year Cooling**	
(MeV)	(MeV)	(MeV/s)	(Photons/s)	(MeV/s)	(Photons/s)	(MeV/s)	(Photons/s)
0.45	0.7	3.21E+14	5.57E+14	*3.96E+14*	*6.89E+14*	*3.87E+14*	*6.73E+14*
0.7	1.0	4.54E+13	5.34E+13	*2.03E+13*	*2.39E+13*	*1.09E+13*	*1.28E+13*
1.0	1.5	1.46E+13	1.17E+13	*1.38E+13*	*1.10E+13*	*1.05E+13*	*8.38E+12*
1.5	2.0	7.68E+11	4.39E+11	*7.59E+11*	*4.34E+11*	*6.17E+11*	*3.53E+11*
2.0	2.5	1.65E+11	7.34E+10	*5.26E+09*	*2.34E+09*	*3.34E+09*	*1.48E+09*
2.5	3.0	9.32E+09	3.39E+09	*4.16E+08*	*1.51E+08*	*2.84E+08*	*1.03E+08*
Total		3.82E+14	6.23E+14	*4.31E+14*	*7.24E+14*	*4.09E+14*	*6.95E+14*

~~Lower Energy~~	**~~Upper Energy~~**	**~~39,500 MWD/MTU 15 Year Cooling~~**	
~~(MeV)~~	~~(MeV)~~	~~(MeV/s)~~	~~(Photons/s)~~
~~0.45~~	~~0.7~~	~~3.84E+14~~	~~6.68E+14~~
~~0.7~~	~~1.0~~	~~1.68E+13~~	~~1.98E+13~~
~~1.0~~	~~1.5~~	~~1.26E+13~~	~~1.01E+13~~
~~1.5~~	~~2.0~~	~~7.06E+11~~	~~4.04E+11~~
~~2.0~~	~~2.5~~	~~4.20E+09~~	~~1.87E+09~~
~~2.5~~	~~3.0~~	~~3.16E+08~~	~~1.15E+08~~
~~Totals~~		~~4.14E+14~~	~~6.99E+14~~

Table 5.2.6

CALCULATED MPC-68 and MPC-68F BWR FUEL GAMMA SOURCE PER ASSEMBLY FOR DESIGN BASIS ZIRCALOY CLAD DAMAGED FUEL

Lower Energy	**Upper Energy**	**30,000 MWD/MTU 18 Year Cooling**	
(MeV)	(MeV)	(MeV/s)	(Photons/s)
0.45	0.7	1.52E+14	2.65E+14
0.7	1.0	4.07E+12	4.79E+12
1.0	1.5	3.80E+12	3.04E+12
1.5	2.0	2.24E+11	1.28E+11
2.0	2.5	1.26E+9	5.58E+8
2.5	3.0	7.42E+7	2.70E+7
Totals		1.61E+14	2.73E+14

Table 5.2.7

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

Region	**PWR**	**BWR**
Handle	N/A	0.05
Top 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
Grid spacer spring	N/A	1.0
Bottom end fitting	0.2	0.15

Table 5.2.8

CALCULATED MPC-24 *and MPC-32* ^{60}Co SOURCE PER ASSEMBLY FOR DESIGN BASIS ZIRCALOY CLAD FUEL WITH NON-ZIRCALOY INCORE SPACERS AT VARYING BURNUPS AND COOLING TIMES

Location	**24,500 MWD/MTU 10 Year Cooling (curies)**	**~~29~~*24*,500 MWD/MTU 12 Year Cooling (curies)**	**~~34~~*42*,500 MWD/MTU ~~14~~*20* Year Cooling (curies)**	**~~37,500 MWD/MTU 15 Year Cooling (curies)~~**
Lower end fitting	83.92	*64.65*	*30.42*	~~54.25~~
Gas plenum springs	16.33	*12.58*	*5.92*	~~10.56~~
Gas plenum spacer	9.37	*7.22*	*3.40*	~~6.06~~
Expansion springs	N/A	N/A	N/A	~~N/A~~
Grid spacers	762.29	*587.27*	*276.36*	~~492.84~~
Upper end fitting	30.72	*23.66*	*11.14*	~~19.86~~
Handle	N/A	N/A	N/A	~~N/A~~

Table 5.2.9

CALCULATED MPC-24 *and MPC-32* ^{60}Co SOURCE PER ASSEMBLY FOR DESIGN BASIS ZIRCALOY CLAD FUEL WITH ZIRCALOY INCORE SPACERS AT VARYING BURNUPS AND COOLING TIMES

Location	**~~24~~*34*,500 MWD/MTU ~~7~~ *9* Year Cooling (curies)**	**~~29~~*34*,500 MWD/MTU ~~8~~ *12* Year Cooling (curies)**	**~~34~~*44*,500 MWD/MTU ~~10~~ *18* Year Cooling (curies)**	**~~39,500 MWD/MTU 12 Year Cooling (curies)~~**	**~~44,500 MWD/MTU 15 Year Cooling (curies)~~**
Lower end fitting	*116.12*	*78.34*	*41.33*	~~84.17~~	~~61.10~~
Gas plenum springs	*22.59*	*15.24*	*8.04*	~~16.38~~	~~11.89~~
Gas plenum spacer	*12.96*	*8.74*	*4.61*	~~9.40~~	~~6.82~~
Expansion springs	N/A	N/A	N/A	~~N/A~~	~~N/A~~
Grid spacers†	N/A	N/A	N/A	~~N/A~~	~~N/A~~
Upper end fitting	*42.50*	*28.68*	*15.13*	~~30.81~~	~~22.36~~
Handle	N/A	N/A	N/A	~~N/A~~	~~N/A~~

† These burnup and cooling times represent fuel with zircaloy grid spacers. Therefore, the cobalt activation is negligible.

Table 5.2.10

CALCULATED MPC-68 ^{60}CO SOURCE PER ASSEMBLY FOR DESIGN BASIS ZIRCALOY CLAD FUEL AT VARYING BURNUPS AND COOLING TIMES

Location	**24,500 MWD/MTU 8 Year Cooling (curies)**	**~~29~~*39*,500 MWD/MTU ~~9~~ *14* Year Cooling (curies)**	**~~34~~*44*,500 MWD/MTU ~~12~~ *19* Year Cooling (curies)**	**~~39,500 MWD/MTU 15 Year Cooling (curies)~~**
Lower end fitting	34.04	*19.64*	*11.08*	~~17.21~~
Gas plenum springs	10.40	*6.00*	*3.39*	~~5.26~~
Gas plenum spacer	N/A	N/A	N/A	~~N/A~~
Expansion springs	1.89	*1.09*	*0.62*	~~0.96~~
Grid spacers	73.32	*42.30*	*23.88*	~~37.08~~
Upper end fitting	9.45	*5.45*	*3.08*	~~4.78~~
Handle	1.18	*0.68*	*0.38*	~~0.60~~

Table 5.2.11

CALCULATED MPC-24 *and MPC-32* PWR NEUTRON SOURCE PER ASSEMBLY FOR DESIGN BASIS ZIRCALOY CLAD FUEL WITH NON-ZIRCALOY INCORE SPACERS FOR VARYING BURNUPS AND COOLING TIMES

Lower Energy (MeV)	**Upper Energy (MeV)**	**24,500 MWD/MTU 10 Year Cooling (Neutrons/s)**	**~~29~~*24*,500 MWD/MTU 12 Year Cooling (Neutrons/s)**	**~~34~~*42*,500 MWD/MTU ~~14~~ *20* Year Cooling (Neutrons/s)**	**~~37,500 MWD/MTU 15 Year Cooling (Neutrons/s)~~**
1.0E-01	4.0E-01	2.41E+06	*2.24E+06*	*8.08E+06*	~~6.56E+06~~
4.0E-01	9.0E-01	1.23E+07	*1.15E+07*	*4.13E+07*	~~3.35E+07~~
9.0E-01	1.4	1.14E+07	*1.06E+07*	*3.79E+07*	~~3.08E+07~~
1.4	1.85	8.46E+06	*7.88E+06*	*2.81E+07*	~~2.28E+07~~
1.85	3.0	1.52E+07	*1.43E+07*	*5.02E+07*	~~4.08E+07~~
3.0	20.0	1.35E+07	*1.26E+07*	*4.50E+07*	~~3.66E+07~~
6.43	20.0	1.18E+06	*1.10E+06*	*3.95E+06*	~~3.21E+06~~
TOTALS		6.45E+07	*6.01E+07*	*2.15E+08*	~~1.74E+08~~

Table 5.2.12

CALCULATED MPC-24 *and MPC-32* PWR NEUTRON SOURCE PER ASSEMBLY
FOR DESIGN BASIS ZIRCALOY CLAD FUEL WITH ZIRCALOY
INCORE SPACERS FOR VARYING BURNUPS AND COOLING TIMES

Lower Energy (MeV)	**Upper Energy (MeV)**	**~~24~~*34*,500 MWD/MTU ~~7~~ *9* Year Cooling (Neutrons/s)**	**~~29~~*34*,500 MWD/MTU ~~8~~ *12* Year Cooling (Neutrons/s)**	**~~34~~*44*,500 MWD/MTU ~~10~~ *18* Year Cooling (Neutrons/s)**	**~~39,500 MWD/MTU 12 Year Cooling (Neutrons/s)~~**	**~~44,500 MWD/MTU 15 Year Cooling (Neutrons/s)~~**
1.0E-01	4.0E-01	*6.98E+06*	*6.24E+06*	*1.05E+07*	~~9.05E+06~~	~~1.17E+07~~
4.0E-01	9.0E-01	*3.57E+07*	*3.19E+07*	*5.35E+07*	~~4.63E+07~~	~~5.99E+07~~
9.0E-01	1.4	*3.27E+07*	*2.93E+07*	*4.91E+07*	~~4.24E+07~~	~~5.49E+07~~
1.4	1.85	*2.42E+07*	*2.17E+07*	*3.63E+07*	~~3.14E+07~~	~~4.06E+07~~
1.85	3.0	*4.30E+07*	*3.86E+07*	*6.47E+07*	~~5.58E+07~~	~~7.21E+07~~
3.0	20.0	*3.88E+07*	*3.47E+07*	*5.82E+07*	~~5.03E+07~~	~~6.50E+07~~
6.43	20.0	*3.42E+06*	*3.05E+06*	*5.13E+06*	~~4.43E+06~~	~~5.74E+06~~
TOTALS		*1.85E+08*	*1.65E+08*	*2.77E+08*	~~2.40E+08~~	~~3.10E+08~~

Table 5.2.13

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)**	**24,500 MWD/MTU 8 Year Cooling (Neutrons/s)**	**~~29~~*39*,500 MWD/MTU ~~9~~-*14* Year Cooling (Neutrons/s)**	**~~34~~*44*,500 MWD/MTU ~~12~~-*19* Year Cooling (Neutrons/s)**	**~~39,500 MWD/MTU 15 Year Cooling (Neutrons/s)~~**
1.0E-01	4.0E-01	1.08E+06	*3.61E+06*	*4.57E+06*	~~3.48E+06~~
4.0E-01	9.0E-01	5.52E+06	*1.84E+07*	*2.33E+07*	~~1.78E+07~~
9.0E-01	1.4	5.08E+06	*1.69E+07*	*2.14E+07*	~~1.63E+07~~
1.4	1.85	3.77E+06	*1.25E+07*	*1.58E+07*	~~1.20E+07~~
1.85	3.0	6.75E+06	*2.22E+07*	*2.80E+07*	~~2.14E+07~~
3.0	6.43	6.04E+06	*2.00E+07*	*2.53E+07*	~~1.93E+07~~
6.43	20.0	5.29E+05	*1.77E+06*	*2.24E+06*	~~1.70E+06~~
TOTALS		2.88E+07	*9.54E+07*	*1.21E+08*	~~9.19E+07~~

Table 5.2.14

CALCULATED MPC-68 and MPC-68F BWR NEUTRON SOURCE PER ASSEMBLY FOR DESIGN BASIS DAMAGED ZIRCALOY CLAD FUEL

Lower Energy (MeV)	**Upper Energy (MeV)**	**30,000 MWD/MTU 18 Year Cooling (Neutrons/s)**
1.0E-01	4.0E-01	1.18E+6
4.0E-01	9.0E-01	6.05E+6
9.0E-01	1.4	5.55E+6
1.4	1.85	4.11E+6
1.85	3.0	7.34E+6
3.0	6.43	6.59E+6
6.43	20.0	5.79E+5
Totals		3.14E+7

Table 5.2.15

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)†	1.8 (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
Incore spacers (kg inconel)	1.07

† See Table 5.3.3 for detailed composition of $PuUO_2$ rods.

†† Derived from parameters in this table.

Table 5.2.16

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)	(MeV)	(MeV/s)	(Photons/s)
0.45	0.7	1.45E+14	2.52E+14
0.7	1.0	3.95E+12	4.65E+12
1.0	1.5	3.82E+12	3.06E+12
1.5	2.0	2.22E+11	1.27E+11
2.0	2.5	1.11E+9	4.93E+8
2.5	3.0	9.31E+7	3.39E+7
Totals		1.53E+14	2.60E+14

Table 5.2.17

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	1.50E+6
4.0E-01	9.0E-01	7.67E+6
9.0E-01	1.4	7.09E+6
1.4	1.85	5.31E+6
1.85	3.0	9.67E+6
3.0	6.43	8.47E+6
6.43	20.0	7.33E+5
Totals		4.04E+7

Table 5.2.18
DESCRIPTION OF DESIGN BASIS INTACT STAINLESS STEEL CLAD FUEL

	PWR	BWR
Fuel type	WE 15x15	A/C 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)	3.1	3.5
Burnup (MWD/MTU)	30,000 @ 19 yr (MPC-24) 40,000 @ 24 yr (MPC-24)	22,500 (MPC-68)
Cooling Time (years)	19 (MPC-24) 24 (MPC-24)	16 (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
Incore spacers (kg inconel)	5.1	0.83

Notes:
1. The WE 15x15 is the design basis assembly for the following fuel assembly classes listed in Table 1.2.8: Haddam Neck and San Onofre 1.
2. The A/C 10x10 is the design basis assembly for the following fuel assembly class listed in Table 1.2.9: LaCrosse.

Table 5.2.19

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

Lower Energy	**Upper Energy**	**22,500 MWD/MTU 16-Year Cooling**	
(MeV)	(MeV)	(MeV/s)	(Photons/s)
0.45	0.7	2.26E+14	3.94E+14
0.7	1.0	6.02E+12	7.08E+12
1.0	1.5	4.04E+13	3.23E+13
1.5	2.0	2.90E+11	1.66E+11
2.0	2.5	2.94E+9	1.31E+9
2.5	3.0	7.77E+7	2.83E+7
Totals		2.73E+14	4.33E+14

Note:
1. These source terms were calculated for a 144 inch active fuel length. The actual active fuel length is 83 inches.
2. The ^{60}Co activation from incore spacers is included in the 1.0-1.5 MeV energy group.

Table 5.2.20

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

Lower Energy	**Upper Energy**	**30,000 MWD/MTU 19-Year Cooling**		**40,000 MWD/MTU 24-Year Cooling**	
(MeV)	(MeV)	(MeV/s)	(Photons/s)	(MeV/s)	(Photons/s)
0.45	0.7	6.81E+14	1.18E+15	7.97E+14	1.39E+15
0.7	1.0	1.83E+13	2.16E+13	1.70E+13	2.01E+13
1.0	1.5	1.13E+14	9.06E+13	8.24E+13	6.60E+13
1.5	2.0	1.06E+12	6.04E+11	1.12E+12	6.42E+11
2.0	2.5	7.25E+9	3.22E+9	7.42E+9	3.30E+9
2.5	3.0	3.52E+8	1.28E+8	6.43E+8	2.34E+8
Totals		8.14E+14	1.30E+15	8.98E+14	1.47E+15

Note:
1. These source terms were calculated for a 144 inch active fuel length. The actual active fuel length is 122 inches.
2. The ^{60}Co activation from incore spacers is included in the 1.0-1.5 MeV energy group.

Table 5.2.21

CALCULATED BWR NEUTRON SOURCE PER ASSEMBLY
FOR STAINLESS STEEL CLAD FUEL

Lower Energy (MeV)	**Upper Energy (MeV)**	**22,500 MWD/MTU 16-Year Cooling (Neutrons/s)**
1.0E-01	4.0E-01	1.81E+5
4.0E-01	9.0E-01	9.26E+5
9.0E-01	1.4	8.75E+5
1.4	1.85	6.85E+5
1.85	3.0	1.34E+6
3.0	6.43	1.08E+6
6.43	20.0	8.77E+4
Total		5.18E+6

Note:
These source terms were calculated for a 144 inch active fuel length. The actual active fuel length is 83 inches.

Table 5.2.22

CALCULATED PWR NEUTRON SOURCE PER ASSEMBLY
FOR STAINLESS STEEL CLAD FUEL

Lower Energy (MeV)	**Upper Energy (MeV)**	**30,000 MWD/MTU 19-Year Cooling (Neutrons/s)**	**40,000 MWD/MTU 24-Year Cooling (Neutrons/s)**
1.0E-01	4.0E-01	2.68E+6	7.07E+6
4.0E-01	9.0E-01	1.37E+7	3.61E+7
9.0E-01	1.4	1.27E+7	3.32E+7
1.4	1.85	9.50E+6	2.47E+7
1.85	3.0	1.74E+7	4.43E+7
3.0	6.43	1.52E+7	3.95E+7
6.43	20.0	1.31E+6	3.46E+6
Totals		7.24E+7	1.88E+8

Note:
These source terms were calculated for a 144 inch active fuel length. The actual active fuel length is 122 inches.

Table 5.2.23

MINIMUM ENRICHMENTS AS A FUNCTION OF BURNUP FOR THE SHIELDING ANALYSIS

Minimum Enrichment (wt.% ^{235}U)	Maximum Burnup Analyzed (MWD/MTU)	
PWR assemblies with non-zircaloy incore spacers	*MPC-24*	*MPC-32*
2.3	*24,500*	*24,500*
2.6	*29,500*	*29,500*
2.9	*34,500*	*34,500*
3.2	*39,500*	*39,500*
3.4	*44,500*	*42,500*
PWR assemblies with zircaloy incore spacers	*MPC-24*	*MPC-32*
2.3	*24,500*	*24,500*
2.6	*29,500*	*29,500*
2.9	*34,500*	*34,500*
3.2	*39,500*	*39,500*
3.4	*44,500*	*44,500*
MPC-68		
2.1	24,500	
2.4	29,500	
2.6	34,500	
2.9	39,500	
3.0	*44,500*	

Table 5.2.24

DESCRIPTION OF EVALUATED INTACT ZIRCALOY CLAD PWR FUEL

Assembly class	WE 14×14	WE 15×15	WE 17×17	CE 14×14	CE 16×16	B&W 15×15	B&W 17×17
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 5.2.25

DESCRIPTION OF EVALUATED INTACT 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	63	74	92
Rod pitch (in.)	0.738	0.640	0.566	0.510
Cladding material	Zr-2	Zr-2	Zr-2	Zr-2
Rod diameter (in.)	0.570	0.493	0.440	0.404
Cladding thickness (in.)	0.0355	0.0340	0.0280	0.0260
Pellet diameter (in.)	0.488	0.416	0.376	0.345
Pellet material	UO_2	UO_2	UO_2	UO_2
Pellet density (gm/cc) (95% of theoretical)	10.412	10.412	10.412	10.412
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	210.385	201.881	211.307
Weight of U (kg)†	198.516	185.475	177.978	186.288
No. of Water Rods	0	1	2	2
Water Rod O.D. (in.)	n/a	0.493	0.980	0.980
Water Rod Thickness (in.)	n/a	0.0340	0.0300	0.0300

† Derived from parameters in this table.

Table 5.2.26

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

Assembly class	WE 14×14	WE 15×15	WE 17×17	CE 14×14	CE 16×16	B&W 15×15	B&W 17×17
Neutrons/sec	2.29E+8 / 2.28E+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/ 3.32E+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 / 934.9	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 5.2.27

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

Assembly class	7×7	8×8	9×9	10×10
Neutrons/sec	1.33E+8	1.17E+8	1.11E+8	1.22E+8
Photons/sec (0.45-3.0 MeV)	1.55E+15	1.44E+15	1.38E+15	1.46E+15
Thermal power (watts)	435.5	402.3	385.3	407.4

Table 5.2.28

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 5.2.27, 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 5.2.29
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 5.2.30

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

Lower Energy	**Upper Energy**	**16,000 MWD/MTIHM 18-Year Cooling**	
(MeV)	(MeV)	(MeV/s)	(Photons/s)
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 5.2.31

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

Table 5.2.32

DESCRIPTION OF DESIGN BASIS TROJAN FUEL

	PWR
Assembly type/class	*WE 17×17*
Active fuel length (in.)	*144*
No. of fuel rods	*264*
Rod pitch (in.)	*0.496*
Cladding material	*zircaloy-4*
Rod diameter (in.)	*0.374*
Cladding thickness (in.)	*0.0225*
Pellet diameter (in.)	*0.3225*
Pellet material	UO_2
Pellet density (gm/cc)	*10.412 (95% of theoretical)*
Enrichment (w/o ^{235}U)	*3.09*
Burnup (MWD/MTU)	*42,000*
Cooling Time (years)	*16*
Specific power (MW/MTU)	*40*
Weight of UO_2 (kg)†	*529.85*
Weight of U (kg)†	*467.11*
No. of Water Rods/Guide Tubes	*25*
Water Rod O.D. (in.)	*0.482*
Water Rod Thickness (in.)	*0.016*
Lower End Fitting (kg)	*5.9 (steel)*
Gas Plenum Springs (kg)	*1.15 (steel)*
Gas Plenum Spacer (kg)	*0.84 (steel) 0.79 (inconel)*
Upper End Fitting (kg)	*6.89 (steel) 0.96 (inconel)*
Incore Grid Spacers (kg)	*4.9 (inconel)*

† Derived from parameters in this table.

Table 5.2.33
CALCULATED TROJAN PWR FUEL GAMMA SOURCE PER ASSEMBLY

Lower Energy	***Upper Energy***	***42,000 MWD/MTU 16 Year Cooling***	
(MeV)	*(MeV)*	*(MeV/s)*	*(Photons/s)*
0.45	*0.7*	*9.44E+14*	*1.64E+15*
0.7	*1.0*	*3.82E+13*	*4.50E+13*
1.0	*1.5*	*3.09E+13*	*2.47E+13*
1.5	*2.0*	*1.75E+12*	*9.99E+11*
2.0	*2.5*	*9.33E+09*	*4.15E+09*
2.5	*3.0*	*7.47E+08*	*2.72E+08*
Totals		*1.01E+15*	*1.71E+15*

Table 5.2.34
CALCULATED TROJAN FUEL ^{60}Co SOURCE PER ASSEMBLY

Location	***42,000 MWD/MTU 16 Year Cooling (curies)***
Lower End Fitting	*24.19*
Gas Plenum Springs	*4.72*
Gas Plenum Spacer	*18.02*
Grid Spacers	*472.12*
Upper End Fitting	*20.60*

Table 5.2.35
CALCULATED TROJAN FUEL NEUTRON SOURCE PER ASSEMBLY

Lower Energy (MeV)	***Upper Energy (MeV)***	***42,000 MWD/MTU 16 Year Cooling (Neutrons/s)***
1.0E-01	*4.0E-01*	*9.55E+06*
4.0E-01	*9.0E-01*	*4.88E+07*
9.0E-01	*1.4*	*4.47E+07*
1.4	*1.85*	*3.31E+07*
1.85	*3.0*	*5.88E+07*
3.0	*6.43*	*5.30E+07*
6.43	*20.0*	*4.67E+06*
Total		*2.53E+08*

Table 5.2.36

DESCRIPTION OF TROJAN BURNABLE POISON ROD ASSEMBLY AND THIMBLE PLUG DEVICE

Region	***BPRA***	***TPD***
Upper End Fitting (kg of steel)	*2.62*	*2.31*
Upper End Fitting (kg of inconel)	*0.42*	*0.42*
Gas Plenum Spacer (kg of steel)	*0.72*	*1.6*
Gas Plenum Springs (kg of steel)	*0.73*	*1.6*
In-core (kg of steel)	*12.10*	*N/A*

Table 5.2.37

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

Region	***BPRA***	***TPD***
Upper End Fitting (curies Co-60)	*1.20*	*18.86*
Gas Plenum Spacer (curies Co-60)	*0.34*	*12.80*
Gas Plenum Springs (curies Co-60)	*0.34*	*12.80*
In-core (curies Co-60)	*28.68*	*N/A*

Table 5.2.38

DESCRIPTION OF TROJAN ROD CLUSTER CONTROL ASSEMBLY FOR SOURCE TERM CALCULATIONS

Physical Description

Axial Dimensions Relative to Bottom of Active Fuel			***Flux Weighting Factor***	***Mass of cladding (kg Steel)***	***Mass of absorber (kg AgInCd)***
Start (in)	***Finish (in)***	***Length (in)***			
Configuration - Fully Removed					
0.0	*8.358*	*8.358*	*0.2*	*0.76*	*3.18*
8.358	*12.028*	*3.67*	*0.1*	*0.34*	*1.40*

Radiological Description

Axial Dimensions Relative to Bottom of Active Fuel			***Photons/sec from AgInCd***			***Curies Co-60 from Steel***
Start (in)	***Finish (in)***	***Length (in)***	***0.3-0.45 MeV***	***0.45-0.7 MeV***	***0.7-1.0 MeV***	
Configuration - Fully Removed						
0.0	*8.358*	*8.358*	*7.66E+12*	*7.12E+12*	*5.66E+12*	*7.34*
8.358	*12.028*	*3.67*	*1.68E+12*	*1.56E+13*	*1.24E+12*	*1.61*

5.3 MODEL SPECIFICATIONS

The shielding analysis of the HI-STAR 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-STAR 100 System in the shielding analysis. A sample input file for MCNP is provided in Appendix 5.C.

Subsection 5.1.2 discussed the accident conditions and stated that the only accident that would impact the shielding analysis would be a loss of the neutron shield and impact limiters. Therefore, the MCNP models of the HI-STAR 100 System normal condition have the neutron shield and impact limiters in place while the hypothetical accident condition replaces the neutron shield with void and removes the impact limiters. The aluminum honeycomb in the impact limiters was conservatively neglected in the MCNP modeling. However, credit was taken for the outer dimensions of the impact limiters.

5.3.1 Description of the Radial and Axial Shielding Configuration

Section 1.4 provides the ~~Design D~~*d*rawings that describe the HI-STAR 100 System. These drawings were used to create the MCNP models used in the radiation transport calculations. Figures *5.3.1* ~~5.3.2 and~~*through* 5.3.3 show cross sectional views of the HI-STAR 100 overpack and MPC as it was modeled in MCNP for each of the MPCs. These figures were created with the MCNP two-dimensional plotter and are drawn to scale. The figures clearly illustrate the radial steel fins and pocket trunnions in the neutron shield region. 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 Subsection 5.4.1, the dose effect of localized streaming through these compartments is analyzed. Figures 5.3.~~5~~*-4* ~~and~~ *through* 5.3.6 show the MCNP models of the *MPC-32,* MPC-24, and MPC-68 fuel baskets including the as-modeled dimensions. Figure 5.3.9 shows a cross sectional view of the HI-STAR 100 overpack with the as-modeled thickness of the various materials. Figure 5.3.10 is an axial representation of the HI-STAR 100 overpack with the various as-modeled dimensions indicated. As Figure 5.3.10 indicates, the thickness of the MPC~~-68~~ lid ~~and the thickness of the MPC-24 lid are 10.0 and~~ *is* 9.5 inches~~, respectively~~. *Earlier versions of the MPC-68 used a 10 inch thick lid with a correspondingly smaller MPC-internal cavity height. The analysis in this chapter conservatively represents the 9.5 inch thick lid.* ~~Correspondingly, the MPC internal cavity heights differ by 0.5 inch. In the MCNP models of the MPC-24 and MPC-68, the actual lid thickness and internal cavity height for that particular MPC was used.~~ Figures 5.3.11 and 5.3.12 provide the as-modeled dimensions of the impact limiters during normal conditions. The aluminum honeycomb material in the impact limiter is not shown in Figure 5.3.11 because it was conservatively not modeled in the MCNP calculations.

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.

Several conservative approximations were made in modeling the MPC and overpack. 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. The fuel spacers are not needed on all fuel assembly types. However, most PWR fuel assemblies will have upper and lower fuel spacers. The positioning of the fuel assembly for the shielding analysis is determined by the fuel spacer length for the design basis fuel assembly type, but the fuel spacer materials are not modeled. This is conservative since it removes steel which would provide a small amount of additional shielding.

3. For the MPC-24, *MPC-32,* and the MPC-68, the MPC basket supports are not modeled. This is conservative since it removes steel which would provide a small increase in shielding. The *optional* aluminum heat conduction elements were also conservatively not modeled.

4. The MPC-24 basket is fabricated from 5/16 inch thick cell plates~~ and 9/32 inch thick angles~~. 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 which 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 modeling of the impact limiters, only the neutron shield (Holtite-A) and the steel, shown in Figure 5.1.1, were represented. Conservatively, the aluminum honeycomb of the impact limiters was not modeled. However, credit was taken for the outer boundary of the impact limiter as the external surface of the HI-STAR 100 System.

7. In the MPC-24, ~~12 of the 24 Boral panels on the periphery have a reduced width.~~ ~~C~~conservatively, all Boral panels on the periphery were modeled with a reduced width of 5 inches *compared to 6.25 inches.*

8. *The Trojan MPC-24E was modeled explicitly with its shorter cavity length and larger cell sizes with shorter height on the four corner locations. The Trojan MPC was properly positioned in the bottom portion of the HI-STAR and the spacer device between the top of the MPC and the underside of the HI-STAR lid was conservatively not modeled.*

During this project several design changes occurred that affected the drawings, but did not significantly affect the MCNP models of the HI-STAR 100 overpack or MPC. Therefore, in some cases, these models do not exactly represent the drawings. The discrepancies between models and drawings are listed and discussed here.

MPC Modeling Discrepancies

1. *In the newer 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 the HI-STAR outside the MPC baseplate, this localized reduction in shielding will not affect the calculated dose rates outside the HI-STAR.*
2. *The design configuration of the MPC-24 has been enhanced for criticality purposes. The general location of the 24 assemblies remains 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 5.3.13 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.*

5.3.1.1 Fuel Configuration

As described above, 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*s* 5.2.1 *and 5.2.32*. The axial description of the design basis fuel assemblies is provided in Table 5.3.1. *The axial description of the Trojan fuel assembly is provided in Table 5.3.4.* Figures 5.3.7 and 5.3.8 graphically depict the location of the PWR and BWR fuel assemblies within the HI-STAR 100 System. The impact limiters are not depicted in the figures for clarity. The axial locations of the Boral, basket, pocket trunnion, and transition areas are shown in these figures.

The axial position of the fuel assembly within the basket is maintained with the use of the upper and lower fuel spacers. These fuel spacers are used to position the active fuel region next to the Boral. Chapter 2 demonstrates that these fuel spacers do not fail under all normal and hypothetical accident conditions. Therefore, movement of the fuel assembly during transport is not considered.

5.3.1.2 Streaming Considerations

The streaming from the radial channels and pocket trunnions in the neutron shield is evaluated in Subsection 5.4.1. The MCNP model of the HI-STAR 100 completely describes the radial channels and pocket trunnions, thereby properly accounting for the streaming effect. *In newer designs of the HI-STAR 100 overpack, the pocket trunnion has been removed. However, the analysis presented in this chapter using the pocket trunnion bounds the new configuration due to the increased streaming through the pocket trunnion.*

The design of the HI-STAR 100 System, as described in the ~~Design D~~*d*rawings in Section 1.4, 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 overpack top flange. No credit is taken for any part of the trunnion that extends outside of the overpack.

- The pocket trunnions are modeled as solid blocks of steel. The pocket trunnion will be filled with a solid steel rotation trunnion attached to the transport frame during handling and shipping *or a plug will be installed if rotation trunnions are not inserted into the pocket trunnion.*

- The threaded holes in the MPC lid are plugged with solid plugs during shipping 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 steel lost in the MPC lid at the port location is replaced with a block of steel approximately 6 inches thick below the port opening and attached to the underside of the lid. This design feature is shown on the ~~Design D~~*d*rawings in Section 1.4. The MCNP model did not explicitly represent this arrangement but, rather, modeled the MPC lid as a solid piece.

- The penetrations in the overpack are filled with bolts that extend into the penetration, thereby eliminating any potential direct streaming paths. Cover plates are also designed in such a way as to maintain the thickness of the overpack to the maximum extent practical. Therefore, the MCNP model does not represent any streaming paths due to penetrations in the overpack.

5.3.2 Regional Densities

Composition and densities of the various materials used in the HI-STAR 100 System 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. All steel in the MPC was modeled as stainless steel and all steel in the overpack was modeled as carbon steel.

Section 3.4 demonstrates that all materials used in the HI-STAR 100 System remain below their design temperatures as specified in Table 2.1.2 during all normal conditions. Therefore, the shielding analysis does not address changes in the material density or composition as a result of temperature changes.

During normal operations, the depletion of B-10 in the Boral and the Holtite-A neutron shield is negligible. The fraction of B-10 atoms that are depleted in 50 years is approximately 3.0E-9 and 4.0E-8 in the Boral and Holtite-A, respectively. Therefore, the shielding analysis does not address changes in the composition of the Boral or Holtite-A as a result of neutron absorption.

As discussed in Section 1.2.1.4.2, the density of the Holtite-A during normal condition was reduced by approximately 4% to account for any potential water loss. In addition, the Hydrogen weight percent was conservatively reduced from 6% to 5.92%.

Section 3.5 discusses the effect of the hypothetical accident condition (fire) on the temperatures of the shielding materials and the resultant impact on their shielding effectiveness. As stated in Subsection 5.1.2, the only consequence that has any significant impact on the shielding configuration is the loss of the neutron shield in the HI-STAR 100 System as a result of fire. The change in the neutron shield was conservatively analyzed by assuming that the entire volume of the neutron shield was replaced by void.

Table 5.3.1

DESCRIPTION OF THE AXIAL MCNP MODEL OF THE *DESIGN BASIS* 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-STAR 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) 4.367 (MPC-24)
		^{11}B	20.1474 (MPC-68) 19.893 (MPC-24)
		Al	68.61 (MPC-68) 69.01 (MPC-24)
		C	6.82 (MPC-68) 6.73 (MPC-24)
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

Table 5.3.2 (continued)

COMPOSITION OF THE MATERIALS IN THE HI-STAR 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	16.701
PWR Fuel Region Mixture	3.853705	^{235}U	2.6944
		^{238}U	70.1276
		O	9.7895
		Zr	17.3885

Table 5.3.2 (continued)

COMPOSITION OF THE MATERIALS IN THE HI-STAR 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

Table 5.3.3

COMPOSITION OF THE FUEL IN THE MIXED OXIDE FUEL ASSEMBLIES IN THE MPC-68 OF THE HI-STAR 100 SYSTEM

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

Table 5.3.4

DESCRIPTION OF THE AXIAL MCNP MODEL OF THE TROJAN FUEL ASSEMBLY†

Region	***Start (in.)***	***Finish (in.)***	***Length (in.)***	***Actual Material***	***Modeled Material***
		PWR			
Lower End Fitting	*0.0*	*2.738*	*2.738*	*SS304*	*SS304*
Space	*2.738*	*3.738*	*1.0*	*zircaloy*	*Void*
Fuel	*3.738*	*147.738*	*144*	*fuel & zircaloy*	*Fuel*
Gas Plenum Springs	*147.738*	*151.916*	*4.178*	*SS304 & zircaloy*	*SS304*
Gas Plenum Spacer	*151.916*	*156.095*	*4.179*	*SS304 & zircaloy*	*SS304*
Upper End Fitting	*156.095*	*159.765*	*3.67*	*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.

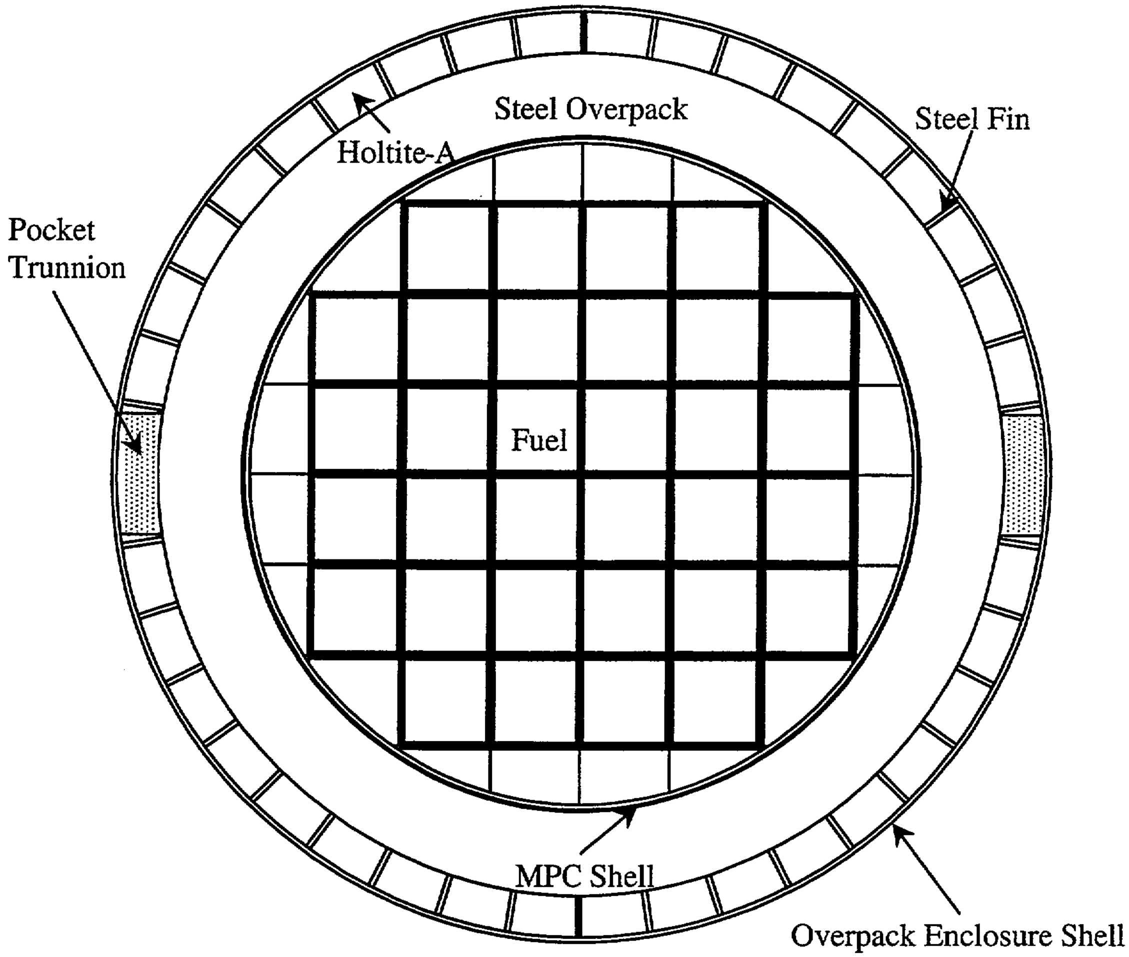

FIGURE 5.3.1; HI-STAR 100 OVERPACK WITH MPC-32 CROSS SECTIONAL VIEW AS MODELLED IN MCNP†

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

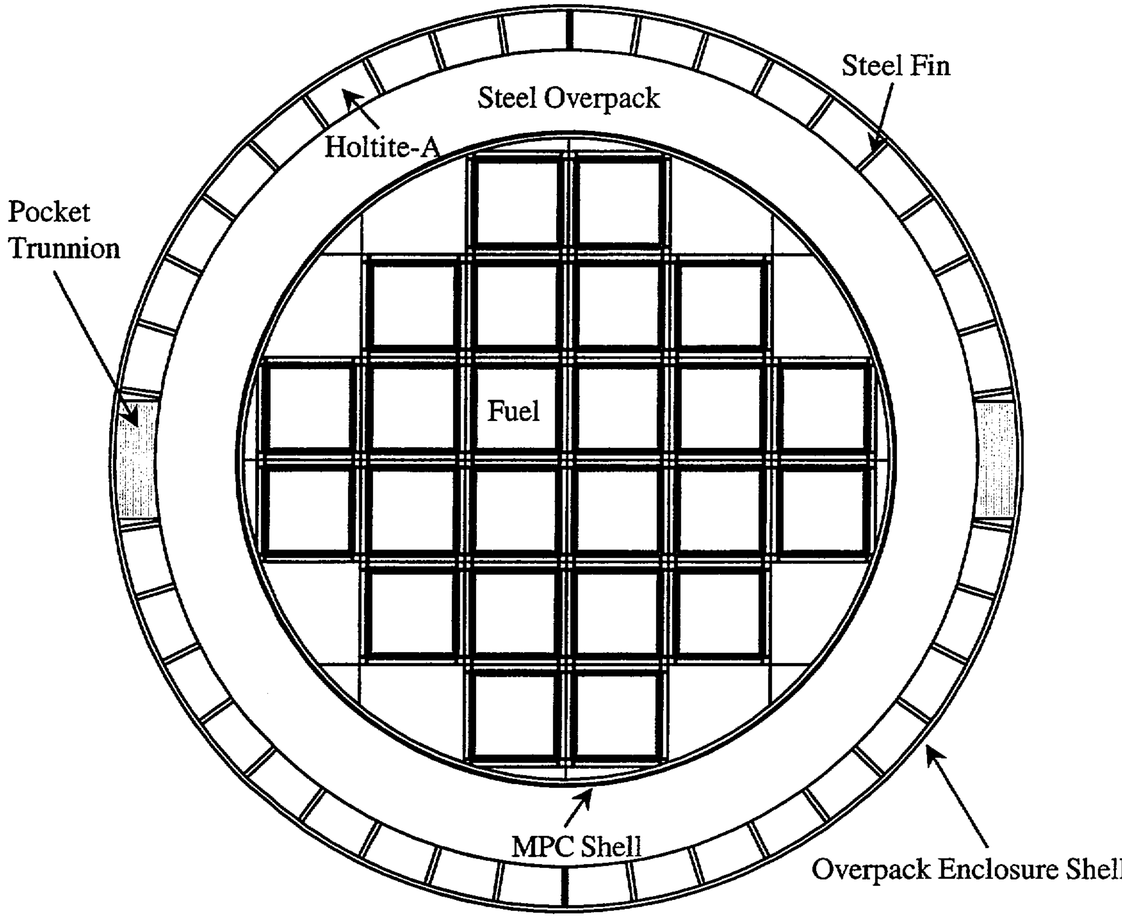

FIGURE 5.3.2; HI-STAR 100 OVERPACK WITH MPC-24 CROSS SECTIONAL VIEW AS MODELLED IN MCNP†

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

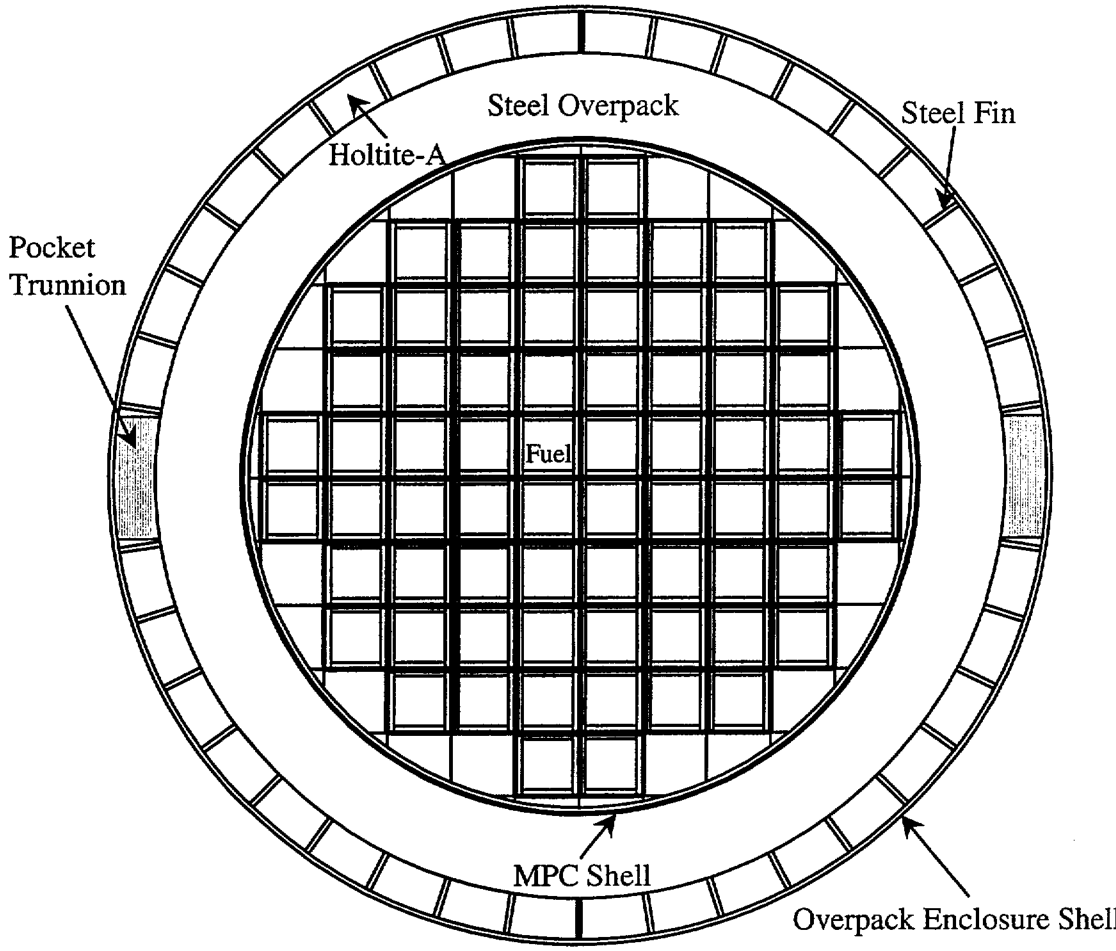

FIGURE 5.3.3; HI-STAR 100 OVERPACK WITH MPC-68 CROSS SECTIONAL VIEW AS MODELLED IN MCNP†

† 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

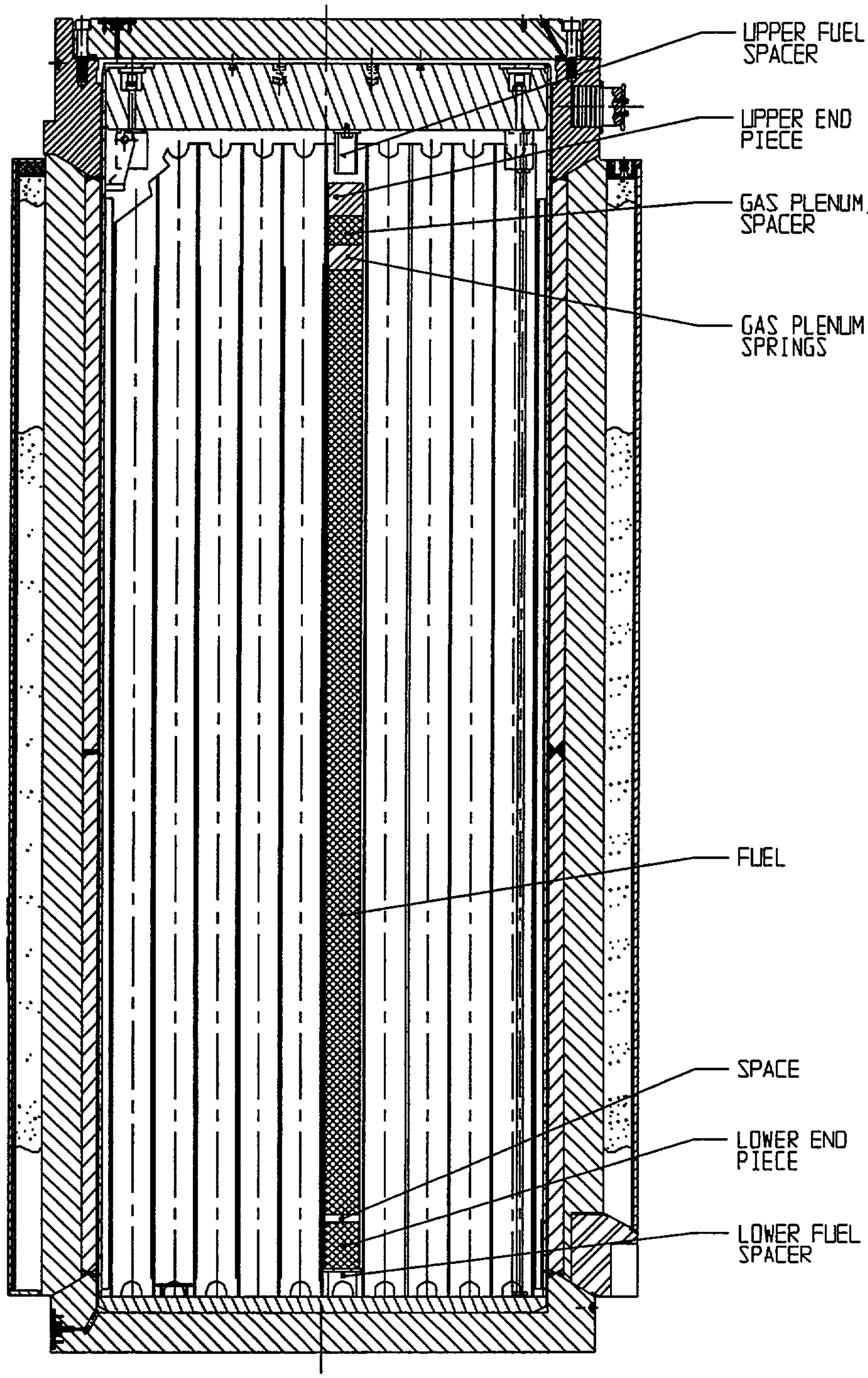

FIGURE 5.3.7; AXIAL LOCATION OF PWR DESIGN BASIS FUEL IN THE HI-STAR 100 SYSTEM

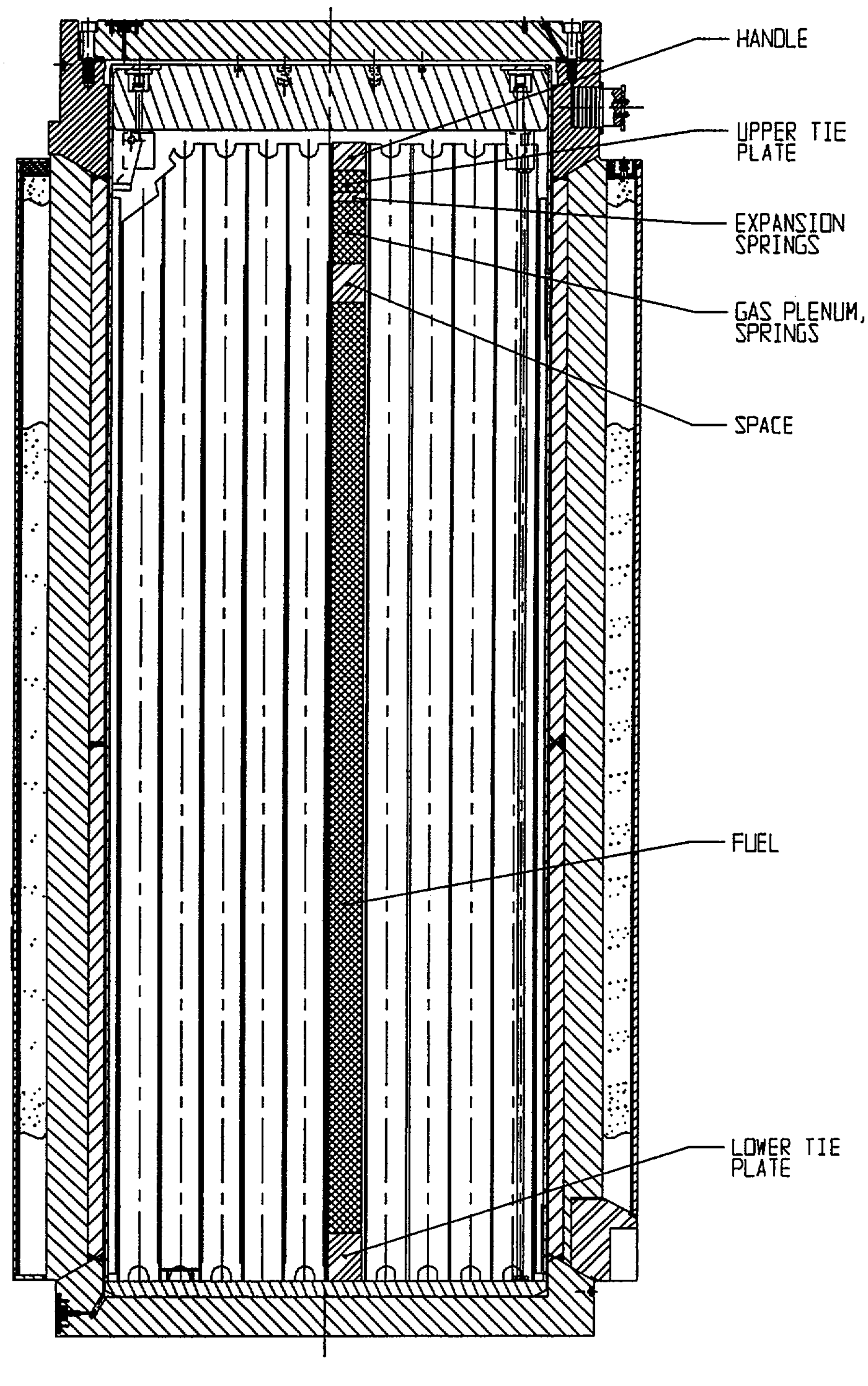

FIGURE 5.3.8; AXIAL LOCATION OF BWR DESIGN BASIS FUEL IN THE HI-STAR 100 SYSTEM

FIGURE WITHHELD AS SENSITIVE UNCLASSIFIED INFORMATION

FIGURE 5.3.9; HI-STAR 100 OVERPACK WITH MPC-24 CROSS SECTIONAL VIEW SHOWING THE THICKNESS OF THE MPC SHELL AND OVERPACK AS MODELED IN MCNP

FIGURE WITHHELD AS SENSITIVE UNCLASSIFIED INFORMATION

FIGURE 5.3.10; AXIAL VIEW OF HI-STAR 100 OVERPACK **AND** MPC WITH AXIAL DIMENSIONS SHOWN AS MODELED IN MCNP

FIGURE WITHHELD AS SENSITIVE UNCLASSIFIED INFORMATION

FIGURE 5.3.11; CROSS SECTION ELEVATION VIEW OF THE HI-STAR 100 SYSTEM WITH AS MODELED DIMENSIONS FOR THE IMPACT LIMITER

FIGURE WITHHELD UNDER 10 CFR 2.390

FIGURE 5.3.12; CROSS SECTIONAL VIEW OF IMPACT LIMITER WITH AS MODELED DIMENSIONS

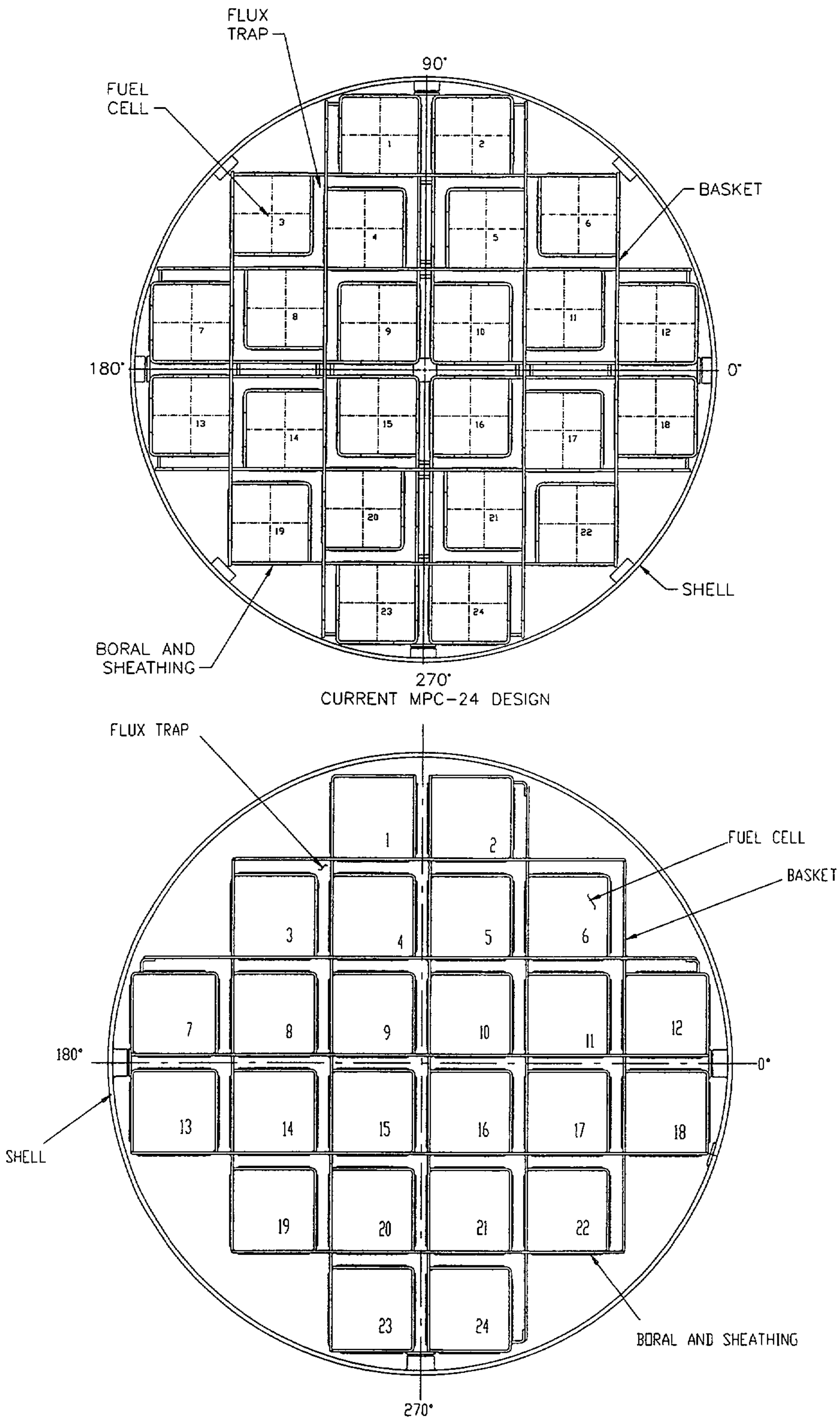

FIGURE 5.3.13: CROSS SECTIONAL VIEWS OF THE CURRENT MPC-24 DESIGN AND THE SUPERSEDED MPC-24 WHICH IS USED IN THE MCNP MODELS.

5.4 SHIELDING EVALUATION

The MCNP-4A code[5.1.1] was used for all of the shielding analyses. MCNP is a continuous energy, three-dimensional, coupled neutron-photon-electron Monte Carlo transport code. Continuous energy cross section data is represented with sufficient energy points to permit linear-linear interpolation between these points. The individual cross section libraries used for each nuclide are those recommended by the MCNP manual. All of these data are based on ENDF/B-V data. MCNP has been extensively benchmarked against experimental data by the large user community. References [5.4.2], [5.4.3], and [5.4.4] are three examples of the benchmarking that has been performed.

The energy distribution of the source term, as described earlier, is used explicitly in the MCNP model. A different MCNP calculation is performed for each of the three source terms (neutron, decay gamma, and ^{60}Co). The axial distribution of the fuel source term is described in Table 1.2.15 and Figures 1.2.13 and 1.2.14. The PWR and BWR axial burnup distributions were obtained from References [5.4.5] and [5.4.6] respectively. These axial distributions were obtained from operating plants and are representative of PWR and BWR fuel with burnups greater than 30,000 MWD/MTU. The ^{60}Co source in the hardware was assumed to be uniformly distributed over the appropriate regions. *The axial distribution used for the Trojan Plant fuel was similar but not identical to the PWR distribution in Table 1.2.15 and can be found in the Trojan FSAR [5.1.6].*

It has been shown that the neutron source strength varies as the burnup level raised by the power of 4.2. Since this relationship is non-linear and since the burnup in the axial center of a fuel assembly is greater than the average burnup, the neutron source strength in the axial center of the assembly is greater than the relative burnup times the average neutron source strength. In order to account for this effect, the neutron source strength in each of the 10 axial nodes listed in Table 1.2.15 was determined by multiplying the average source strength by the relative burnup level raised to the power of 4.2. The peak relative burnups listed in Table 1.2.15 for the PWR and BWR fuels are 1.105 and 1.195 respectively. Using the power of 4.2 relationship results in a 37.6% ($1.105^{4.2}/1.105$) and 76.8% ($1.195^{4.2}/1.195$) increase in the neutron source strength in the peak nodes for the PWR and BWR fuel respectively. The total neutron source strength increases by 15.6% for the PWR fuel assemblies and 36.9% for the BWR fuel assemblies.

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

Figures 5.1.1 and 5.1.2 depict the dose point locations during normal and hypothetical accident conditions of transport. Dose point location 3*a* in Figure 5.1.1 covers two regions of different radii. The outermost region is 5.75 inches in height and the innermost region is 6.875 inches in height. The dose rate was calculated over both segments and the highest value was reported for dose location 3*a*. Dose point locations 1 through 4 in Figure 5.1.2 are conservatively locat*ed*~~ion~~ at a radial position that is approximately 1 meter from the outer radial surface of the bottom plate.

Tables 5.4.2 through 5.4.*6*~~7 and 5.4.16 through 5.4.18~~ list the *maximum* dose rates ~~(from each of the three radiation sources)~~ on the surface of the HI-STAR 100 System for *the allowable burnup and cooling time combinations listed in Appendix A to the CoC for the MPC-24, MPC-32, and MPC-68. As discussed in Section 5.1, the personnel barrier is credited in the shielding analysis as an enclosure for the HI-STAR 100 overpack. Therefore, the dose limit for dose location 2a and 3a in Figure 5.1.1 is higher than the other dose locations. The dose rates for the 2 meter location and the accident scenario are provided in Section 5.l.* ~~each of the burnup levels and cooling times evaluated for the MPC-24 and the MPC-68. Tables 5.4.8, 5.4.9, and 5.4.19 provide the total dose rate on the surface of the HI-STAR 100 System for each burnup level and cooling time. Tables 5.4.10 through 5.4.13 and 5.4.20 and 5.4.21 provide the total dose rate at 2 meters for normal conditions and at 1 meter for accident conditions for each burnup level and cooling time for the MPC-24 and the MPC-68. This information was used to determine the worst case burnup level and cooling time and corresponding maximum dose rates reported in Section 5.1.~~

Since MCNP is a statistical code, there is an uncertainty associated with the calculated values. In MCNP the uncertainty is expressed as the relative error which is defined as the standard deviation of the mean divided by the mean. Therefore, the standard deviation is represented as a percentage of the mean. The relative error for the total dose rates presented in this chapter were typically less than 2% and the relative error for the individual dose components was typically less than 5%.

5.4.1 Streaming Through Radial Steel Fins and Pocket Trunnions

The HI-STAR 100 overpack utilizes 0.5 inch thick radial channels for structural support and cooling. The attenuation of neutrons through steel is substantially less than the attenuation of neutrons through the neutron shield. Therefore, it is possible to have neutron streaming through the channels which could result in a localized dose peak. The reverse is true for photons which would result in a localized reduction in the photon dose. Analyses were performed to determine the magnitude of the dose peaks and depressions and the impact on localized dose as compared to average total dose. This effect was evaluated at the radial surface of the HI-STAR 100 System and a distance of two meters.

In addition to the radial channels, the pocket trunnions are essentially blocks of steel that are approximately 12 inches wide and 12 inches high. The effect of the pocket trunnion on neutron streaming and photon transmission will be more substantial than the effect of a single fin.

Therefore, analyses were performed to quantity this effect. Figures 5.3.7 and 5.3.8 illustrate the location of the pocket trunnion and its axial position relative to the active fuel.

The fuel loading pattern in the *MPC-32,* MPC-24 and the MPC-68, as depicted in Figures 5.3.~~2~~*1* ~~and~~ *through* 5.3.3, is not cylindrical. Therefore, there is a potential to experience peaking as a result of azimuthal variations in the fuel loading. Since the MCNP models represent the fuel in the correct positions (i.e., cylindrical homogenization is not performed) the effect of azimuthal variations in the loading pattern is automatically accounted for in the calculations that are discussed below.

The effect of streaming through the pocket trunnion and the radial channels was analyzed using the full three-dimensional MCNP models of the MPC-24 and the MPC-68. The effect of peaking was calculated on the surface of the overpack adjacent to the pocket trunnion and dose locations 2*a* and 3*a* in Figures 5.1.1. The effect of peaking was also analyzed at 2 meters from the overpack at dose location 2 and at the axial height of the impact limiter. Dose location 3 was not analyzed at two meters because the dose at that point is less than the dose at location 2 as demonstrated in the tables ~~at the end of this section~~*in Section 5.1.* Figure 5.4.1 shows a quarter of the HI-STAR 100 overpack with 41 azimuthal bins drawn. There is one bin per steel fin and 3 bins in each neutron shield region. This azimuthal binning structure was used over the axial height of the overpack. The dose was calculated in each of these bins and then compared to the average dose calculated over the surface to determine a peak-to-average ratio for the dose in that bin. The azimuthal location of the pocket trunnion is shown in Figure 5.4.1. The pocket trunnion was modeled as solid steel. During shipping, a steel rotation trunnion *or plug* shall be placed in the pocket trunnion recess. To conservatively evaluate the peak to average ratio, the pocket trunnion is assumed to be solid steel.

Table 5.4.14 provides the peak-to-average ratios that were calculated for the various dose components and locations. Table 5.4.15 presents the dose rates at the dose locations analyzed. The peak dose on the surface of the overpack at dose location 2 occurs at a steel channel (fin). This is evident by the high neutron peaking at dose location 2 on the surface of the overpack. The dose rate at the pocket trunnion is higher than the dose rate at dose location 2 on the surface of the overpack. However, these results clearly indicate that, at two meters, the peaking associated with the pocket trunnion is not present and that the peak dose location is #2.

The MPC-32 was not explicitly analyzed for azimuthal peaking. It is expected that the peaking in the MPC-24 will be similar if not larger than in the MPC-32 due to the fact that the fuel assemblies in the MPC-24 are not as closely positioned as in the MPC-32.

5.4.2 Damaged Fuel Post-Accident Shielding Evaluation

As discussed in Subsection 5.2.5.2, the analysis presented below, even though it is for damaged fuel, demonstrates the acceptability of transporting intact Humboldt Bay 6x6 and intact Dresden

1 6x6 fuel assemblies. *As discussed in Subsection 5.2.8, the Trojan damaged fuel and fuel debris were not explicitly analyzed because they are bounded by the intact fuel assemblies.*

For the damaged fuel and fuel debris accident condition, it is conservatively assumed the damaged fuel cladding ruptures and all the fuel pellets fall and collect at the bottom of the damaged fuel container. The inner dimension of the damaged fuel container, specified in the Design Drawings of Section 1.4, and the design basis damaged fuel and fuel debris assembly dimensions in Table 5.2.2 are used to calculate the axial height of the rubble in the damaged fuel container assuming 50% compaction. Neglecting the fuel pellet to cladding inner diameter gap, the volume of cladding and fuel pellets available for deposit is calculated assuming the fuel rods are solid. Using the volume in conjunction with the damaged fuel container, the axial height of rubble is calculated to be 80 inches.

Some of the 6x6 assemblies described in Table 5.2.2 were manufactured with Inconel grid spacers (the mass of inconel is listed in Table 5.2.2). The calculated ^{60}Co activity from these spacers was 66.7 curies for a burnup of 30,000 MWD/MTU and a cooling time of 18 years. Including this source with the total fuel gamma source for damaged fuel in Table 5.2.6 and dividing by the 80 inch rubble height provides a gamma source per inch of 3.47E+12 photon/s. Dividing the total neutron source for damaged fuel in Table 5.2.14 by 80 inches provides a neutron source per inch of 3.93E+5 neutron/s. These values are both bounded by the BWR design basis fuel gamma source ~~(including the ^{60}Co activity from incore spacers)~~ per inch and neutron source per inch values of ~~4.87~~*5.03*E+12 photon/s and 6.~~38E~~*63E*+5 neutron/s. These BWR design basis values were calculated by dividing the total source strengths as calculated from Tables 5.2.5~~, 5.2.10~~ and 5.2.13 (39,500 MWD/MTU and ~~15~~ *14* year cooling values) by the active fuel length of 144 inches. Therefore, the design basis damaged fuel assembly is bounded by the design basis intact BWR fuel assembly for accident conditions. No explicit analysis of the damaged fuel dose rates are provided as they are bounded by the intact fuel analysis.

5.4.3 Mixed Oxide Fuel Evaluation

The source terms calculated for the Dresden Unit 1 GE 6x6 MOX fuel assemblies can be compared to the design basis source terms for the BWR assemblies which demonstrates that the MOX fuel source terms are bounded by the design basis source terms and no additional shielding analysis is needed.

Since the active fuel length of the MOX fuel assemblies is shorter than the active fuel length of the design basis fuel, the source terms must be compared on a per inch basis. Including the ^{60}Co source from grid spacers as calculated in the previous subsection (66.7 curies) with the total fuel gamma source for the MOX fuel in Table 5.2.16 and dividing by the 110 inch active fuel height provides a gamma source per inch of 2.41E+12 photons/s. Dividing the total neutron source for the MOX fuel assemblies in Table 5.2.17 by 110 inches provides a neutron source strength per inch of 3.67E+5 neutrons/s. These values are both bounded by the BWR design basis fuel gamma source per inch and neutron source per inch values of ~~4.87~~*5.03*E+12 photons/s and 6.~~38E~~*63E*+5 neutrons/s. These BWR design basis values were calculated by dividing the total source strengths as calculated from Tables 5.2.5~~, 5.2.10~~ and 5.2.13 (39,500 MWD/MTU and ~~15~~ *14* year cooling values) by the active fuel length of 144 inches. This comparison shows that the MOX fuel source terms are bound by the design basis source terms. Therefore, no explicit analysis of dose rates is provided for MOX fuel.

Since the MOX fuel assemblies are Dresden Unit 1 6x6 assemblies, they can also be considered as damaged fuel. Using the same methodology as described in Subsection 5.4.2, the source term for the MOX fuel is calculated on a per inch basis assuming a post-accident rubble height of 80 inches. The resulting gamma and neutron source strengths are 3.31E+12 photons/s and 5.05E+5 neutrons/s. These values are also bounded by the design basis fuel gamma source per inch and neutron source per inch. Therefore, no explicit analysis of dose rates is provided for MOX fuel in a post-accident configuration.

5.4.4 Stainless Steel Clad Fuel Evaluation

Tables 5.4.22 through 5.4.24 present the dose rates from the stainless steel clad fuel at various dose locations around the HI-STAR 100 overpack for the MPC-24 and the MPC-68 for normal and hypothetical accident conditions. These dose rates are below the regulatory limits indicating that these fuel assemblies are acceptable for transport.

As described in Subsection 5.2.3, the source term for the stainless steel fuel was calculated conservatively with an artificial active fuel length of 144 inches. The end fitting masses of the stainless steel clad fuel are also assumed to be identical to the end fitting masses of the zircaloy clad fuel. In addition, the fuel assembly configuration used in the MCNP calculations was identical to the configuration used for the design basis fuel assemblies as described in Table 5.3.1.

5.4.5 Dresden Unit 1 Antimony-Beryllium Neutron Sources

Dresden Unit 1 has antimony-beryllium neutron sources which are placed in the water rod location of their fuel assemblies. These sources are steel rods which contain a cylindrical antimony-beryllium source which is 77.25 inches in length. The steel rod is approximately 95 inches in length. Information obtained from Dresden Unit 1 characterizes these sources in the following manner: "About one-quarter pound of beryllium will be employed as a special neutron source material. The beryllium produces neutrons upon gamma irradiation. The gamma rays for the source at initial start-up will be provided by neutron-activated antimony (about 865 curies). The source strength is approximately 1E+8 neutrons/second."

As stated above, beryllium produces neutrons through gamma irradiation and in this particular case antimony is used as the gamma source. The threshold gamma energy for producing neutrons from beryllium is 1.666 MeV. The outgoing neutron energy increases as the incident gamma energy increases. Sb-124, which decays by Beta decay with a half life of 60.2 days, produces a gamma of energy 1.69 MeV which is just energetic enough to produce a neutron from beryllium. Approximately 54% of the Beta decays for Sb-124 produce gammas with energies greater than or equal to 1.69 MeV. Therefore, the neutron production rate in the neutron source can be specified as 5.8E-6 neutrons per gamma (1E+8/865/3.7E+10/0.54) with energy greater than 1.666 MeV or 1.16E+5 neutrons/curie (1E+8/865) of Sb-124.

With the short half life of 60.2 days all of the initial Sb-124 is decayed and any Sb-124 that was produced while the neutron source was in the reactor is also decayed since these neutron sources are assumed to have the same minimum cooling time as the Dresden 1 fuel assemblies (array classes 6x6A, 6x6B, 6x6C, and 8x8A) of 18 years. Therefore, there are only two possible gamma sources which can produce neutrons from this antimony-beryllium source. The first is the gammas from the decay of fission products in the fuel assemblies in the MPC. The second gamma source is from Sb-124 which is being produced in the MPC from neutron activation from neutrons from the decay of fission products.

MCNP calculations were performed to determine the gamma source as a result of decay gammas from fuel assemblies and Sb-124 activation. The calculations explicitly modeled the 6x6 fuel assembly described in Table 5.2.2. A single fuel rod was removed and replaced by a guide tube. In order to determine the amount of Sb-124 that is being activated from neutrons in the MPC it was necessary to estimate the amount of antimony in the neutron source. The O.D. of the source was assumed to be the I.D. of the steel rod encasing the source (0.345 in.). The length of the source is 77.25 inches. The beryllium is assumed to be annular in shape encompassing the antimony. Using the assumed O.D. of the beryllium and the mass and length, the I.D. of the beryllium was calculated to be 0.24 inches. The antimony is assumed to be a solid cylinder with an O.D. equal to the I.D. of the beryllium. These assumptions are conservative since the antimony and beryllium are probably encased in another material which would reduce the mass of antimony. A larger mass of antimony is conservative since the calculated activity of Sb-124 is directly proportional to the initial mass of antimony.

The number of gammas from fuel assemblies with energies greater than 1.666 MeV entering the 77.25 inch long neutron source was calculated to be 1.04E+8 gammas/sec which would produce a neutron source of 603.2 neutrons/sec (1.04E+8 * 5.8E-6). The steady state amount of Sb-124 activated in the antimony was calculated to be 39.9 curies. This activity level would produce a neutron source of 4.63E+6 neutrons/sec (39.9 * 1.16E+5) or 6.0E+4 neutrons/sec/inch (4.63E+6/77.25). These calculations conservatively neglect the reduction in antimony and beryllium which would have occurred while the neutron sources were in the core and being irradiated at full reactor power.

Since this is a localized source (77.25 inches in length) it is appropriate to compare the neutron source per inch from the design basis Dresden Unit 1 fuel assembly, 6x6, containing an Sb-Be neutron source to the design basis fuel neutron source per inch. This comparison, presented in Table 5.4.25, demonstrates that a Dresden Unit 1 fuel assembly containing an Sb-Be neutron source is bounded by the design basis fuel.

As stated above, the Sb-Be source is encased in a steel rod. Therefore, the gamma source from the activation of the steel was considered assuming a burnup of 120,000 MWD/MTU which is the maximum burnup assuming the Sb-Be source was in the reactor for the entire 18 year life of Dresden Unit 1. The cooling time assumed was 18 years which is the minimum cooling time for Dresden Unit 1 fuel. The source from the steel was bounded by the design basis fuel assembly. In conclusion, transport of a Dresden Unit 1 Sb-Be neutron source in a Dresden Unit 1 fuel assembly is acceptable and bounded by the current analysis.

5.4.6 Thoria Rod Canister

Based on a comparison of the gamma spectra from Tables 5.2.30 and 5.2.6 for the thoria rod canister and design basis 6x6 fuel assembly, respectively, it is difficult to determine if the thoria rods will be bounded by the 6x6 fuel assemblies. However, it is obvious that the neutron spectra from the 6x6, Table 5.2.14, bounds the thoria rod neutron spectra, Table 5.2.31, with a significant margin. In order to demonstrate that the gamma spectrum from the single thoria rod canister is bounded by the gamma spectrum from the design basis 6x6 fuel assembly, the gamma dose rate on the outer radial surface of the overpack was estimated conservatively assuming an MPC full of thoria rod canisters. This gamma dose rate was compared to an estimate of the dose rate from an MPC full of design basis 6x6 fuel assemblies. The gamma dose rate from the 6x6 fuel was higher than the dose rate from an MPC full of thoria rod canisters. This in conjunction with the significant margin in neutron spectrum and the fact that there is only one thoria rod canister clearly demonstrates that the thoria rod canister is acceptable for transport in the MPC-68 or the MPC-68F.

5.4.7 Trojan Fuel Contents

Tables 5.4.7 through 5.4.9 present the results for the Trojan MPC-24E for normal surface and 2 meter as well as accident results. Since the Trojan MPCs will contain BPRAs, RCCAs, and TPDs, the source from these devices was considered in the analysis. The source from BPRAs and TPDs were added to the fuel source in the appropriate location. The mass from these devices was conservatively neglected. Separate calculations were performed for the BPRAs and the TPDs since both devices can not be present in the same fuel assembly. Separate MCNP calculations were performed for the consideration of the RCCAs since this source is localized at the bottom of the MPC. The results presented in Tables 5.4.7 through 5.4.9 represent the configuration (fuel plus non-fuel hardware) that produces the highest dose rate at that location. Separate results for the different non-fuel hardware are not provided.

These dose rates are less than the values reported for the design basis assemblies in the other MPCs and therefore the Trojan contents are approved for transportation.

5.4.8 Trojan Antimony-Beryllium Neutron Sources

The analysis of the Trojan secondary antimony-beryllium neutron sources was performed in a manner very similar to that described above in Subsection 5.4.5. The secondary sources are basically BPRAs with four rods containing the antimony-beryllium with a length of 88 inches. As mentioned in Subsection 5.4.5, the antimony-beryllium source is a regenerative source in which the antimony is activated and the gammas released from the antimony induce a gamma,n reaction in the beryllium.

The steady state production of neutrons from this antimony-beryllium source was conservatively calculated in the MPC using an approach very similar to that described in Subsection 5.4.5. The depletion of antimony from the operation in the reactor core was conservatively neglected in the analysis. MCNP calculations were performed with explicitly modeled fuel assemblies in a Trojan MPC model to calculate the steady state activity of Sb-124 in the antimony-beryllium source due to the neutrons from the spent fuel. This activity level was used in a subsequent MCNP calculation to determine the gamma,n reaction rate in the beryllium assuming a conservatively flat cross section of 1.5E-3 barns for the gamma,n reaction (the cross section exhibits peaks at 1.5E-3 with lows at approximately 0.3E-3 barns). Additionally, the gamma,n reaction rate due to gammas from the spent fuel was determined. Finally, the gamma,n reaction rate was converted to neutrons/sec to yield the neutron source per secondary source assembly. This value was calculated to be 1.9E+8 neutrons/sec compared to a design basis Trojan assembly of 2.53E+8 neutrons/sec. This is a very conservative estimate of the neutrons/sec from the secondary source because it neglects depletion of the antimony that has occurred during core operation, and the average cross section below 5 MeV is approximately 0.6E-3 barns which is a factor of 2.5 lower than the value used. Therefore, based on the magnitude of the neutron production rate from the secondary source and the level of conservatism in the calculation, this source was not considered in the dose rate analysis reported in this chapter.

Two of the four secondary sources operated in the core for up to 11 cycles. The activation of the steel cladding in these sources was not considered in the analysis reported in this chapter because of the limited quantity of these devices. The activation of the upper portion of the secondary source is bounded by the analysis of the TPDs. Considering the limited quantity of these devices (two) and the fact that activated portion of these devices in the fuel region is well shielded by the fuel assembly rods surrounding the thimble plugs, the gamma source from this activated material has been neglected in this analysis.

Table 5.4.1

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

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

Table 5.4.1 (continued)

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

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

Table 5.4.1 (continued)

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

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

† Includes the Quality Factor.

Table 5.4.2

DOSE RATES AT THE SURFACE OF THE HI-STAR 100 SYSTEM FOR NORMAL CONDITIONS MPC-24 WITH DESIGN BASIS ZIRCALOY CLAD FUEL WITH ZIRCALOY INCORE SPACERS AT WORST CASE BURNUP AND COOLING TIME 34,500 MWD/MTU AND 9-YEAR COOLING

Dose Point† Location	Fuel Gammas†† (mrem/hr)	Gammas from Incore Spacers (mrem/hr)	^{60}Co Gammas (mrem/hr)	Neutrons (mrem/hr)	Totals (mrem/hr)	10 CFR 71.47 Limit
2a	15.97	0.00	24.16	5.67	45.79	1000
3a	1.38	0.00	79.89	64.38	145.65	1000
1	2.36	0.00	26.19	12.86	41.41	200
2	10.45	0.00	14.23	4.65	29.33	200
3	1.49	0.00	22.00	13.45	36.95	200
4	0.83	0.00	19.96	12.98	33.78	200
5	0.30	0.00	0.03	2.42	2.75	200
6	4.28	0.00	97.73	19.04	121.05	200

† Refer to Figure 5.1.1.

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

Table 5.4.3

DOSE RATES AT THE SURFACE OF THE HI-STAR 100 SYSTEM FOR NORMAL CONDITIONS
MPC-24 WITH DESIGN BASIS ZIRCALOY CLAD FUEL WITH NON-ZIRCALOY INCORE SPACERS
AT WORST CASE BURNUP AND COOLING TIME
24,500 MWD/MTU AND 10-YEAR COOLING

Dose Point† Location	Fuel Gammas†† (mrem/hr)	Gammas from Incore Spacers (mrem/hr)	^{60}Co Gammas (mrem/hr)	Neutrons (mrem/hr)	Totals (mrem/hr)	10 CFR 71.47 Limit
2a	15.59	21.93	0.05	4.07	41.63	1000
3a	0.71	0.76	57.74	22.50	81.70	1000
1	1.17	1.65	18.93	4.49	26.25	200
2	10.02	13.76	0.13	2.73	26.64	200
3	0.74	1.01	15.90	4.70	22.36	200
4	0.41	0.54	14.42	4.54	19.91	200
5	0.10†††	-	0.02	0.85	0.97	200
6	4.26†††	-	70.63	6.66	81.55	200

† Refer to Figure 5.1.1.

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

††† Gammas from incore spacers are included with fuel gammas.

Table 5.4.4

DOSE RATES AT THE SURFACE OF THE HI-STAR 100 SYSTEM FOR NORMAL CONDITIONS
MPC-68 WITH DESIGN BASIS ZIRCALOY CLAD FUEL
AT WORST CASE BURNUP AND COOLING TIME
24,500 MWD/MTU AND 8-YEAR COOLING

Dose Point† Location	Fuel Gammas†† (mrem/hr)	Gammas from Incore Spacers (mrem/hr)	^{60}Co Gammas (mrem/hr)	Neutrons (mrem/hr)	Totals (mrem/hr)	10 CFR 71.47 Limit
2a	25.76	6.10	0.01	6.57	38.44	1000
3a	0.33	0.10	117.34	14.32	132.09	1000
1	1.91	0.45	26.39	5.48	34.23	200
2	16.26	3.82	0.10	4.03	24.21	200
3	0.57	0.12	28.50	3.06	32.25	200
4	0.30	0.07	27.25	2.90	30.52	200
5	0.08†††	-	0.03	0.60	0.71	200
6	2.30†††	-	90.75	7.93	100.98	200

† Refer to Figure 5.1.1.

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

††† Gammas from incore spacers are included with fuel gammas.

Table 5.4.5

DOSE RATES AT THE SURFACE OF THE HI-STAR 100 SYSTEM FOR NORMAL CONDITIONS MPC-32 WITH DESIGN BASIS ZIRCALOY CLAD FUEL WITH ZIRCALOY INCORE SPACERS AT WORST CASE BURNUP AND COOLING TIME 34,500 MWD/MTU AND 12-YEAR COOLING

Dose Point*† *Location	***Fuel Gammas*†† *(mrem/hr)***	***Gammas from Incore Spacers (mrem/hr)***	***^{60}Co Gammas (mrem/hr)***	***Neutrons (mrem/hr)***	***Totals (mrem/hr)***	***10 CFR 71.47 Limit***
2a	*7.95*	*0.00*	*30.44*	*4.76*	*43.15*	*1000*
3a	*1.50*	*0.00*	*59.18*	*142.32*	*203.00*	*1000*
1	*1.99*	*0.00*	*24.67*	*17.93*	*44.59*	*200*
2	*8.83*	*0.00*	*12.58*	*6.24*	*27.65*	*200*
3	*1.39*	*0.00*	*20.71*	*23.83*	*45.93*	*200*
4	*0.74*	*0.00*	*19.41*	*23.14*	*43.29*	*200*
5	*0.55*	*0.00*	*0.03*	*5.00*	*5.58*	*200*
6	*4.64*	*0.00*	*92.20*	*29.59*	*126.43*	*200*

† Refer to Figure 5.1.1.

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

Table 5.4.6

DOSE RATES AT THE SURFACE OF THE HI-STAR 100 SYSTEM FOR NORMAL CONDITIONS
MPC-32 WITH DESIGN BASIS ZIRCALOY CLAD FUEL WITH NON-ZIRCALOY INCORE SPACERS
AT WORST CASE BURNUP AND COOLING TIME
24,500 MWD/MTU AND 12-YEAR COOLING

Dose Point*† *Location	***Fuel Gammas*†† *(mrem/hr)***	***Gammas from Incore Spacers (mrem/hr)***	***^{60}Co Gammas (mrem/hr)***	***Neutrons (mrem/hr)***	***Totals (mrem/hr)***	***10 CFR 71.47 Limit***
2a	*8.15*	*11.18*	*17.34*	*2.93*	*39.60*	*1000*
3a	*0.84*	*1.25*	*66.15*	*41.50*	*109.74*	*1000*
1	*1.12*	*1.49*	*20.36*	*6.52*	*29.48*	*200*
2	*5.15*	*6.94*	*10.39*	*2.27*	*24.75*	*200*
3	*0.76*	*0.96*	*17.09*	*8.67*	*27.48*	*200*
4	*0.40*	*0.50*	*16.01*	*8.42*	*25.34*	*200*
5	*0.20*†††	-	*0.02*	*1.82*	*2.05*	*200*
6	*4.69*†††	-	*76.09*	*10.76*	*91.54*	*200*

† Refer to Figure 5.1.1.

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

††† Gammas from incore spacers are included with fuel gammas.

Table 5.4.7

DOSE RATES AT THE SURFACE OF THE HI-STAR 100 SYSTEM FOR NORMAL CONDITIONS MPC-24 WITH TROJAN ZIRCALOY CLAD FUEL WITH NON-ZIRCALOY INCORE SPACERS 42,000 MWD/MTU AND 16-YEAR COOLING

Dose Point† Location	Fuel Gammas†† (mrem/hr)	Gammas from Incore Spacers (mrem/hr)	^{60}Co Gammas (mrem/hr)	Neutrons (mrem/hr)	Totals (mrem/hr)	10 CFR 71.47 Limit
2a	17.03	11.71	0.00	16.30	45.04	1000
3a	0.39	0.07	7.86	49.81	58.13	1000
1	1.62	1.09	4.54	14.43	21.68	200
2	10.94	7.72	0.05	9.69	28.40	200
3	0.62	0.34	5.71	8.00	14.67	200
4	0.36	0.17	2.70	7.81	11.04	200
5	0.34†††	-	0.04	3.17	3.55	200
6	6.99†††	-	21.45	23.26	51.70	200

† Refer to Figure 5.1.1.

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

††† Gammas from incore spacers are included with fuel gammas.

Table 5.4.8

DOSE RATES AT TWO METERS FROM THE HI-STAR 100 SYSTEM FOR NORMAL CONDITIONS
MPC-24 WITH TROJAN ZIRCALOY CLAD FUEL WITH NON-ZIRCALOY INCORE SPACERS
42,000 MWD/MTU AND 16-YEAR COOLING

Dose Point† Location	Fuel Gammas†† (mrem/hr)	Gammas from Incore Spacers (mrem/hr)	^{60}Co Gammas (mrem/hr)	Neutrons (mrem/hr)	Totals (mrem/hr)
1	1.52	1.11	0.55	2.52	5.69
2	3.41	2.68	0.37	2.67	9.12
3	1.21	0.87	1.21	1.71	4.99
4	0.97	0.66	1.12	1.50	4.24
5	0.02†††	-	0.03	0.27	0.32
6	0.56†††	-	2.05	0.87	3.49
10CFR71.47 Limit					10.00

† Refer to Figure 5.1.1.

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

††† Gammas from incore spacers are included with fuel gammas.

Table 5.4.9

*DOSE RATES AT ONE METER FOR ACCIDENT CONDITIONS
MPC-24 WITH TROJAN ZIRCALOY CLAD FUEL WITH NON- ZIRCALOY INCORE SPACERS
42,000 MWD/MTU AND 16-YEAR COOLING*

Dose Point† Location	Fuel Gammas†† (mrem/hr)	^{60}Co Gammas (mrem/hr)	Neutrons (mrem/hr)	Totals (mrem/hr)
1	7.01	5.69	106.18	118.88
2	31.31	0.30	356.39	387.99
3	3.36	8.38	69.43	81.16
4	1.91	4.56	49.17	55.64
5	0.11	0.25	12.42	12.78
6	34.74	128.20	82.48	245.42
10CFR71.51 Limit				1000.00

† Refer to Figure 5.1.2.

†† Gammas generated by neutron capture and gammas from incore spacers are included with fuel gammas.

Table 5.4.10

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~~TOTAL DOSE RATES~~
~~DOSE LOCATION AT TWO METERS FOR NORMAL CONDITIONS~~
~~MPC-24 DESIGN BASIS ZIRCALOY CLAD FUEL WITH ZIRCALOY INCORE SPACERS~~
~~AT VARYING BURNUPS AND COOLING TIMES~~

~~**Dose Point[†] Location**~~	~~**24,500 MWD/MTU 7 Year Cooling (mrem/hr)**~~	~~**29,500 MWD/MTU 8 Year Cooling (mrem/hr)**~~	~~**34,500 MWD/MTU 10 Year Cooling (mrem/hr)**~~	~~**39,500 MWD/MTU 12 Year Cooling (mrem/hr)**~~	~~**44,500 MWD/MTU 15 Year Cooling (mrem/hr)**~~
~~1~~	~~6.91~~	~~7.41~~	~~7.30~~	~~7.44~~	~~7.45~~
~~2~~	~~8.11~~	~~8.69~~	~~8.64~~	~~9.01~~	~~9.29~~
~~3~~	~~6.53~~	~~7.00~~	~~6.89~~	~~7.01~~	~~7.00~~
~~4~~	~~6.24~~	~~6.75~~	~~6.72~~	~~6.89~~	~~6.96~~
~~5~~	~~0.12~~	~~0.19~~	~~0.28~~	~~0.37~~	~~0.47~~
~~6~~	~~9.39~~	~~9.38~~	~~8.20~~	~~7.20~~	~~5.89~~
~~10CFR71.47 Limit~~	~~10.00~~	~~10.00~~	~~10.00~~	~~10.00~~	~~10.00~~

† Refer to Figure 5.1.1.

Table 5.4.11

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~~TOTAL DOSE RATES~~

~~DOSE LOCATION AT TWO METERS FOR NORMAL CONDITIONS~~

~~MPC-68 DESIGN BASIS ZIRCALOY CLAD FUEL AT VARYING BURNUPS AND COOLING TIMES~~

~~**Dose Point† Location**~~	~~**24,500 MWD/MTU 8 Year Cooling (mrem/hr)**~~	~~**29,500 MWD/MTU 9 Year Cooling (mrem/hr)**~~	~~**34,500 MWD/MTU 12 Year Cooling (mrem/hr)**~~	~~**39,500 MWD/MTU 15 Year Cooling (mrem/hr)**~~
~~1~~	~~6.57~~	~~7.04~~	~~6.92~~	~~6.80~~
~~2~~	~~8.01~~	~~8.92~~	~~8.78~~	~~8.90~~
~~3~~	~~6.09~~	~~6.22~~	~~5.82~~	~~5.38~~
~~4~~	~~5.77~~	~~5.86~~	~~5.47~~	~~5.02~~
~~5~~	~~0.07~~	~~0.11~~	~~0.16~~	~~0.21~~
~~6~~	~~8.38~~	~~7.80~~	~~6.59~~	~~5.19~~
~~10CFR71.47 Limit~~	~~10.00~~	~~10.00~~	~~10.00~~	~~10.00~~

† Refer to Figure 5.1.1.

Table 5.4.12

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~~TOTAL DOSE RATES~~

~~DOSE LOCATION AT ONE METER FOR ACCIDENT CONDITIONS~~

~~MPC-24 DESIGN BASIS ZIRCALOY CLAD FUEL WITH ZIRCALOY INCORE SPACERS~~

~~AT VARYING BURNUPS AND COOLING TIMES~~

~~**Dose Point† Location**~~	~~**24,500 MWD/MTU 7 Year Cooling (mrem/hr)**~~	~~**29,500 MWD/MTU 8 Year Cooling (mrem/hr)**~~	~~**34,500 MWD/MTU 10 Year Cooling (mrem/hr)**~~	~~**39,500 MWD/MTU 12 Year Cooling (mrem/hr)**~~	~~**44,500 MWD/MTU 15 Year Cooling (mrem/hr)**~~
~~1~~	~~76.86~~	~~98.98~~	~~120.33~~	~~144.25~~	~~170.56~~
~~2~~	~~144.84~~	~~217.78~~	~~302.20~~	~~393.66~~	~~497.08~~
~~3~~	~~51.17~~	~~67.08~~	~~82.95~~	~~100.58~~	~~120.08~~
4	~~37.46~~	~~49.10~~	~~60.70~~	~~73.56~~	~~87.79~~
~~5~~	~~4.34~~	~~7.07~~	~~10.35~~	~~13.82~~	~~17.79~~
~~6~~	~~623.93~~	~~630.03~~	~~559.32~~	~~500.23~~	~~419.18~~
~~10CFR71.51 Limit~~	~~1000.00~~	~~1000.00~~	~~1000.00~~	~~1000.00~~	~~1000.00~~

† Refer to Figure 5.1.2.

Table 5.4.13

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~~TOTAL DOSE RATES~~

~~DOSE LOCATION AT ONE METER FOR ACCIDENT CONDITIONS~~

~~MPC-68 DESIGN BASIS ZIRCALOY CLAD FUEL AT VARYING BURNUPS AND COOLING TIMES~~

~~**Dose Point† Location**~~	~~**24,500 MWD/MTU 8 Year Cooling (mrem/hr)**~~	~~**29,500 MWD/MTU 9 Year Cooling (mrem/hr)**~~	~~**34,500 MWD/MTU 12 Year Cooling (mrem/hr)**~~	~~**39,500 MWD/MTU 15 Year Cooling (mrem/hr)**~~
~~1~~	~~82.33~~	~~108.40~~	~~137.13~~	~~160.98~~
~~2~~	~~180.07~~	~~276.37~~	~~387.36~~	~~485.19~~
~~3~~	~~45.31~~	~~58.28~~	~~72.49~~	~~84.05~~
~~4~~	~~34.78~~	~~43.57~~	~~53.02~~	~~60.52~~
~~5~~	~~2.42~~	~~3.92~~	~~5.76~~	~~7.35~~
~~6~~	~~564.34~~	~~530.51~~	~~455.73~~	~~367.87~~
~~10CFR71.51 Limit~~	~~1000.00~~	~~1000.00~~	~~1000.00~~	~~1000.00~~

† Refer to Figure 5.1.2.

Table 5.4.14

PEAK-TO-AVERAGE RATIOS FOR THE DOSE COMPONENTS
AT VARIOUS LOCATIONS

Location	Fuel Gammas	Gammas from Neutrons	^{60}Co Gammas	Neutron
MPC-24				
Surface				
Pocket Trunnion	0.081	0.262	0.075	6.695
2*a*	0.713	0.955	0.407	2.362
3*a*	1.317	1.011	1.005	1.177
2 meter				
Pocket Trunnion	1.109	1.232	1.059	0.809
2	1.034	0.974	1.086	0.990
MPC-68				
Surface				
Pocket Trunnion	0.070	0.432	0.074	7.340
2*a*	0.737	0.977	1.123	2.284
3*a*	0.908	0.816	1.217	0.940
2 meter				
Pocket Trunnion	1.121	0.982	1.144	1.171
2	1.070	0.939	1.146	0.950

Table 5.4.15

DOSE RATES FOR NORMAL CONDITIONS SHOWING THE EFFECT OF PEAKING

Dose Point† Location	Fuel Gammas (mrem/hr)	Gammas from Neutrons (mrem/hr)	^{60}Co Gammas (mrem/hr)	Neutrons (mrem/hr)	Total (mrem/hr)
MPC-24					
Surface 44,500 MWD/MTU 15-Year Cooling					
Pocket Trunnion	0.16	0.40	1.91	102.56	105.03
2*a*	13.00	6.38	0.00	55.08	74.46
3*a*	0.72	0.77	33.40	156.37	191.26
2 meter 24,500 MWD/MTU 10-Year Cooling					
Pocket Trunnion	4.28	0.15	2.34	0.64	7.41
2	7.89	0.19	0.86	0.78	9.72
MPC-68					
Surface 27,500 MWD/MTU 9-Year Cooling					
Pocket Trunnion	0.27	0.29	2.19	50.16	52.91
2*a*	22.18	3.29	0.01	25.01	50.49
3*a*	0.13	0.14	97.91	27.37	125.55
2 meter 27,500 MWD/MTU 9-Year Cooling					
Pocket Trunnion	3.55	0.30	2.28	1.95	8.08
2	6.33	0.44	0.82	1.73	9.32

† Refer to Figure 5.1.1.

Table 5.4.16

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~~DOSE RATES FROM FUEL GAMMAS~~†
~~DOSE LOCATION ON THE SURFACE OF THE HI-STAR 100 SYSTEM FOR NORMAL CONDITIONS~~
~~MPC-24 DESIGN BASIS ZIRCALOY CLAD FUEL WITH NON-ZIRCALOY INCORE SPACERS~~
~~AT VARYING BURNUPS AND COOLING TIMES~~

~~Dose Point~~†† ~~Location~~	~~24,500 MWD/MTU 10 Year Cooling (mrem/hr)~~	~~29,500 MWD/MTU 12 Year Cooling (mrem/hr)~~	~~34,500 MWD/MTU 14 Year Cooling (mrem/hr)~~	~~37,500 MWD/MTU 15 Year Cooling (mrem/hr)~~
~~1~~	~~2.82~~	~~2.68~~	~~2.59~~	~~2.54~~
~~2~~	~~37.51~~	~~34.31~~	~~32.74~~	~~31.93~~
~~3~~	~~1.46~~	~~1.40~~	~~1.31~~	~~1.32~~
~~4~~	~~0.95~~	~~0.91~~	~~0.89~~	~~0.88~~
~~5~~	~~0.11~~	~~0.17~~	~~0.25~~	~~0.28~~
~~6~~	~~4.26~~	~~4.27~~	~~4.42~~	~~4.45~~

† Gammas generated by neutron capture and gammas from incore spacers are included with fuel gammas.

†† Refer to Figure 5.1.1.

Table 5.4.17

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~~DOSE RATES FROM ^{60}Co GAMMAS~~
~~DOSE LOCATION ON THE SURFACE OF THE HI-STAR 100 SYSTEM FOR NORMAL CONDITIONS~~
~~MPC-24 DESIGN BASIS ZIRCALOY CLAD FUEL WITH NON-ZIRCALOY INCORE SPACERS~~
~~AT VARYING BURNUPS AND COOLING TIMES~~

~~**Dose Point† Location**~~	~~**24,500 MWD/MTU 10 Year Cooling (mrem/hr)**~~	~~**29,500 MWD/MTU 12 Year Cooling (mrem/hr)**~~	~~**34,500 MWD/MTU 14 Year Cooling (mrem/hr)**~~	~~**37,500 MWD/MTU 15 Year Cooling (mrem/hr)**~~
~~1~~	~~18.93~~	~~16.24~~	~~13.55~~	~~12.24~~
~~2~~	~~0.05~~	~~0.01~~	~~0.01~~	~~0.01~~
~~3~~	~~57.74~~	~~49.54~~	~~32.69~~	~~29.51~~
~~4~~	~~14.42~~	~~12.38~~	~~10.33~~	~~9.33~~
~~5~~	~~0.02~~	~~0.02~~	~~0.02~~	~~0.02~~
~~6~~	~~70.63~~	~~60.60~~	~~50.57~~	~~45.66~~

† Refer to Figure 5.1.1.

Table 5.4.18

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~~DOSE RATES FROM NEUTRONS~~

~~DOSE LOCATION ON THE SURFACE OF THE HI-STAR 100 SYSTEM FOR NORMAL CONDITIONS~~

~~MPC-24 DESIGN BASIS ZIRCALOY CLAD FUEL WITH NON-ZIRCALOY INCORE SPACERS~~

~~AT VARYING BURNUPS AND COOLING TIMES~~

~~**Dose Point† Location**~~	~~**24,500 MWD/MTU 10 Year Cooling (mrem/hr)**~~	~~**29,500 MWD/MTU 12 Year Cooling (mrem/hr)**~~	~~**34,500 MWD/MTU 14 Year Cooling (mrem/hr)**~~	~~**37,500 MWD/MTU 15 Year Cooling (mrem/hr)**~~
~~1~~	~~4.49~~	~~7.21~~	~~10.70~~	~~12.13~~
~~2~~	~~4.07~~	~~7.81~~	~~11.58~~	~~13.13~~
~~3~~	~~22.50~~	~~36.12~~	~~65.94~~	~~74.78~~
~~4~~	~~4.54~~	~~7.29~~	~~10.81~~	~~12.25~~
~~5~~	~~0.85~~	~~1.36~~	~~2.02~~	~~2.29~~
~~6~~	~~6.66~~	~~10.68~~	~~15.85~~	~~17.97~~

† Refer to Figure 5.1.1.

Table 5.4.19

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~~TOTAL DOSE RATES~~

~~DOSE LOCATION ON THE SURFACE OF THE HI-STAR 100 SYSTEM FOR NORMAL CONDITIONS~~

~~MPC-24 DESIGN BASIS ZIRCALOY CLAD FUEL WITH NON-ZIRCALOY INCORE SPACERS~~

~~AT VARYING BURNUPS AND COOLING TIMES~~

~~**Dose Point† Location**~~	~~**24,500 MWD/MTU 10 Year Cooling (mrem/hr)**~~	~~**29,500 MWD/MTU 12 Year Cooling (mrem/hr)**~~	~~**34,500 MWD/MTU 14 Year Cooling (mrem/hr)**~~	~~**37,500 MWD/MTU 15 Year Cooling (mrem/hr)**~~
~~1~~	~~26.25~~	~~26.14~~	~~26.84~~	~~26.91~~
~~2~~	~~41.63~~	~~42.12~~	~~44.33~~	~~45.06~~
~~3~~	~~81.70~~	~~87.06~~	~~99.94~~	~~105.61~~
~~4~~	~~19.91~~	~~20.57~~	~~22.03~~	~~22.46~~
~~5~~	~~0.97~~	~~1.55~~	~~2.28~~	~~2.58~~
~~6~~	~~81.55~~	~~75.56~~	~~70.83~~	~~68.09~~
~~10CFR71.47 Limit~~	~~200.00~~	~~200.00~~	~~200.00~~	~~200.00~~

† Refer to Figure 5.1.1.

Table 5.4.20

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~~TOTAL DOSE RATES~~

~~DOSE LOCATION AT TWO METERS FOR NORMAL CONDITIONS~~

~~MPC-24 DESIGN BASIS ZIRCALOY CLAD FUEL WITH NON-ZIRCALOY INCORE SPACERS~~

~~AT VARYING BURNUPS AND COOLING TIMES~~

~~Dose Point~~[†] ~~Location~~	~~24,500 MWD/MTU 10 Year Cooling (mrem/hr)~~	~~29,500 MWD/MTU 12 Year Cooling (mrem/hr)~~	~~34,500 MWD/MTU 14 Year Cooling (mrem/hr)~~	~~37,500 MWD/MTU 15 Year Cooling (mrem/hr)~~
~~1~~	~~6.42~~	~~6.32~~	~~6.39~~	~~6.37~~
~~2~~	~~9.40~~	~~9.23~~	~~9.30~~	~~9.28~~
~~3~~	~~5.89~~	~~5.80~~	~~5.87~~	~~5.86~~
~~4~~	~~5.47~~	~~5.44~~	~~5.58~~	~~5.60~~
~~5~~	~~0.10~~	~~0.16~~	~~0.24~~	~~0.27~~
~~6~~	~~6.55~~	~~5.85~~	~~5.22~~	~~4.90~~
~~10CFR71.47 Limit~~	~~10.00~~	~~10.00~~	~~10.00~~	~~10.00~~

† Refer to Figure 5.1.1.

Table 5.4.21

DELETED

~~TOTAL DOSE RATES~~

~~DOSE LOCATION AT ONE METER FOR ACCIDENT CONDITIONS~~

~~MPC-24 DESIGN BASIS ZIRCALOY CLAD FUEL WITH NON-ZIRCALOY INCORE SPACERS~~

~~AT VARYING BURNUPS AND COOLING TIMES~~

~~**Dose Point† Location**~~	~~**24,500 MWD/MTU 10 Year Cooling (mrem/hr)**~~	~~**29,500 MWD/MTU 12 Year Cooling (mrem/hr)**~~	~~**34,500 MWD/MTU 14 Year Cooling (mrem/hr)**~~	~~**37,500 MWD/MTU 15 Year Cooling (mrem/hr)**~~
~~1~~	~~63.62~~	~~78.18~~	~~98.10~~	~~105.98~~
~~2~~	~~144.48~~	~~200.19~~	~~273.11~~	~~302.89~~
~~3~~	~~43.10~~	~~53.85~~	~~68.42~~	~~74.22~~
~~4~~	~~30.93~~	~~38.84~~	~~49.53~~	~~53.79~~
~~5~~	~~3.85~~	~~6.05~~	~~8.89~~	~~10.05~~
~~6~~	~~438.40~~	~~396.68~~	~~359.52~~	~~339.81~~
~~10CFR71.51 Limit~~	~~1000.00~~	~~1000.00~~	~~1000.00~~	~~1000.00~~

† Refer to Figure 5.1.2.

Table 5.4.22

DOSE RATES FOR
MPC-68 DESIGN BASIS STAINLESS STEEL CLAD FUEL
22,500 MWD/MTU AND 16-YEAR COOLING

Dose Point† Location	**Fuel Gammas†† (mrem/hr)**	**^{60}Co Gammas (mrem/hr)**	**Neutrons (mrem/hr)**	**Totals (mrem/hr)**
Dose Location at Surface for Normal Condition				
1	2.92	9.19	0.99	13.10
2*a*	39.26	0.00	1.20	40.46
3*a*	0.33	33.55	2.52	36.40
4	0.37	8.50	0.51	9.38
5	0.01	0.01	0.10	0.12
6	2.32	31.59	1.46	35.37
10CFR71.47 Limit				200.00
Dose Location at Two Meters for Normal Condition				
1	3.45	1.00	0.17	4.62
2	7.68	0.27	0.20	8.15
3	2.26	1.28	0.11	3.65
4	1.69	1.33	0.11	3.13
5	0.00	0.00	0.01	0.01
6	0.26	2.73	0.06	3.05
10CFR71.47 Limit				10.00
Dose Location at One Meter for Accident Condition				
1	9.41	10.90	7.94	28.25
2	46.41	0.23	26.02	72.66
3	3.52	6.97	4.08	14.57
4	2.10	6.03	2.87	11.00
5	0.02	0.05	0.41	0.48
6	11.57	182.92	5.42	199.91
10CFR71.51 Limit				1000.00

Note: The more conservative limit of 200 mrem/hr was applied for dose locations 2a and 3a while dose locations 2 and 3 were not analyzed.

† Refer to Figures 5.1.1 and 5.1.2.

†† Gammas generated by neutron capture and gammas from incore spacers are included with fuel gammas.

Table 5.4.23

DOSE RATES FOR
MPC-24 DESIGN BASIS STAINLESS STEEL CLAD FUEL
30,000 MWD/MTU AND 19-YEAR COOLING

Dose Point† Location	Fuel Gammas†† (mrem/hr)	^{60}Co Gammas (mrem/hr)	Neutrons (mrem/hr)	Totals (mrem/hr)
Dose Location at Surface for Normal Condition				
1	3.02	6.23	5.05	14.30
2*a*	39.57	0.02	4.57	44.16
3*a*	1.38	15.03	31.13	47.54
4	1.01	4.75	5.10	10.86
5	0.12	0.01	0.95	1.08
6	4.57	23.26	7.49	35.32
10CFR71.47 Limit				200.00
Dose Location at Two Meters for Normal Condition				
1	3.42	0.81	0.83	5.06
2	8.24	0.26	0.89	9.39
3	2.86	0.81	0.80	4.47
4	2.33	0.82	0.86	4.01
5	0.01	0.00	0.10	0.11
6	0.44	1.92	0.32	2.68
10CFR71.47 Limit				10.00
Dose Location at One Meter for Accident Condition				
1	8.78	8.02	34.69	51.49
2	47.52	0.27	110.45	158.24
3	6.48	4.90	24.80	36.18
4	3.50	3.76	18.17	25.43
5	0.03	0.05	4.12	4.20
6	23.76	128.53	28.78	181.07
10CFR71.51 Limit				1000.00

Note: The more conservative limit of 200 mrem/hr was applied for dose locations 2a and 3a while dose locations 2 and 3 were not analyzed.

† Refer to Figures 5.1.1 and 5.1.2.

†† Gammas generated by neutron capture and gammas from incore spacers are included with fuel gammas.

Table 5.4.24

DOSE RATES FOR
MPC-24 DESIGN BASIS STAINLESS STEEL CLAD FUEL
40,000 MWD/MTU AND 24-YEAR COOLING

Dose Point[†] Location	Fuel Gammas[††] (mrem/hr)	^{60}Co Gammas (mrem/hr)	Neutrons (mrem/hr)	Totals (mrem/hr)
Dose Location at Surface for Normal Condition				
1	2.58	6.52	13.11	22.21
2*a*	32.07	0.01	14.19	46.27
3*a*	1.36	15.72	80.81	97.89
4	0.89	4.97	13.24	19.10
5	0.30	0.01	2.47	2.78
6	4.58	24.33	19.42	48.33
10CFR71.47 Limit				200.00
Dose Location at Two Meters for Normal Condition				
1	2.81	0.84	2.16	5.81
2	6.65	0.27	2.30	9.22
3	2.35	0.85	2.09	5.29
4	1.92	0.86	2.24	5.02
5	0.02	0.01	0.26	0.29
6	0.37	2.00	0.83	3.20
10CFR71.47 Limit				10.00
Dose Location at One Meter for Accident Condition				
1	6.80	8.39	90.10	105.29
2	36.58	0.28	286.87	323.73
3	5.00	5.13	64.39	74.52
4	2.72	3.94	47.18	53.84
5	0.05	0.05	10.71	10.81
6	18.30	134.43	74.69	227.42
10CFR71.51 Limit				1000.00

Note: The more conservative limit of 200 mrem/hr was applied for dose locations 2a and 3a while dose locations 2 and 3 were not analyzed.

† Refer to Figures 5.1.1 and 5.1.2.

†† Gammas generated by neutron capture and gammas from incore spacers are included with fuel gammas.

Table 5.4.25

COMPARISON OF NEUTRON SOURCE PER INCH PER SECOND FOR DESIGN BASIS 7X7 FUEL AND DESIGN BASIS DRESDEN UNIT 1 FUEL

Assembly	**Active fuel length (inch)**	**Neutrons per sec per inch**	**Neutrons per sec per inch with Sb-Be source**	**Reference for neutrons per sec per inch**
7x7 design basis	144	6.*63*~~38~~E+5	N/A	Table 5.2.13 39.5 GWD/MTU and 1*4*~~5~~ year cooling
6x6 design basis	110	2.85E+5	3.45E+5	Table 5.2.14
6x6 design basis MOX	110	3.67E+5	4.27E+5	Table 5.2.17

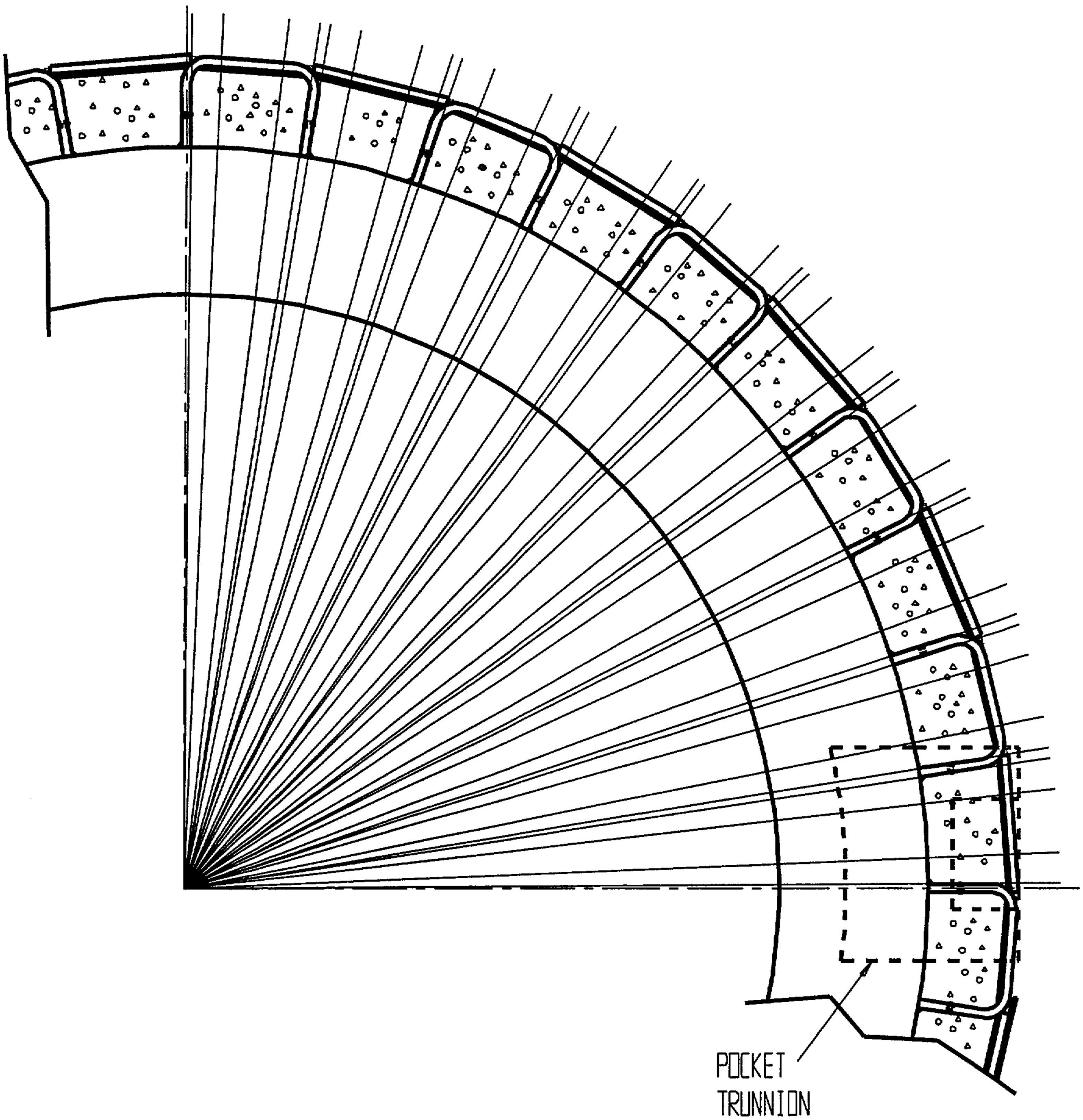

FIGURE 5.4.1; DEPICTION OF THE AZIMUTHAL SEGMENTATION OF THE OVERPACK USED IN ANALYZING NEUTRON AND PHOTON STREAMING

5.5 REGULATORY COMPLIANCE

The analysis presented in this chapter has shown that the external radiation levels will not increase during normal conditions of transport consistent with the tests specified in 10CFR71.71. This chapter also confirms that the external dose rates from HI-STAR 100 System, when fully loaded with fuel assemblies that meet the acceptance criteria specified in Chapter 1, are less than the regulatory limits specified in 10CFR71.47.

This chapter also demonstrates that the maximum external radiation level at one meter from the external surface of the package does not exceed 1 Rem/hr (10 mSv/hr) during the hypothetical accident conditions consistent with the tests specified in 10CFR71.73.

Table*s* 5.5.1 ~~and~~ *through* 5.5.3 summarize the maximum dose rates, including the effect of radiation peaking as discussed in Subsection 5.4.1, and demonstrate the HI-STAR 100 System's compliance with the regulatory requirements of 10CFR71.47 and 10CFR71.51. Since these dose rates include the effect of peaking, they may not be equivalent to values reported earlier in this chapter *which were surface average dose rates*. In these tables "Side" refers to the dose point location that has the maximum dose rate from locations 1-4 on Figures 5.1.1 and 5.1.2; "Top" and "Bottom" refer to locations 5 and 6, respectively, on Figures 5.1.1 and 5.1.2. *Dose location 2a and 3a from Figure 5.1.1 are not used in Tables 5.5.1 through 5.5.3.* Since the maximum dose rate at each location is provided, the corresponding burnup and cooling time may be different between locations and therefore is not listed in the tables. *Some of the dose rates in these tables are very close to the regulatory limit with one value equaling the limit due to rounding. These high dose rates are acceptable because the analysis has been demonstrated to be conservative. In addition, it is extremely unlikely that the casks would be loaded with all fuel assemblies containing the same identical burnup and cooling time analyzed. Finally, the ultimate demonstration of compliance with the 10 CFR 71 regulations will be the measurements that are taken before shipment of the fuel.*

Table 5.5.1
MAXIMUM EXTERNAL DOSE RATES FOR THE
HI-STAR 100 SYSTEM CONTAINING THE MPC-24

Normal Conditions of Transport			
	External Surface of Package		
Radiation (mrem/hr)	**Top**	**Side**	**Bottom**
Gamma†	0.81	*30.66*	*102.04*
Neutron	4.06	*14.23*	*19.04*
Total	4.57	*44.89*	*121.05*
10 CFR 71.47(b)(1) Limit	200	200	200
	2 Meters from Vehicle Outer Surface ††		
Radiation (mrem/hr)	**Top**	**Side**	**Bottom**
Gamma†	0.04	*7.70*	9.10
Neutron	0.43	*2.30*	0.29
Total	0.47	*10.00*	9.39
10 CFR 71.47(b)(3) Limit	10	10	10
Hypothetical Accident Conditions			
	1 Meter from Surface of Package †††		
Radiation (mrem/hr)	**Top**	**Side**	**Bottom**
Gamma†	0.19	25.28	*557.20*
Neutron	17.60	471.80	*73.21*
Total	17.79	497.08	*630.41*
10 CFR 71.51(a)(2) Limit	1000	1000	1000

† This includes fuel gammas, gammas from hardware activation including incore spacers, and gammas generated by neutron capture.

†† The vehicle outer surface is the outer radial surface of the impact limiters, the end of the top impact limiter, and 6 feet from the end of the bottom impact limiter.

††† The impact limiters are not present.

Table 5.5.2

MAXIMUM EXTERNAL DOSE RATES FOR THE HI-STAR 100 SYSTEM CONTAINING THE MPC-32

Normal Conditions of Transport			
	External Surface of Package		
Radiation (mrem/hr)	***Top***	***Side***	***Bottom***
Gamma†	*0.94*	*12.14*	*96.84*
Neutron	*8.38*	*44.79*	*29.59*
Total	*9.32*	*56.93*	*126.43*
10 CFR 71.47(b)(1) Limit	*200*	*200*	*200*
	2 Meters from Vehicle Outer Surface ††		
Radiation (mrem/hr)	***Top***	***Side***	***Bottom***
Gamma†	*0.06*	*5.10*	*1.12*
Neutron	*0.79*	*4.81*	*8.43*
Total	*0.85*	*9.91*	*9.55*
10 CFR 71.47(b)(3) Limit	*10*	*10*	*10*
Hypothetical Accident Conditions			
	1 Meter from Surface of Package †††		
Radiation (mrem/hr)	***Top***	***Side***	***Bottom***
Gamma†	*0.23*	*17.27*	*541.10*
Neutron	*33.64*	*354.51*	*107.25*
Total	*33.87*	*371.78*	*648.35*
10 CFR 71.51(a)(2) Limit	*1000*	*1000*	*1000*

~~DELETED~~

† This includes fuel gammas, gammas from hardware activation including incore spacers, and gammas generated by neutron capture.

†† The vehicle outer surface is the outer radial surface of the impact limiters, the end of the top impact limiter, and 6 feet from the end of the bottom impact limiter.

††† The impact limiters are not present.

Table 5.5.3
MAXIMUM EXTERNAL DOSE RATES FOR THE
HI-STAR 100 SYSTEM CONTAINING THE MPC-68

Normal Conditions of Transport			
	External Surface of Package		
Radiation (mrem/hr)	**Top**	**Side**	**Bottom**
Gamma†	*2.53*	*12.71*	*7.93*
Neutron	*0.33*	*39.98*	*93.05*
Total	*2.86*	*52.69*	*100.98*
10 CFR 71.47(b)(1) Limit	200	200	200
	2 Meters from Vehicle Outer Surface ††		
Radiation (mrem/hr)	**Top**	**Side**	**Bottom**
Gamma†	*0.03*	*5.92*	8.07
Neutron	*0.29*	*4.05*	0.31
Total	*0.32*	*9.97*	8.38
10 CFR 71.47(b)(3) Limit	10	10	10
Hypothetical Accident Conditions			
	1 Meter from Surface of Package †††		
Radiation (mrem/hr)	**Top**	**Side**	**Bottom**
Gamma†	*0.11*	*22.15*	*534.51*
Neutron	*10.92*	*600.71*	*30.19*
Total	*11.03*	*622.86*	*564.70*
10 CFR 71.51(a)(2) Limit	1000	1000	1000

† This includes fuel gammas, gammas from hardware activation including incore spacers, and gammas generated by neutron capture.

†† The vehicle outer surface is the outer radial surface of the impact limiters, the end of the top impact limiter, and 6 feet from the end of the bottom impact limiter.

††† The impact limiters are not present.

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[5.1.3] O.W. Hermann, R.M. Westfall, "ORIGEN-S: SCALE System Module to Calculate Fuel Depletion, Actinide Transmutation, Fission Product Buildup and Decay, and Associated Radiation Source Terms," NUREG/CR-0200, Revision 5, (ORNL/NUREG/CSD-2/V2/R5), Oak Ridge National Laboratory, September 1995.

[5.1.4] O.W. Hermann, C.V. Parks, "SAS2H: A Coupled One-Dimensional Depletion and Shielding Analysis Module," NUREG/CR-0200, Revision 6, (ORNL/NUREG/CSD-2/V2/R6), Oak Ridge National Laboratory, September 1998.

[5.1.5] O.W. Hermann, R.M. Westfall, "ORIGEN-S: SCALE System Module to Calculate Fuel Depletion, Actinide Transmutation, Fission Product Buildup and Decay, and Associated Radiation Source Terms," NUREG/CR-0200, Revision 6, (ORNL/NUREG/CSD-2/V2/R6), Oak Ridge National Laboratory, September 1998.

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[5.2.7] "Characteristics Database System LWR Assemblies Database," DOE/RW-0184-R1, U.S. Department of Energy, July 1992.

[5.2.8] S.B. Ludwig and J.P. Renier, "Standard and Extended Burnup PWR and BWR Reactor Models for the ORIGEN2 Computer Code," ORNL/TM-11018, Oak Ridge National Laboratory, December 1989.

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[5.4.1] "American National Standard Neutron and Gamma-Ray Flux-to-Dose Rate Factors", ANSI/ANS-6.1.1-1977.

[5.4.2] D. J. Whalen, et al., "MCNP: Photon Benchmark Problems," LA-12196, Los Alamos National Laboratory, September 1991.

[5.4.3] D. J. Whalen, et al., "MCNP: Neutron Benchmark Problems," LA-12212, Los Alamos National Laboratory, November 1991.

[5.4.4] J. C. Wagner, et al., "MCNP: Criticality Safety Benchmark Problems," LA-12415, Los Alamos National Laboratory, October 1992.

[5.4.5] S. E. Turner, "Uncertainty Analysis - Axial Burnup Distribution Effects," presented in "Proceedings of a Workshop on the Use of Burnup Credit in Spent Fuel Transport Casks," SAND-89-0018, Sandia National Laboratory, Oct. 1989.

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APPENDIX 5.A

SAMPLE INPUT FILE FOR SAS2H

(Total number of pages in this appendix : 3)

```
=SAS2H    PARM='halt08,skipshipdata'
bw 15x15 PWR assembly
' fuel temp 923
44groupndf5    LATTICECELL
UO2 1 0.95 923 92234 0.03026 92235 3.4 92236 0.01564
    92238 96.5541 END
'
' Zirc 4 composition
ARBM-ZIRC4 6.55 4 1 0 0 50000 1.7 26000 0.24 24000 0.13 40000 97.93
          2 1.0 595 END
'
' water with 652.5 ppm boron
H2O       3 DEN=0.7135 1  579 END
ARBM-BORMOD  0.7135 1 1 0 0 5000 100 3 652.5E-6  579  END
'
co-59  3 0 1-20 579 end
kr-83  1 0 1-20 923 end
kr-84  1 0 1-20 923 end
kr-85  1 0 1-20 923 end
kr-86  1 0 1-20 923 end
sr-90  1 0 1-20 923 end
y-89   1 0 1-20 923 end
zr-94  1 0 1-20 923 end
zr-95  1 0 1-20 923 end
mo-94  1 0 1-20 923 end
mo-95  1 0 1-20 923 end
nb-94  1 0 1-20 923 end
nb-95  1 0 1-20 923 end
tc-99  1 0 1-20 923 end
ru-106  1 0 1-20 923 end
rh-103  1 0 1-20 923 end
rh-105  1 0 1-20 923 end
sb-124  1 0 1-20 923 end
sn-126  1 0 1-20 923 end
xe-131  1 0 1-20 923 end
xe-132  1 0 1-20 923 end
xe-134  1 0 1-20 923 end
'
xe-135  1 0 1-09 923 end
'
xe-136  1 0 1-20 923 end
cs-133  1 0 1-20 923 end
cs-134  1 0 1-20 923 end
cs-135  1 0 1-20 923 end
cs-137  1 0 1-20 923 end
ba-136  1 0 1-20 923 end
la-139  1 0 1-20 923 end
ce-144  1 0 1-20 923 end
pr-143  1 0 1-20 923 end
nd-143  1 0 1-20 923 end
nd-144  1 0 1-20 923 end
nd-145  1 0 1-20 923 end
nd-146  1 0 1-20 923 end
nd-147  1 0 1-20 923 end
nd-148  1 0 1-20 923 end
nd-150  1 0 1-20 923 end
pm-147  1 0 1-20 923 end
pm-148  1 0 1-20 923 end
pm-149  1 0 1-20 923 end
```

```
sm-147  1 0 1-20 923 end
sm-148  1 0 1-20 923 end
sm-149  1 0 1-20 923 end
sm-150  1 0 1-20 923 end
sm-151  1 0 1-20 923 end
sm-152  1 0 1-20 923 end
eu-151  1 0 1-20 923 end
eu-153  1 0 1-20 923 end
eu-154  1 0 1-20 923 end
eu-155  1 0 1-20 923 end
gd-154  1 0 1-20 923 end
gd-155  1 0 1-20 923 end
gd-157  1 0 1-20 923 end
gd-158  1 0 1-20 923 end
gd-160  1 0 1-20 923 end
'
END COMP
'
' - - - - - - - - - - - - - - - - - - - - - - - - - - - - - - - -
'
'  FUEL-PIN-CELL GEOMETRY:
'
SQUAREPITCH  1.44272 0.950468 1 3 1.08712 2 0.97028 0 END
'
' - - - - - - - - - - - - - - - - - - - - - - - - - - - - - - - -
'
'  MTU in this model is 0.495485 based on fuel dimensions provided
'
'  1 power cycle will be used and a library will be generated every
'  2500 MWD/MTU  power level is 40 MW/MTU
'  therefore 62.5 days per 2500 MWD/MTU
'  Below
'  BURN=62.5*NLIB/CYC
'  POWER=MTU*40
'
'  Number of libraries is 17 which is 42,500 MWD/MTU burnup (17*2500)
'
'  ASSEMBLY AND CYCLE PARAMETERS:
'
NPIN/ASSM=208 FUELNGTH=365.76 NCYCLES=1 NLIB/CYC=17
PRINTLEVEL=1
LIGHTEL=5  INPLEVEL=1   NUMHOLES=17
NUMINStr= 0  ORTUBE= 0.6731  SRTUBE=0.63246   END
POWER=19.81938 BURN=1062.5 END
'
 O  66.54421
 FE  0.24240868
 ZR 98.78151  CR  0.1311304   SN  1.714782
'
END
=SAS2H    PARM='restarts,halt17,skipshipdata'
bw 15x15 PWR assembly
END
```

APPENDIX 5.B

SAMPLE INPUT FILE FOR ORIGEN-S

(Total number of pages in this appendix : 6)

```
#ORIGENS
0$$  A4 33 A8 26 A11 71  E
1$$  1 T
bw 15x15 FUEL -- FT33F001 -
'
' SUBCASE  1 LIBRARY POSITION  1
'
'   lib      pos       grms   photon group
3$$ 33  A3  1  0  A16 2   E T
35$$ 0 T
56$$  5  5  A6 3  A10  0  A13 9  A15 3 A19 1 E
57** 0.0 A3 1.E-5 0.0625 E T
 FUEL 3.4
BW 15x15 0.495485 MTU
58** 19.81938  19.81938 19.81938 19.81938  19.81938
60**  1.0000   3.0000  15.0000  30.0000  62.5
66$$ A1 2 A5 2 A9 2  E
73$$  922350  922340  922360  922380  80000  500000
      260000  240000  400000
74** 16846.48 149.9336 77.49379 478410.7 66544.21 1714.782
     242.0868 131.1304 98781.51
75$$  2  2  2  2  4  4  4  4  4   T
'
' SUBCASE  2 LIBRARY POSITION  2
'
3$$ 33  A3   2  0  A16 2  A33 0  E T
35$$ 0 T
56$$  3  3  A6 3  A10   5  A15 3 A19 1 E
57** 0.0 A3 1.E-5  0.0625 E T
 fuel
BW 15X15
58** 19.81938 19.81938 19.81938
60** 18.5 37.0 62.5
66$$ A1 2 A5 2 A9 2  E T
'
' SUBCASE  3 LIBRARY POSITION  3
'
3$$ 33  A3   3  0  A16 2  A33 0  E T
35$$ 0 T
56$$  3  3  A6 3  A10   3  A15 3 A19 1 E
57** 0.0 A3 1.E-5  0.0625 E T
 fuel
BW 15X15
58** 19.81938 19.81938 19.81938
60** 18.5 37.0 62.5
66$$ A1 2 A5 2 A9 2  E T
'
' SUBCASE  4 LIBRARY POSITION  4
'
3$$ 33  A3   4  0  A16 2  A33 0  E T
35$$ 0 T
56$$  3  3  A6 3  A10   3  A15 3 A19 1 E
57** 0.0 A3 1.E-5  0.0625 E T
 fuel
BW 15X15
58** 19.81938 19.81938 19.81938
60** 18.5 37.0 62.5
66$$ A1 2 A5 2 A9 2  E T
'
```

```
' SUBCASE  5 LIBRARY POSITION  5
'
3$$ 33  A3   5  0  A16 2  A33 0  E T
35$$ 0 T
56$$  3  3  A6 3  A10   3  A15 3 A19 1 E
57** 0.0 A3 1.E-5  0.0625 E T
 fuel
BW 15X15
58** 19.81938 19.81938 19.81938
60** 18.5 37.0 62.5
66$$ A1 2 A5 2 A9 2  E T
'
' SUBCASE  6 LIBRARY POSITION  6
'
3$$ 33  A3   6  0  A16 2  A33 0  E T
35$$ 0 T
56$$  3  3  A6 3  A10   3  A15 3 A19 1 E
57** 0.0 A3 1.E-5  0.0625 E T
 fuel
BW 15X15
58** 19.81938 19.81938 19.81938
60** 18.5 37.0 62.5
66$$ A1 2 A5 2 A9 2  E T
'
' SUBCASE  7 LIBRARY POSITION  7
'
3$$ 33  A3   7  0  A16 2  A33 0  E T
35$$ 0 T
56$$  3  3  A6 3  A10   3  A15 3 A19 1 E
57** 0.0 A3 1.E-5  0.0625 E T
 fuel
BW 15X15
58** 19.81938 19.81938 19.81938
60** 18.5 37.0 62.5
66$$ A1 2 A5 2 A9 2  E T
'
' SUBCASE  8 LIBRARY POSITION  8
'
3$$ 33  A3   8  0  A16 2  A33 0  E T
35$$ 0 T
56$$  3  3  A6 3  A10   3  A15 3 A19 1 E
57** 0.0 A3 1.E-5  0.0625 E T
 fuel
BW 15X15
58** 19.81938 19.81938 19.81938
60** 18.5 37.0 62.5
66$$ A1 2 A5 2 A9 2  E T
'
' SUBCASE  9 LIBRARY POSITION  9
'
3$$ 33  A3   9  0  A16 2  A33 0  E T
35$$ 0 T
56$$  3  3  A6 3  A10   3  A15 3 A19 1 E
57** 0.0 A3 1.E-5  0.0625 E T
 fuel
BW 15X15
58** 19.81938 19.81938 19.81938
60** 18.5 37.0 62.5
66$$ A1 2 A5 2 A9 2  E T
```

```
'
' SUBCASE 10 LIBRARY POSITION 10
'
3$$ 33  A3  10  0  A16 2  A33 0  E T
35$$ 0 T
56$$  3  3  A6 3  A10   3  A15 3 A19 1 E
57** 0.0 A3 1.E-5  0.0625 E T
 fuel
BW 15X15
58** 19.81938 19.81938 19.81938
60** 18.5 37.0 62.5
66$$ A1 2 A5 2 A9 2  E T
'
' SUBCASE 11 LIBRARY POSITION 11
'
3$$ 33  A3  11  0  A16 2  A33 0  E T
35$$ 0 T
56$$  3  3  A6 3  A10   3  A15 3 A19 1 E
57** 0.0 A3 1.E-5  0.0625 E T
 fuel
BW 15X15
58** 19.81938 19.81938 19.81938
60** 18.5 37.0 62.5
66$$ A1 2 A5 2 A9 2  E T
'
' SUBCASE 12 LIBRARY POSITION 12
'
3$$ 33  A3  12  0  A16 2  A33 0  E T
35$$ 0 T
56$$  3  3  A6 3  A10   3  A15 3 A19 1 E
57** 0.0 A3 1.E-5  0.0625 E T
 fuel
BW 15X15
58** 19.81938 19.81938 19.81938
60** 18.5 37.0 62.5
66$$ A1 2 A5 2 A9 2  E T
'
' SUBCASE 13 LIBRARY POSITION 13
'
3$$ 33  A3  13  0  A16 2  A33 0  E T
35$$ 0 T
56$$  3  3  A6 3  A10   3  A15 3 A19 1 E
57** 0.0 A3 1.E-5  0.0625 E T
 fuel
BW 15X15
58** 19.81938 19.81938 19.81938
60** 18.5 37.0 62.5
66$$ A1 2 A5 2 A9 2  E T
'
' SUBCASE 14 LIBRARY POSITION 14
'
3$$ 33  A3  14  0  A16 2  A33 0  E T
35$$ 0 T
56$$  3  3  A6 3  A10   3  A15 3 A19 1 E
57** 0.0 A3 1.E-5  0.0625 E T
 fuel
BW 15X15
58** 19.81938 19.81938 19.81938
60** 18.5 37.0 62.5
```

```
66$$ A1 2 A5 2 A9 2  E T
'
' SUBCASE 15 LIBRARY POSITION 15
'
3$$ 33  A3  15  0  A16 2  A33 0  E T
35$$ 0 T
56$$  3  3  A6 3  A10   3  A15 3 A19 1 E
57** 0.0 A3 1.E-5  0.0625 E T
 fuel
BW 15X15
58** 19.81938 19.81938 19.81938
60** 18.5 37.0 62.5
66$$ A1 2 A5 2 A9 2  E T
'
' SUBCASE 16 LIBRARY POSITION 16
'
3$$ 33  A3  16  A4 7 0  A16 2  A33 18  E T
35$$ 0 T
56$$  3  3  A6 1  A10  3  A15 3 A19 1 E
57** 0.0 A3 1.E-5  0.0625 E T
 fuel
BW 15X15
58** 19.81938 19.81938 19.81938
60** 18.5 37.0 62.5
66$$ A1 2 A5 2 A9 2  E T
'
' SUBCASE - decay
'
54$$ A8 1 E
56$$ 0 9 A6 1 A10 3 A14 3 A15 1 A19 1 E
57** 0.0 0 1.E-5  E T
 fuel enrichment above
60** 0.5 0.75 1.0 4.0 8.0 12.0 24.0 48.0 96.0
61** F0.1
65$$
'GRAM-ATOMS   GRAMS    CURIES     WATTS-ALL  WATTS-GAMMA
     3Z      0  1  0     0 0 0      1 0 0        3Z         6Z
     3Z      0  1  0     0 0 0      1 0 0        3Z         6Z
     3Z      0  1  0     0 0 0      1 0 0        3Z         6Z   T
'
' SUBCASE - decay
'
54$$ A8 1 E
56$$ 0 9 A6 1 A10 9 A14 4 A15 1 A19 1 E
57** 4.0 0 1.E-5  E T
 fuel enrichment above
60** 10.0 20.0 30.0 60.0 90.0 120.0 180.0 240.0 365.0
61** F0.1
65$$
'GRAM-ATOMS   GRAMS    CURIES     WATTS-ALL  WATTS-GAMMA
     3Z      0  1  0     0 0 0      1 0 0        3Z         6Z
     3Z      0  1  0     0 0 0      1 0 0        3Z         6Z
     3Z      0  1  0     0 0 0      1 0 0        3Z         6Z   T
'
' SUBCASE - decay
'
54$$ A8 0 E
56$$ 0 9 A6 1 A10 9 A14 5 A15 1 A19 1 E
57** 1.0 0 1.E-5  E T
```

```
 fuel enrichment above
60** 1.5 3.0 4.0 5.0 6.0 7.0 8.0 9.0 10.0
61** F1.0e-5
65$$
'GRAM-ATOMS   GRAMS    CURIES     WATTS-ALL  WATTS-GAMMA
     3Z      0  1  0     1 0 0      1 0 0        3Z          6Z
     3Z      0  1  0     1 0 0      1 0 0        3Z          6Z
     3Z      0  1  0     1 0 0      1 0 0        3Z          6Z
81$$ 2 0 26 1 E
82$$ 0 2 2 2 2 2 2 2 2
83**  1.1E+7  8.0E+6 6.0E+6 4.0E+6 3.0E+6 2.5E+6 2.0E+6 1.5E+6
      1.0E+6  7.0E+5 4.5E+5 3.0E+5 1.5E+5 1.0E+5 7.0E+4 4.5E+4
      3.0E+4  2.0E+4 1.0E+4
84**  20.0E+6 6.43E+6 3.0E+6 1.85E+6 1.40E+6 9.00E+5 4.00E+5 1.0E+5 T
'
'
'
'
'
'
'
'
' SUBCASE - decay
'
54$$ A8 0 E
56$$ 0 10 A6 1 A10 9 A14 5 A15 1 A19 1 E
57** 10.0 0 1.E-5  E T
 fuel enrichment above
60** 11.0 12.0 13.0 14.0 15.0 16.0 17.0 18.0 19.0 20.0
61** F1.0e-5
65$$
'GRAM-ATOMS   GRAMS    CURIES     WATTS-ALL  WATTS-GAMMA
     3Z      0  1  0     1 0 0      1 0 0        3Z          6Z
     3Z      0  1  0     1 0 0      1 0 0        3Z          6Z
     3Z      0  1  0     1 0 0      1 0 0        3Z          6Z
81$$ 2 0 26 1 E
82$$ 2 2 2 2 2 2 2 2 2 2
83**  1.1E+7  8.0E+6 6.0E+6 4.0E+6 3.0E+6 2.5E+6 2.0E+6 1.5E+6
      1.0E+6  7.0E+5 4.5E+5 3.0E+5 1.5E+5 1.0E+5 7.0E+4 4.5E+4
      3.0E+4  2.0E+4 1.0E+4
84**  20.0E+6 6.43E+6 3.0E+6 1.85E+6 1.40E+6 9.00E+5 4.00E+5 1.0E+5 T
'
'
'
'
'
'
'
'
'
'
' END
'
56$$ F0 T
END
```

APPENDIX 5.C

SAMPLE INPUT FILE FOR MCNP

(Total number of pages in this appendix : 34)

```
message:    outp=m68n61co  srctp=m68n61cs  runtpe=m68n61cr
            mctal=m68n61cm  wssa=m68n61cw  rssa=m68n61cw

m68n61c
c        HI STAR 100 MPC68
c
c      two pocket trunions modeled
c      holtite present
c      impact limiters present
c      radial model
c
c      origen is at the bottom of the overpack - as an example of the origen
c      item 1 on drawing 1397 is 6 inches thick and extends
c      from 0.0 to 6.0 inches in the axial direction
c
c      universe 1
c
c       egg crate
301     0           -30             -400 u=1
302     0            31             -400 u=1
303     0         30 -31 -32        -400 u=1
304     0         30 -31  33        -400 u=1
305     5 -7.92         28      400 -670 u=1
306     5 -7.92             -23 400 -670 u=1
307     5 -7.92      -15 23 -28 400 -670 u=1
308     5 -7.92   20     23 -28 400 -670 u=1
309     5 -7.92         28      670 -460 u=1
310     5 -7.92             -23 670 -460 u=1
311     5 -7.92      -15 23 -28 670 -460 u=1
312     5 -7.92   20     23 -28 670 -460 u=1
c       boral and inside of egg crate and outside of fuel
314     0         15 -30 23 -28 400 -410 u=-1
315     0         31 -20 23 -28 400 -410 u=-1
316     0         30 -31 33 -28 400 -410 u=-1
317     0         30 -31 23 -32 400 -410 u=-1
318     0         15 -18 26 -28 410 -435 u=-1
319     0         19 -20 26 -28 410 -435 u=-1
320     0         15 -17 23 -24 410 -435 u=-1
321     0         15 -17 25 -26 410 -435 u=-1
322     6 -2.644  18 -19 27 -28 410 -435 u=-1
323     5 -7.92   18 -19 26 -27 410 -435 u=-1
324     6 -2.644  15 -16 24 -25 410 -435 u=-1
325     5 -7.92   16 -17 24 -25 410 -435 u=-1
326     0         17 -30 23 -26 410 -435 u=-1
327     0         31 -20 23 -26 410 -435 u=-1
328     0         30 -31 23 -32 410 -435 u=-1
329     0         30 -31 33 -26 410 -435 u=-1
330     0         15 -30 23 -28 435 -460 u=-1
331     0         31 -20 23 -28 435 -460 u=-1
332     0         30 -31 33 -28 435 -460 u=-1
333     0         30 -31 23 -32 435 -460 u=-1
c       fuel element
338     5 -1.48621   30 -31 32 -33     -420 u=1
339     2 -4.29251 30 -31 32 -33 420 -425 u=-1
340     0            30 -31 32 -33 425 -430 u=-1
341     5 -0.265322 30 -31 32 -33 430 -440 u=-1
342     5 -0.677510 30 -31 32 -33 440 -445 u=-1
343     5 -1.369221 30 -31 32 -33 445 -450 u=-1
344     5 -0.257210 30 -31 32 -33 450 -670 u=-1
345     5 -0.257210 30 -31 32 -33 670 -455 u=-1
346     0            30 -31 32 -33 455  u=1
347     0           -30             460 u=1
348     0            31             460 u=1
349     0         30 -31 -32        460 u=1
350     0         30 -31  33        460 u=1
c
c      universe 2
c
c       egg crate
```

```
401     0           -30             -400 u=2
402     0            31             -400 u=2
403     0         30 -31 -32        -400 u=2
404     0         30 -31  33        -400 u=2
405     5 -7.92          28     400 -670 u=2
406     5 -7.92              -23 400 -670 u=2
407     5 -7.92     -15 23 -28 400 -670 u=2
408     5 -7.92  20      23 -28 400 -670 u=2
409     5 -7.92          28     670 -460 u=2
410     5 -7.92              -23 670 -460 u=2
411     5 -7.92     -15 23 -28 670 -460 u=2
412     5 -7.92  20      23 -28 670 -460 u=2
c       boral and inside of egg crate and outside of fuel
414     0         15 -30 23 -28 400 -460 u=-2
415     0         31 -20 23 -28 400 -460 u=-2
416     0         30 -31 33 -28 400 -460 u=-2
417     0         30 -31 23 -32 400 -460 u=-2
c       fuel element
438     5 -1.48621  30 -31 32 -33     -420 u=2
439     2 -4.29251 30 -31 32 -33 420 -425 u=-2
440     0           30 -31 32 -33 425 -430 u=-2
441     5 -0.265322 30 -31 32 -33 430 -440 u=-2
442     5 -0.677510 30 -31 32 -33 440 -445 u=-2
443     5 -1.369221 30 -31 32 -33 445 -450 u=-2
444     5 -0.257210 30 -31 32 -33 450 -670 u=-2
445     5 -0.257210 30 -31 32 -33 670 -455 u=-2
446     0           30 -31 32 -33 455 u=2
447     0           -30             460 u=2
448     0            31             460 u=2
449     0         30 -31 -32        460 u=2
450     0         30 -31  33        460 u=2
c
c     universe 3
c
c       egg crate
501     0           -30             -400 u=3
502     0            31             -400 u=3
503     0         30 -31 -32        -400 u=3
504     0         30 -31  33        -400 u=3
505     5 -7.92          28     400 -670 u=3
506     5 -7.92              -23 400 -670 u=3
507     5 -7.92     -15 23 -28 400 -670 u=3
508     5 -7.92  20      23 -28 400 -670 u=3
509     5 -7.92          28     670 -460 u=3
510     5 -7.92              -23 670 -460 u=3
511     5 -7.92     -15 23 -28 670 -460 u=3
512     5 -7.92  20      23 -28 670 -460 u=3
c       boral and inside of egg crate and outside of fuel
514     0         15 -30 23 -28 400 -410 u=-3
515     0         31 -20 23 -28 400 -410 u=-3
516     0         30 -31 33 -28 400 -410 u=-3
517     0         30 -31 23 -32 400 -410 u=-3
518     0         15 -18 26 -28 410 -435 u=-3
519     0         19 -20 26 -28 410 -435 u=-3
522     6 -2.644 18 -19 27 -28 410 -435 u=-3
523     5 -7.92  18 -19 26 -27 410 -435 u=-3
526     0         15 -30 23 -26 410 -435 u=-3
527     0         31 -20 23 -26 410 -435 u=-3
528     0         30 -31 23 -32 410 -435 u=-3
529     0         30 -31 33 -26 410 -435 u=-3
530     0         15 -30 23 -28 435 -460 u=-3
531     0         31 -20 23 -28 435 -460 u=-3
532     0         30 -31 33 -28 435 -460 u=-3
533     0         30 -31 23 -32 435 -460 u=-3
c       fuel element
538     5 -1.48621  30 -31 32 -33     -420 u=3
539     2 -4.29251 30 -31 32 -33 420 -425 u=-3
540     0           30 -31 32 -33 425 -430 u=-3
541     5 -0.265322 30 -31 32 -33 430 -440 u=-3
```

```
542     5 -0.677510 30 -31 32 -33 440 -445 u=-3
543     5 -1.369221 30 -31 32 -33 445 -450 u=-3
544     5 -0.257210 30 -31 32 -33 450 -670 u=-3
545     5 -0.257210 30 -31 32 -33 670 -455 u=-3
546     0           30 -31 32 -33 455 u=3
547     0           -30             460 u=3
548     0            31             460 u=3
549     0        30 -31 -32         460 u=3
550     0        30 -31  33         460 u=3
c
c      universe 4
c
c       egg crate
601     0            -30              -400 u=4
602     0             31              -400 u=4
603     0         30 -31 -32          -400 u=4
604     0         30 -31  33          -400 u=4
605     5 -7.92           28      400 -670 u=4
606     5 -7.92               -23 400 -670 u=4
607     5 -7.92       -15 23 -28 400 -670 u=4
608     5 -7.92   20      23 -28 400 -670 u=4
609     5 -7.92           28      670 -460 u=4
610     5 -7.92               -23 670 -460 u=4
611     5 -7.92       -15 23 -28 670 -460 u=4
612     5 -7.92   20      23 -28 670 -460 u=4
c       boral and inside of egg crate and outside of fuel
614     0         15 -30 23 -28 400 -410 u=-4
615     0         31 -20 23 -28 400 -410 u=-4
616     0         30 -31 33 -28 400 -410 u=-4
617     0         30 -31 23 -32 400 -410 u=-4
620     0         15 -17 23 -24 410 -435 u=-4
621     0         15 -17 25 -28 410 -435 u=-4
624     6 -2.644 15 -16 24 -25 410 -435 u=-4
625     5 -7.92  16 -17 24 -25 410 -435 u=-4
626     0         17 -30 23 -28 410 -435 u=-4
627     0         31 -20 23 -28 410 -435 u=-4
628     0         30 -31 23 -32 410 -435 u=-4
629     0         30 -31 33 -28 410 -435 u=-4
630     0         15 -30 23 -28 435 -460 u=-4
631     0         31 -20 23 -28 435 -460 u=-4
632     0         30 -31 33 -28 435 -460 u=-4
633     0         30 -31 23 -32 435 -460 u=-4
c       fuel element
638     5 -1.48621  30 -31 32 -33     -420 u=4
639     2 -4.29251 30 -31 32 -33 420 -425 u=-4
640     0           30 -31 32 -33 425 -430 u=-4
641     5 -0.265322 30 -31 32 -33 430 -440 u=-4
642     5 -0.677510 30 -31 32 -33 440 -445 u=-4
643     5 -1.369221 30 -31 32 -33 445 -450 u=-4
644     5 -0.257210 30 -31 32 -33 450 -670 u=-4
645     5 -0.257210 30 -31 32 -33 670 -455 u=-4
646     0           30 -31 32 -33 455  u=4
647     0           -30             460 u=4
648     0            31             460 u=4
649     0        30 -31 -32         460 u=4
650     0        30 -31  33         460 u=4
c
c      universe 5
c
701     0                -400 u=5
702     5 -7.92 20 400 -670 u=5
704     5 -7.92 20 670 -460 u=5
705     0       -20 400 -460 u=5
706     0           460      u=5
c
c      universe 6
c
710     0                    -400 u=6
711     5 -7.92  -23 400 -670 u=6
```

```
713     5 -7.92   -23 670 -460 u=6
714     0          23 400 -460 u=6
715     0             460      u=6
c
c      universe 7
c
720     0                        -400 u=7
721     5 -7.92 20        23 400 -670 u=7
722     5 -7.92    -23       400 -670 u=7
724     5 -7.92 20        23 670 -460 u=7
725     5 -7.92    -23       670 -460 u=7
726     0          -20    23 400 -460 u=7
727     0                    460      u=7
c
c      universe 8
c
730     0                          -400 u=8
731     0        15            400 -460 u=8
739     5 -7.92     -15        400 -670 u=8
740     5 -7.92     -15        670 -460 u=8
741     0                      460      u=8
c
c      universe 9
c
745     0        15      23    400 -460 u=9
753     0                          -400 u=9
754     5 -7.92     -15 23     400 -670 u=9
755     5 -7.92     -23        400 -670 u=9
756     5 -7.92     -15 23     670 -460 u=9
757     5 -7.92     -23        670 -460 u=9
758     0                      460      u=9
c
c      universe 10
c
760     0        15        -28 400 -460 u=10
772     0                          -400 u=10
773     5 -7.92     -15    -28 400 -670 u=10
774     5 -7.92         28     400 -670 u=10
775     5 -7.92     -15    -28 670 -460 u=10
776     5 -7.92         28     670 -460 u=10
777     0                      460      u=10
c
c      universe 11
c
780     0                  -28 400 -460 u=11
788     0                          -400 u=11
789     5 -7.92         28     400 -670 u=11
790     5 -7.92         28     670 -460 u=11
791     0                      460      u=11
c
c      universe 12
c
795     0           -20    -28 400 -460 u=12
803     0                          -400 u=12
804     5 -7.92     -20 28     400 -670 u=12
805     5 -7.92  20            400 -670 u=12
806     5 -7.92     -20 28     670 -460 u=12
807     5 -7.92  20            670 -460 u=12
808     0                      460      u=12
c
c      universe 13
c
c      810   0     -670  u=13
c      811   0      670  u=13
c      storage locations
c
201     0  -301   -106 212 620 -675
202     0  -301   106 -107 212 620 -675
           fill=6 (-8.2423 90.6653 0.0)
```

```
203    0   -301  107 -108 212 620 -675
               fill=6 ( 8.2423 90.6653 0.0)
204    0   -301  108 212 620 -675
c
205    0   -301 -104 211 620 -675
206    0   -301 104 -105 211 620 -675
               fill=6 (-41.2115 74.1807 0.0)
207    0   -301 105 -106 211 -212 620 -675
               fill=7 (-24.7269 74.1807 0.0)
c
101    0   106 -107 211 -212  620 -675
               fill=2 (-8.2423 74.1807 0.0)
102    0   107 -108 211 -212  620 -675
               fill=4 ( 8.2423 74.1807 0.0)
c
208    0   -301 108 -109 211 -212 620 -675
               fill=9 (24.7269 74.1807 0.0)
209    0   -301 109 -110 211 620 -675
               fill=6 (41.2115 74.1807 0.0)
210    0   -301 110 211 620 -675
c
211    0   -301 -103 210 620 -675
212    0   -301 103 -104 210 -211 620 -675
               fill=7 (-57.6961 57.6961 0.0)
c
103    0   104 -105 210 -211  620 -675
               fill=2 (-41.2115 57.6961 0.0)
104    0   105 -106 210 -211  620 -675
               fill=4 (-24.7269 57.6961 0.0)
105    0   106 -107 210 -211  620 -675
               fill=1 (-8.2423 57.6961 0.0)
106    0   107 -108 210 -211  620 -675
               fill=1 ( 8.2423 57.6961 0.0)
107    0   108 -109 210 -211  620 -675
               fill=4 ( 24.7269 57.6961 0.0)
108    0   109 -110 210 -211  620 -675
               fill=4 ( 41.2115 57.6961 0.0)
c
213    0   -301 110 -111 210 -211 620 -675
               fill=9 ( 57.6961 57.6961 0.0)
214    0   -301 111 210 620 -675
c
215    0   -301 -103 209 -210 620 -675
               fill=5 (-74.1807 41.2115 0.0)
c
109    0   103 -104 209 -210  620 -675
               fill=2 (-57.6961 41.2115 0.0)
110    0   104 -105 209 -210  620 -675
               fill=1 (-41.2115 41.2115 0.0)
111    0   105 -106 209 -210  620 -675
               fill=1 (-24.7269 41.2115 0.0)
112    0   106 -107 209 -210  620 -675
               fill=1 (-8.2423 41.2115 0.0)
113    0   107 -108 209 -210  620 -675
               fill=1 ( 8.2423 41.2115 0.0)
114    0   108 -109 209 -210  620 -675
               fill=1 ( 24.7269 41.2115 0.0)
115    0   109 -110 209 -210  620 -675
               fill=1 ( 41.2115 41.2115 0.0)
116    0   110 -111 209 -210  620 -675
               fill=4 ( 57.6961 41.2115 0.0)
c
216    0   -301 111 209 -210 620 -675
               fill=8 (74.1807 41.2115 0.0)
c
217    0   -301 -102 208 620 -675
218    0   -301 102 -103 208 -209 620 -675
               fill=7 (-74.1807 24.7269 0.0)
c
```

```
117    0   103 -104 208 -209  620 -675
           fill=3 (-57.6961 24.7269 0.0)
118    0   104 -105 208 -209  620 -675
           fill=1 (-41.2115 24.7269 0.0)
119    0   105 -106 208 -209  620 -675
           fill=1 (-24.7269 24.7269 0.0)
120    0   106 -107 208 -209  620 -675
           fill=1 (-8.2423 24.7269 0.0)
121    0   107 -108 208 -209  620 -675
           fill=1 ( 8.2423 24.7269 0.0)
122    0   108 -109 208 -209  620 -675
           fill=1 ( 24.7269 24.7269 0.0)
123    0   109 -110 208 -209  620 -675
           fill=1 ( 41.2115 24.7269 0.0)
124    0   110 -111 208 -209  620 -675
           fill=1 ( 57.6961 24.7269 0.0)
c
219    0   -301 111 -112 208 -209 620 -675
           fill=9 (74.1807 24.7269 0.0)
220    0   -301 112 208 620 -675
c
221    0   -301 -102 207 -208 620 -675
           fill=5 (-90.6653 8.2423 0.0)
c
125    0   102 -103 207 -208  620 -675
           fill=2 (-74.1807 8.2423 0.0)
126    0   103 -104 207 -208  620 -675
           fill=1 (-57.6961 8.2423 0.0)
127    0   104 -105 207 -208  620 -675
           fill=1 (-41.2115 8.2423 0.0)
128    0   105 -106 207 -208  620 -675
           fill=1 (-24.7269 8.2423 0.0)
129    0   106 -107 207 -208  620 -675
           fill=1 (-8.2423 8.2423 0.0)
130    0   107 -108 207 -208  620 -675
           fill=1 ( 8.2423 8.2423 0.0)
131    0   108 -109 207 -208  620 -675
           fill=1 ( 24.7269 8.2423 0.0)
132    0   109 -110 207 -208  620 -675
           fill=1 ( 41.2115 8.2423 0.0)
133    0   110 -111 207 -208  620 -675
           fill=1 ( 57.6961 8.2423 0.0)
134    0   111 -112 207 -208  620 -675
           fill=4 ( 74.1807 8.2423 0.0)
c
222    0   -301 112 207 -208 620 -675
           fill=8 (90.6653 8.2423 0.0)
c
223    0   -301 -102 206 -207 620 -675
           fill=5 (-90.6653 -8.2423 0.0)
c
135    0   102 -103 206 -207  620 -675
           fill=3 (-74.1807 -8.2423 0.0)
136    0   103 -104 206 -207  620 -675
           fill=1 (-57.6961 -8.2423 0.0)
137    0   104 -105 206 -207  620 -675
           fill=1 (-41.2115 -8.2423 0.0)
138    0   105 -106 206 -207  620 -675
           fill=1 (-24.7269 -8.2423 0.0)
139    0   106 -107 206 -207  620 -675
           fill=1 (-8.2423 -8.2423 0.0)
140    0   107 -108 206 -207  620 -675
           fill=1 ( 8.2423 -8.2423 0.0)
141    0   108 -109 206 -207  620 -675
           fill=1 ( 24.7269 -8.2423 0.0)
142    0   109 -110 206 -207  620 -675
           fill=1 ( 41.2115 -8.2423 0.0)
143    0   110 -111 206 -207  620 -675
           fill=1 ( 57.6961 -8.2423 0.0)
```

```
144    0  111 -112 206 -207  620 -675
          fill=1 ( 74.1807 -8.2423 0.0)
c
224    0  -301 112 206 -207 620 -675
          fill=8 (90.6653 -8.2423 0.0)
c
225    0  -301 -102 -206 620 -675
226    0  -301 102 -103 205 -206 620 -675
          fill=12 (-74.1807 -24.7269 0.0)
c
145    0  103 -104 205 -206  620 -675
          fill=3 (-57.6961 -24.7269 0.0)
146    0  104 -105 205 -206  620 -675
          fill=1 (-41.2115 -24.7269 0.0)
147    0  105 -106 205 -206  620 -675
          fill=1 (-24.7269 -24.7269 0.0)
148    0  106 -107 205 -206  620 -675
          fill=1 (-8.2423  -24.7269 0.0)
149    0  107 -108 205 -206  620 -675
          fill=1 ( 8.2423  -24.7269 0.0)
150    0  108 -109 205 -206  620 -675
          fill=1 ( 24.7269 -24.7269 0.0)
151    0  109 -110 205 -206  620 -675
          fill=1 ( 41.2115 -24.7269 0.0)
152    0  110 -111 205 -206  620 -675
          fill=1 ( 57.6961 -24.7269 0.0)
c
227    0  -301 111 -112 205 -206 620 -675
          fill=10 (74.1807 -24.7269 0.0)
228    0  -301 112 -206 620 -675
c
229    0  -301 -103 204 -205 620 -675
          fill=5 (-74.1807 -41.2115 0.0)
c
153    0  103 -104 204 -205  620 -675
          fill=3 (-57.6961 -41.2115 0.0)
154    0  104 -105 204 -205  620 -675
          fill=1 (-41.2115 -41.2115 0.0)
155    0  105 -106 204 -205  620 -675
          fill=1 (-24.7269 -41.2115 0.0)
156    0  106 -107 204 -205  620 -675
          fill=1 (-8.2423  -41.2115 0.0)
157    0  107 -108 204 -205  620 -675
          fill=1 ( 8.2423  -41.2115 0.0)
158    0  108 -109 204 -205  620 -675
          fill=1 ( 24.7269 -41.2115 0.0)
159    0  109 -110 204 -205  620 -675
          fill=1 ( 41.2115 -41.2115 0.0)
160    0  110 -111 204 -205  620 -675
          fill=1 ( 57.6961 -41.2115 0.0)
c
230    0  -301 111 204 -205 620 -675
          fill=8 (74.1807 -41.2115 0.0)
c
231    0  -301 -103 -204 620 -675
232    0  -301 103 -104 203 -204 620 -675
          fill=12 (-57.6961 -57.6961 0.0)
c
161    0  104 -105 203 -204  620 -675
          fill=3 (-41.2115 -57.6961 0.0)
162    0  105 -106 203 -204  620 -675
          fill=1 (-24.7269 -57.6961 0.0)
163    0  106 -107 203 -204  620 -675
          fill=1 (-8.2423 -57.6961 0.0)
164    0  107 -108 203 -204  620 -675
          fill=1 ( 8.2423 -57.6961 0.0)
165    0  108 -109 203 -204  620 -675
          fill=1 ( 24.7269 -57.6961 0.0)
166    0  109 -110 203 -204  620 -675
```

```
          fill=1 ( 41.2115 -57.6961 0.0)
c
233    0  -301 110 -111 203 -204 620 -675
          fill=10 ( 57.6961 -57.6961 0.0)
234    0  -301 111 -204 620 -675
c
235    0  -301 -104 -203 620 -675
236    0  -301 104 -105 -203 620 -675
          fill=11 (-41.2115 -74.1807 0.0)
237    0  -301 105 -106 202 -203 620 -675
          fill=12 (-24.7269 -74.1807 0.0)
c
167    0  106 -107 202 -203  620 -675
          fill=3 (-8.2423 -74.1807 0.0)
168    0  107 -108 202 -203  620 -675
          fill=1 ( 8.2423 -74.1807 0.0)
c
238    0  -301 108 -109 202 -203 620 -675
          fill=10 (24.7269 -74.1807 0.0)
239    0  -301 109 -110 -203 620 -675
          fill=11 (41.2115 -74.1807 0.0)
240    0  -301 110 -203 620 -675
c
241    0  -301  -106 -202 620 -675
242    0  -301  106 -107 -202 620 -675
          fill=11 (-8.2423 -90.6653 0.0)
243    0  -301  107 -108 -202 620 -675
          fill=11 ( 8.2423 -90.6653 0.0)
244    0  -301  108 -202 620 -675
c
821    5 -7.92 301 -302 610 -630  $ MPC shell
822    0       302 -501 610 -630  $ Air gap
823    5 -7.92 301 -302 630 -670  $ MPC shell
824    0       302 -501 630 -670  $ Air gap
825    5 -7.92 301 -302 670 -675  $ MPC shell
826    0       302 -501 670 -675  $ Air gap
827    5 -7.92 301 -302 675 -680  $ MPC shell
828    0       301 -302 680 -685  $ Air gap
829    0       302 -501 675 -685  $ Air gap
c
830    5 -7.92     -301 610 -620  $ MPC baseplate
840    5 -7.92     -301 675 -680  $ MPC lid (both)
850    0           -301 680 -685  $ Air gap
c      OVERPACK \/ \/ \/ \/ \/
c
1001   8 -7.82 501 -502 630 -670 $ steel shell
1002   8 -7.82 502 -503 630 -670 $ steel shell
1003   8 -7.82 503 -504 630 -670 $ steel shell
1004   8 -7.82 504 -505 630 -670 $ steel shell
1005   8 -7.82 505 -506 630 -670 $ steel shell
1006   8 -7.82 506 -507 630 -670 $ steel shell
1007   8 -7.82 507 -508 630 -670 $ steel shell
1008   0       508 -509 630 -670  fill=20
1009   0       509 -510 630 -670  fill=20
1010   0       510 -511 630 -670  fill=20
1011   0       511 -512 630 -670  fill=20
1012   8 -7.82 512 -513 630 -670 $ outer steel shell
c      1012  8 -7.82  512 -513 640 -670             $ outer steel shell
c      1013  0        512 -513 630 -640  1103 -1102 $ air in pocket trun.
c      1014  0        512 -513 630 -640  2103 -2102 $ air in pocket trun.
c      1015  8 -7.82  512 -513 630 -640  1102  2102 $ outer steel shell
c      1016  8 -7.82  512 -513 630 -640  1102 -2103 $ outer steel shell
c      1017  8 -7.82  512 -513 630 -640 -1103 -2103 $ outer steel shell
c      1018  8 -7.82  512 -513 630 -640 -1103  2102 $ outer steel shell
c
c       steel spines and holtite
10101   8 -7.82   2000 -2002  645 -660   1000  u=20 $ steel spine
10102   7 -1.61   2002 -2011  645 -660   2000  u=20 $ holtite
10103   8 -7.82   2011 -2012  645 -660   2000  u=20 $ steel spine
```

```
10104    7 -1.61    2012 -2021  645 -660    2000   u=20 $ holtite
10105    8 -7.82    2021 -2022  645 -660    2000   u=20 $ steel spine
10106    7 -1.61    2022 -2031  645 -660    2000   u=20 $ holtite
10107    8 -7.82    2031 -2032  645 -660    2000   u=20 $ steel spine
10108    7 -1.61    2032 -2041  645 -660    2000   u=20 $ holtite
10109    8 -7.82    2041 -2042  645 -660    2000   u=20 $ steel spine
10110    7 -1.61    2042 -2051  645 -660    2000   u=20 $ holtite
10111    8 -7.82    2051 -2052  645 -660    2000   u=20 $ steel spine
10112    7 -1.61    2052 -2061  645 -660    2000   u=20 $ holtite
10113    8 -7.82    2061 -2062  645 -660    2000   u=20 $ steel spine
10114    7 -1.61    2062 -2071  645 -660    2000   u=20 $ holtite
10115    8 -7.82    2071 -2072  645 -660    2000   u=20 $ steel spine
10116    7 -1.61    2072 -2081  645 -660    2000   u=20 $ holtite
10117    8 -7.82    2081 -2082  645 -660    2000   u=20 $ steel spine
10118    7 -1.61    2082 -2091  645 -660    2000   u=20 $ holtite
10119    8 -7.82    2091 -2092  645 -660    2000   u=20 $ steel spine
10120    7 -1.61    2092  1002  645 -660    2000   u=20 $ holtite
10121    8 -7.82    1000 -1002  645 -660    2000   u=20 $ steel spine
c
10122    8 -7.82    1001 -1000  645 -660    2000   u=20 $ steel spine
10123    7 -1.61    1012 -1001  645 -660    2000   u=20 $ holtite
10124    8 -7.82    1011 -1012  645 -660    2000   u=20 $ steel spine
10125    7 -1.61    1022 -1011  645 -660    2000   u=20 $ holtite
10126    8 -7.82    1021 -1022  645 -660    2000   u=20 $ steel spine
10127    7 -1.61    1032 -1021  645 -660    2000   u=20 $ holtite
10128    8 -7.82    1031 -1032  645 -660    2000   u=20 $ steel spine
10129    7 -1.61    1042 -1031  645 -660    2000   u=20 $ holtite
10130    8 -7.82    1041 -1042  645 -660    2000   u=20 $ steel spine
10131    7 -1.61    1052 -1041  645 -660    2000   u=20 $ holtite
10132    8 -7.82    1051 -1052  645 -660    2000   u=20 $ steel spine
10133    7 -1.61    1062 -1051  645 -660    2000   u=20 $ holtite
10134    8 -7.82    1061 -1062  645 -660    2000   u=20 $ steel spine
10135    7 -1.61    1072 -1061  645 -660    2000   u=20 $ holtite
10136    8 -7.82    1071 -1072  645 -660    2000   u=20 $ steel spine
10137    7 -1.61    1082 -1071  645 -660    2000   u=20 $ holtite
10138    8 -7.82    1081 -1082  645 -660    2000   u=20 $ steel spine
10139    7 -1.61    1092 -1081  645 -660    2000   u=20 $ holtite
10140    8 -7.82    1091 -1092  645 -660    2000   u=20 $ steel spine
10141    7 -1.61    2002 -1091  645 -660    2000   u=20 $ holtite
10142    8 -7.82    2000 -2002  645 -660   -1000   u=20 $ steel spine
c
10143    8 -7.82    2001 -2000  645 -660   -1000   u=20 $ steel spine
10144    7 -1.61    2012 -2001  645 -660   -2000   u=20 $ holtite
10145    8 -7.82    2011 -2012  645 -660   -2000   u=20 $ steel spine
10146    7 -1.61    2022 -2011  645 -660   -2000   u=20 $ holtite
10147    8 -7.82    2021 -2022  645 -660   -2000   u=20 $ steel spine
10148    7 -1.61    2032 -2021  645 -660   -2000   u=20 $ holtite
10149    8 -7.82    2031 -2032  645 -660   -2000   u=20 $ steel spine
10150    7 -1.61    2042 -2031  645 -660   -2000   u=20 $ holtite
10151    8 -7.82    2041 -2042  645 -660   -2000   u=20 $ steel spine
10152    7 -1.61    2052 -2041  645 -660   -2000   u=20 $ holtite
10153    8 -7.82    2051 -2052  645 -660   -2000   u=20 $ steel spine
10154    7 -1.61    2062 -2051  645 -660   -2000   u=20 $ holtite
10155    8 -7.82    2061 -2062  645 -660   -2000   u=20 $ steel spine
10156    7 -1.61    2072 -2061  645 -660   -2000   u=20 $ holtite
10157    8 -7.82    2071 -2072  645 -660   -2000   u=20 $ steel spine
10158    7 -1.61    2082 -2071  645 -660   -2000   u=20 $ holtite
10159    8 -7.82    2081 -2082  645 -660   -2000   u=20 $ steel spine
10160    7 -1.61    2092 -2081  645 -660   -2000   u=20 $ holtite
10161    8 -7.82    2091 -2092  645 -660   -2000   u=20 $ steel spine
10162    7 -1.61   -1001 -2091  645 -660   -2000   u=20 $ holtite
10163    8 -7.82    1001 -1000  645 -660   -2000   u=20 $ steel spine
c
10164    8 -7.82    1000 -1002  645 -660   -2000   u=20 $ steel spine
10165    7 -1.61    1002 -1011  645 -660   -2000   u=20 $ holtite
10166    8 -7.82    1011 -1012  645 -660   -2000   u=20 $ steel spine
10167    7 -1.61    1012 -1021  645 -660   -2000   u=20 $ holtite
10168    8 -7.82    1021 -1022  645 -660   -2000   u=20 $ steel spine
10169    7 -1.61    1022 -1031  645 -660   -2000   u=20 $ holtite
```

```
10170   8 -7.82   1031 -1032  645 -660  -2000  u=20 $ steel spine
10171   7 -1.61   1032 -1041  645 -660  -2000  u=20 $ holtite
10172   8 -7.82   1041 -1042  645 -660  -2000  u=20 $ steel spine
10173   7 -1.61   1042 -1051  645 -660  -2000  u=20 $ holtite
10174   8 -7.82   1051 -1052  645 -660  -2000  u=20 $ steel spine
10175   7 -1.61   1052 -1061  645 -660  -2000  u=20 $ holtite
10176   8 -7.82   1061 -1062  645 -660  -2000  u=20 $ steel spine
10177   7 -1.61   1062 -1071  645 -660  -2000  u=20 $ holtite
10178   8 -7.82   1071 -1072  645 -660  -2000  u=20 $ steel spine
10179   7 -1.61   1072 -1081  645 -660  -2000  u=20 $ holtite
10180   8 -7.82   1081 -1082  645 -660  -2000  u=20 $ steel spine
10181   7 -1.61   1082 -1091  645 -660  -2000  u=20 $ holtite
10182   8 -7.82   1091 -1092  645 -660  -2000  u=20 $ steel spine
10183   7 -1.61   1092 -2001  645 -660  -2000  u=20 $ holtite
10184   8 -7.82   2001 -2000  645 -660   1000  u=20 $ steel spine
c
c    10201   8 -7.82   2000 -2002  635 -645   1000  u=20 $ steel spine
c    10202   7 -1.61   2002 -2011  635 -645   2000  u=20 $ holtite
10202   7 -1.61   2101 -2011  635 -645   2000  u=20 $ holtite
10203   8 -7.82   2011 -2012  635 -645   2000  u=20 $ steel spine
10204   7 -1.61   2012 -2021  635 -645   2000  u=20 $ holtite
10205   8 -7.82   2021 -2022  635 -645   2000  u=20 $ steel spine
10206   7 -1.61   2022 -2031  635 -645   2000  u=20 $ holtite
10207   8 -7.82   2031 -2032  635 -645   2000  u=20 $ steel spine
10208   7 -1.61   2032 -2041  635 -645   2000  u=20 $ holtite
10209   8 -7.82   2041 -2042  635 -645   2000  u=20 $ steel spine
10210   7 -1.61   2042 -2051  635 -645   2000  u=20 $ holtite
10211   8 -7.82   2051 -2052  635 -645   2000  u=20 $ steel spine
10212   7 -1.61   2052 -2061  635 -645   2000  u=20 $ holtite
10213   8 -7.82   2061 -2062  635 -645   2000  u=20 $ steel spine
10214   7 -1.61   2062 -2071  635 -645   2000  u=20 $ holtite
10215   8 -7.82   2071 -2072  635 -645   2000  u=20 $ steel spine
10216   7 -1.61   2072 -2081  635 -645   2000  u=20 $ holtite
10217   8 -7.82   2081 -2082  635 -645   2000  u=20 $ steel spine
10218   7 -1.61   2082 -2091  635 -645   2000  u=20 $ holtite
10219   8 -7.82   2091 -2092  635 -645   2000  u=20 $ steel spine
c    10220   7 -1.61   2092  1101  635 -645   2000  u=20 $ holtite
10220   7 -1.61   2092  1002  635 -645   2000  u=20 $ holtite
10221   8 -7.82   1000 -1002  635 -645   2000  u=20 $ steel spine
c
10222   8 -7.82   1001 -1000  635 -645   2000  u=20 $ steel spine
10223   7 -1.61   1012 -1001  635 -645   2000  u=20 $ holtite
c    10223   7 -1.61   1012 -1104  635 -645   2000  u=20 $ holtite
10224   8 -7.82   1011 -1012  635 -645   2000  u=20 $ steel spine
10225   7 -1.61   1022 -1011  635 -645   2000  u=20 $ holtite
10226   8 -7.82   1021 -1022  635 -645   2000  u=20 $ steel spine
10227   7 -1.61   1032 -1021  635 -645   2000  u=20 $ holtite
10228   8 -7.82   1031 -1032  635 -645   2000  u=20 $ steel spine
10229   7 -1.61   1042 -1031  635 -645   2000  u=20 $ holtite
10230   8 -7.82   1041 -1042  635 -645   2000  u=20 $ steel spine
10231   7 -1.61   1052 -1041  635 -645   2000  u=20 $ holtite
10232   8 -7.82   1051 -1052  635 -645   2000  u=20 $ steel spine
10233   7 -1.61   1062 -1051  635 -645   2000  u=20 $ holtite
10234   8 -7.82   1061 -1062  635 -645   2000  u=20 $ steel spine
10235   7 -1.61   1072 -1061  635 -645   2000  u=20 $ holtite
10236   8 -7.82   1071 -1072  635 -645   2000  u=20 $ steel spine
10237   7 -1.61   1082 -1071  635 -645   2000  u=20 $ holtite
10238   8 -7.82   1081 -1082  635 -645   2000  u=20 $ steel spine
10239   7 -1.61   1092 -1081  635 -645   2000  u=20 $ holtite
10240   8 -7.82   1091 -1092  635 -645   2000  u=20 $ steel spine
10241   7 -1.61   2101 -1091  635 -645   2000  u=20 $ holtite
c    10241   7 -1.61   2002 -1091  635 -645   2000  u=20 $ holtite
c    10242   8 -7.82   2000 -2002  635 -645  -1000  u=20 $ steel spine
c
c    10243   8 -7.82   2001 -2000  635 -645  -1000  u=20 $ steel spine
c    10244   7 -1.61   2012 -2001  635 -645  -2000  u=20 $ holtite
10244   7 -1.61   2012 -2104  635 -645  -2000  u=20 $ holtite
10245   8 -7.82   2011 -2012  635 -645  -2000  u=20 $ steel spine
10246   7 -1.61   2022 -2011  635 -645  -2000  u=20 $ holtite
```

```
10247   8 -7.82   2021 -2022  635 -645  -2000  u=20 $ steel spine
10248   7 -1.61   2032 -2021  635 -645  -2000  u=20 $ holtite
10249   8 -7.82   2031 -2032  635 -645  -2000  u=20 $ steel spine
10250   7 -1.61   2042 -2031  635 -645  -2000  u=20 $ holtite
10251   8 -7.82   2041 -2042  635 -645  -2000  u=20 $ steel spine
10252   7 -1.61   2052 -2041  635 -645  -2000  u=20 $ holtite
10253   8 -7.82   2051 -2052  635 -645  -2000  u=20 $ steel spine
10254   7 -1.61   2062 -2051  635 -645  -2000  u=20 $ holtite
10255   8 -7.82   2061 -2062  635 -645  -2000  u=20 $ steel spine
10256   7 -1.61   2072 -2061  635 -645  -2000  u=20 $ holtite
10257   8 -7.82   2071 -2072  635 -645  -2000  u=20 $ steel spine
10258   7 -1.61   2082 -2071  635 -645  -2000  u=20 $ holtite
10259   8 -7.82   2081 -2082  635 -645  -2000  u=20 $ steel spine
10260   7 -1.61   2092 -2081  635 -645  -2000  u=20 $ holtite
10261   8 -7.82   2091 -2092  635 -645  -2000  u=20 $ steel spine
c    10262   7 -1.61  -1104 -2091  635 -645  -2000  u=20 $ holtite
10262   7 -1.61  -1001 -2091  635 -645  -2000  u=20 $ holtite
10263   8 -7.82   1001 -1000  635 -645  -2000  u=20 $ steel spine
c
10264   8 -7.82   1000 -1002  635 -645  -2000  u=20 $ steel spine
10265   7 -1.61   1002 -1011  635 -645  -2000  u=20 $ holtite
c    10265   7 -1.61   1101 -1011  635 -645  -2000  u=20 $ holtite
10266   8 -7.82   1011 -1012  635 -645  -2000  u=20 $ steel spine
10267   7 -1.61   1012 -1021  635 -645  -2000  u=20 $ holtite
10268   8 -7.82   1021 -1022  635 -645  -2000  u=20 $ steel spine
10269   7 -1.61   1022 -1031  635 -645  -2000  u=20 $ holtite
10270   8 -7.82   1031 -1032  635 -645  -2000  u=20 $ steel spine
10271   7 -1.61   1032 -1041  635 -645  -2000  u=20 $ holtite
10272   8 -7.82   1041 -1042  635 -645  -2000  u=20 $ steel spine
10273   7 -1.61   1042 -1051  635 -645  -2000  u=20 $ holtite
10274   8 -7.82   1051 -1052  635 -645  -2000  u=20 $ steel spine
10275   7 -1.61   1052 -1061  635 -645  -2000  u=20 $ holtite
10276   8 -7.82   1061 -1062  635 -645  -2000  u=20 $ steel spine
10277   7 -1.61   1062 -1071  635 -645  -2000  u=20 $ holtite
10278   8 -7.82   1071 -1072  635 -645  -2000  u=20 $ steel spine
10279   7 -1.61   1072 -1081  635 -645  -2000  u=20 $ holtite
10280   8 -7.82   1081 -1082  635 -645  -2000  u=20 $ steel spine
10281   7 -1.61   1082 -1091  635 -645  -2000  u=20 $ holtite
10282   8 -7.82   1091 -1092  635 -645  -2000  u=20 $ steel spine
10283   7 -1.61   1092 -2104  635 -645  -2000  u=20 $ holtite
c    10283   7 -1.61   1092 -2001  635 -645  -2000  u=20 $ holtite
c    10284   8 -7.82   2001 -2000  635 -645   1000  u=20 $ steel spine
c
11000   0         660 -665   u=20 $ foam
11001   8 -7.82   665        u=20 $ item 17 top
c    11002   8 -7.82   1101  2101  -635  u=20 $ item 17 bot
c    11003   8 -7.82  -2104  1101  -635  u=20 $ item 17 bot
c    11004   8 -7.82  -1104 -2104  -635  u=20 $ item 17 bot
c    11005   8 -7.82   2101 -1104  -635  u=20 $ item 17 bot
11002   8 -7.82         2101  -635  u=20 $ item 17 bot
11003   8 -7.82  -2104        -635  u=20 $ item 17 bot
c
c    pocket trunion
c
c    11101   8 -7.82  -514  1104 -1101 -640  u=20 $ pocket trunion before hole
c    11102   8 -7.82   514  1104 -1103 -640  u=20 $ steel on side of hole
c    11103   8 -7.82   514  1103 -1102 -640  u=20 $ hole
c    11104   8 -7.82   514  1102 -1101 -640  u=20 $ steel on side of hole
c    11105   8 -7.82        1104 -1101  640 -645  u=20 $ steel above hole
c
11111   8 -7.82  -514  2104 -2101 -640  u=20 $ pocket trunion before hole
11112   8 -7.82   514  2104 -2103 -640  u=20 $ steel on side of hole
11113   8 -7.82   514  2103 -2102 -640  u=20 $ hole
11114   8 -7.82   514  2102 -2101 -640  u=20 $ steel on side of hole
11115   8 -7.82        2104 -2101  640 -645  u=20 $ steel above hole
c
c     overpack base plate
c
2000  8 -7.82     -301 600 -610
```

```
2001  8 -7.82 301 -501 600 -610
2002  8 -7.82 501 -502 600 -630
2003  8 -7.82 502 -503 600 -630
2004  8 -7.82 503 -504 600 -630
2005  8 -7.82 504 -505 600 -630
2006  8 -7.82 505 -506 600 -630
2007  8 -7.82 506 -515 600 -630
2017  9 -1.17e-3   515 -507 600 -630 fill=27
2008  9 -1.17e-3   507 -508 600 -630 fill=27
2009  9 -1.17e-3   508 -509 600 -630
2010  9 -1.17e-3   509 -510 600 -630
2011  9 -1.17e-3   510 -511 600 -630
2012  9 -1.17e-3   511 -512 600 -630
2013  9 -1.17e-3   512 -513 600 -630
c
c     overpack lid
c
3000  8 -7.82     -301 685 -695
3001  8 -7.82 301 -501 685 -695
3002  8 -7.82 501 -502 685 -695
3003  8 -7.82 502 -503 685 -695
3004  8 -7.82 503 -504 685 -695
3005  8 -7.82 504 -505 685 -695
3006  8 -7.82 505 -518 685 -695
3007  9 -1.17e-3   518 -507 685 -695 fill=28
3008  9 -1.17e-3   507 -508 685 -695 fill=28
3009  9 -1.17e-3   508 -509 685 -695
3010  9 -1.17e-3   509 -510 685 -695
3011  9 -1.17e-3   510 -511 685 -695
3012  9 -1.17e-3   511 -512 685 -695
3013  9 -1.17e-3   512 -513 685 -695
c
3022  8 -7.82      501 -502 675 -685
3023  8 -7.82      502 -503 675 -685
3024  8 -7.82      503 -504 675 -685
3025  8 -7.82      504 -505 675 -685
3026  8 -7.82      505 -506 675 -677
3126  8 -7.82      505 -518 677 -685
3027  8 -7.82      506 -507 675 -676
3127  8 -7.82      506 -516 676 -677
3227  9 -1.17e-3   516 -507 676 -677
3327  9 -1.17e-3   518 -507 677 -685 fill=28
3028  8 -7.82      507 -508 675 -676
3128  9 -1.17e-3   507 -508 676 -677
3228  9 -1.17e-3   507 -508 677 -685 fill=28
3029  8 -7.82      508 -517 675 -676
3129  9 -1.17e-3   517 -509 675 -676
3229  9 -1.17e-3   508 -509 676 -685
3030  9 -1.17e-3   509 -510 675 -685
3031  9 -1.17e-3   510 -511 675 -685
3032  9 -1.17e-3   511 -512 675 -685
3033  9 -1.17e-3   512 -513 675 -685
c
3042  8 -7.82      501 -502 670 -675
3043  8 -7.82      502 -503 670 -675
3044  8 -7.82      503 -504 670 -675
3045  8 -7.82      504 -505 670 -675
3046  8 -7.82      505 -506 670 -675
3047  8 -7.82      506 -507 670 -675
3048  8 -7.82      507 -508 670 -675
3049  8 -7.82      508 -517 670 -675
3149  9 -1.17e-3   517 -509 670 -675
3050  9 -1.17e-3   509 -510 670 -675
3051  9 -1.17e-3   510 -511 670 -675
3052  9 -1.17e-3   511 -512 670 -675
3053  9 -1.17e-3   512 -513 670 -675
c
c     surrounding air regions
9000  9 -1.17e-3       -301 695 -723  fill=26
```

```
9001  9 -1.17e-3   301 -501 695 -723  fill=26
9002  9 -1.17e-3   501 -502 695 -723  fill=26
9003  9 -1.17e-3   502 -503 695 -723  fill=26
9004  9 -1.17e-3   503 -504 695 -723  fill=26
9005  9 -1.17e-3   504 -505 695 -723  fill=26
9006  9 -1.17e-3   505 -506 695 -723  fill=26
9007  9 -1.17e-3   506 -507 695 -723  fill=28
9008  9 -1.17e-3   507 -508 695 -723  fill=28
9009  9 -1.17e-3   508 -509 695 -723
9010  9 -1.17e-3   509 -510 695 -723
9011  9 -1.17e-3   510 -511 695 -723
9012  9 -1.17e-3   511 -512 695 -723
9013  9 -1.17e-3   512 -513 695 -723
c
9100  9 -1.17e-3       -301 701 -600  fill=25
9101  9 -1.17e-3   301 -501 701 -600  fill=25
9102  9 -1.17e-3   501 -502 701 -600  fill=25
9103  9 -1.17e-3   502 -503 701 -600  fill=25
9104  9 -1.17e-3   503 -504 701 -600  fill=25
9105  9 -1.17e-3   504 -505 701 -600  fill=25
9106  9 -1.17e-3   505 -506 701 -600  fill=25
9107  9 -1.17e-3   506 -507 701 -600  fill=27
9108  9 -1.17e-3   507 -508 701 -600  fill=27
9109  9 -1.17e-3   508 -509 701 -600
9110  9 -1.17e-3   509 -510 701 -600
9111  9 -1.17e-3   510 -511 701 -600
9112  9 -1.17e-3   511 -512 701 -600
9113  9 -1.17e-3   512 -513 701 -600
c
9200  9 -1.17e-3   513 -521 701 -630
9201  9 -1.17e-3   521 -522 701 -630
9202  9 -1.17e-3   522 -523 701 -630
9203  9 -1.17e-3   523 -524 701 -630
9204  9 -1.17e-3   524 -525 701 -630
9205  9 -1.17e-3   525 -526 701 -630
9206  9 -1.17e-3   526 -527 701 -630
c
9210  9 -1.17e-3   513 -521 630 -670
9211  9 -1.17e-3   521 -522 630 -670
9212  9 -1.17e-3   522 -523 630 -670
9213  9 -1.17e-3   523 -524 630 -670
9214  9 -1.17e-3   524 -525 630 -670
9215  9 -1.17e-3   525 -526 630 -670
9216  9 -1.17e-3   526 -527 630 -670
c
9220  9 -1.17e-3   513 -521 670 -675
9221  9 -1.17e-3   521 -522 670 -675
9222  9 -1.17e-3   522 -523 670 -675
9223  9 -1.17e-3   523 -524 670 -675
9224  9 -1.17e-3   524 -525 670 -675
9225  9 -1.17e-3   525 -526 670 -675
9226  9 -1.17e-3   526 -527 670 -675
c
9230  9 -1.17e-3   513 -521 675 -685
9231  9 -1.17e-3   521 -522 675 -685
9232  9 -1.17e-3   522 -523 675 -685
9233  9 -1.17e-3   523 -524 675 -685
9234  9 -1.17e-3   524 -525 675 -685
9235  9 -1.17e-3   525 -526 675 -685
9236  9 -1.17e-3   526 -527 675 -685
c
9240  9 -1.17e-3   513 -521 685 -723
9241  9 -1.17e-3   521 -522 685 -723
9242  9 -1.17e-3   522 -523 685 -723
9243  9 -1.17e-3   523 -524 685 -723
9244  9 -1.17e-3   524 -525 685 -723
9245  9 -1.17e-3   525 -526 685 -723
9246  9 -1.17e-3   526 -527 685 -723
c
```

```
c
c     impact limiters both top and bottom
c
9508  7 -1.61           -870 803 -801 u=25 $ holtite
9509  8 -7.82      870  -871  803 -801 u=25 $ steel support item 2
9510  8 -7.82      871   852 -851  803 -801 u=25
9511  8 -7.82      871   854 -853  803 -801 u=25
9512  8 -7.82      871   856 -855   803 -801 u=25
9513  8 -7.82      871   858 -857   803 -801 u=25
9514  8 -7.82      871   860 -859   803 -801 u=25
9515  8 -7.82      871   862 -861   803 -801 u=25
9516  8 -7.82      871   864 -863   803 -801 u=25
9517  8 -7.82      871   866 -865   803 -801 u=25
9518  7 -1.61      871   851 -854 2000  803 -801 u=25
9519  7 -1.61      871   853 -856 2000  803 -801 u=25
9520  7 -1.61      871   855 -858 2000  803 -801 u=25
9521  7 -1.61      871   857  859 2000  803 -801 u=25
9522  7 -1.61      871   861 -860 2000  803 -801 u=25
9523  7 -1.61      871   863 -862 2000  803 -801 u=25
9524  7 -1.61      871   865 -864 2000  803 -801 u=25
9525  7 -1.61      871   851 -866 2000  803 -801 u=25
9526  7 -1.61      871   853 -852 -2000  803 -801 u=25
9527  7 -1.61      871   855 -854 -2000  803 -801 u=25
9528  7 -1.61      871   857 -856 -2000  803 -801 u=25
9529  7 -1.61      871   -858 -860 -2000  803 -801 u=25
9530  7 -1.61      871   859 -862 -2000  803 -801 u=25
9531  7 -1.61      871   861 -864 -2000  803 -801 u=25
9532  7 -1.61      871   863 -866 -2000  803 -801 u=25
9533  7 -1.61      871   865 -852 -2000  803 -801 u=25
c
9534  8 -7.82                    801 u=25 $ steel plate
9535  9 -1.17e-3                -803  u=25 $ air below
c
9536  8 -7.82     -519  803 u=27
9537  9 -1.17e-3  519   803 u=27
9538  9 -1.17e-3        -803 u=27
c
9608  7 -1.61           -870 821 -823 u=26 $ holtite
9609  8 -7.82      870  -871  821 -823 u=26 $ steel support item 2
9610  8 -7.82      871   852 -851  821 -823 u=26
9611  8 -7.82      871   854 -853  821 -823 u=26
9612  8 -7.82      871   856 -855   821 -823 u=26
9613  8 -7.82      871   858 -857   821 -823 u=26
9614  8 -7.82      871   860 -859   821 -823 u=26
9615  8 -7.82      871   862 -861   821 -823 u=26
9616  8 -7.82      871   864 -863   821 -823 u=26
9617  8 -7.82      871   866 -865   821 -823 u=26
9618  7 -1.61      871   851 -854 2000  821 -823 u=26
9619  7 -1.61      871   853 -856 2000  821 -823 u=26
9620  7 -1.61      871   855 -858 2000  821 -823 u=26
9621  7 -1.61      871   857  859 2000  821 -823 u=26
9622  7 -1.61      871   861 -860 2000  821 -823 u=26
9623  7 -1.61      871   863 -862 2000  821 -823 u=26
9624  7 -1.61      871   865 -864 2000  821 -823 u=26
9625  7 -1.61      871   851 -866 2000  821 -823 u=26
9626  7 -1.61      871   853 -852 -2000  821 -823 u=26
9627  7 -1.61      871   855 -854 -2000  821 -823 u=26
9628  7 -1.61      871   857 -856 -2000  821 -823 u=26
9629  7 -1.61      871   -858 -860 -2000  821 -823 u=26
9630  7 -1.61      871   859 -862 -2000  821 -823 u=26
9631  7 -1.61      871   861 -864 -2000  821 -823 u=26
9632  7 -1.61      871   863 -866 -2000  821 -823 u=26
9633  7 -1.61      871   865 -852 -2000  821 -823 u=26
c
9634  8 -7.82                   -821 u=26 $ steel plate
9635  9 -1.17e-3                 823  u=26 $ air above
9637  8 -7.82     -519        -823 u=28
9638  9 -1.17e-3               823 u=28
9639  9 -1.17e-3   519        -823 u=28
```

```
c
9999  0  527:-701:723
c
c     BLANK LINE

c     BLANK LINE
c
c     MPC surfaces\/ \/ \/ \/ \/
c
1     cz          0.52832
2     cz          0.53213
3     cz          0.61341
4     cz          0.67437
5     cz          0.75057
6     px          0.8128
7     px         -0.8128
8     py          0.8128
9     py         -0.8128
10    px         -4.445
11    px          4.445
12    py         -4.445
13    py          4.445
c     14    px          -8.2423
14    px         -8.242301
15    px         -7.9248
16    px         -7.66826
17    px         -7.47776
18    px         -6.0325
19    px          6.0325
20    px          7.9248
c     21    px           8.2423
c     22    py          -8.2423
21    px          8.242301
22    py         -8.242301
23    py         -7.9248
24    py         -6.0325
25    py          6.0325
26    py          7.47776
27    py          7.66826
28    py          7.9248
c     29    py           8.2423
29    py          8.242301
c
30    px   -6.56082
31    px    6.56082
32    py   -6.56082
33    py    6.56082
c
101   px  -98.9076
102   px  -82.423
103   px  -65.9384
104   px  -49.4538
105   px  -32.9692
106   px  -16.4846
107   px   0.0
108   px   16.4846
109   px   32.9692
110   px   49.4538
111   px   65.9384
112   px   82.423
113   px   98.9076
c
201   py  -98.9076
202   py  -82.423
203   py  -65.9384
204   py  -49.4538
205   py  -32.9692
206   py  -16.4846
207   py   0.0
```

```
208    py   16.4846
209    py   32.9692
210    py   49.4538
211    py   65.9384
212    py   82.423
213    py   98.9076
c
301    cz   85.4075
302    cz   86.6775
c
c      620   pz    21.59     $ MPC baseplate        - 2.5 inches
c      400   pz   24.765  $ start of egg crate
400    pz   23.876  $ start of egg crate
410    pz   33.9725  $ start of boral
420    pz   40.3479 $ beginning of fuel
425    pz   406.1079 $ end of fuel
430    pz   421.3479 $ space
435    pz   430.2125  $ end of boral
440    pz   445.4271 $ plenum
445    pz   448.8561 $ expansion springs
450    pz   457.3397 $ top end fitting
455    pz   468.63   $ top of element
460    pz   466.344  $ top of egg crate
c
c      MPC surfaces/\ /\ /\ /\ /\
c
c
c      OVERPACK survaces \/ \/ \/ \/ \/
c
501    cz     87.3125  $ IR for overpack
502    cz     90.4875  $ item 2  1.25 inch
503    cz     93.6625  $ item 2  1.25 inch
504    cz     96.8375    $ item 12  1.25 inch
505    cz     100.0125    $ item 13  1.25 inch
506    cz     103.1875    $ item 14  1.25 inch
507    cz     106.3625    $ item 15  1.25 inch
508    cz     108.9025    $ item 16  1 inch
509    cz     111.521875  $ holtite
510    cz     114.14125   $ holtite
511    cz     116.760625  $ holtite
512    cz     119.53875      $ holtite - total 4.1875 inches
513    cz     120.80875      $ outer steel shell - 0.5 inches
c      512    cz    119.38      $ holtite - total 4.125 inches
c      513    cz    120.65      $ outer steel shell - 0.5 inches
514    cz     111.54       $ hole in pocket trunion
515    cz     105.7275     $ flange bottom of overpack
516    cz     105.7275     $ flange top    of overpack
517    cz     109.5375     $ shear ring
518    cz     103.1875     $ item 14  1.25 inch
519    cz     108.2675     $ impact limiter - 2 inch steel
c
521    cz     162.56       $ surface of impact limiters
522    cz     203.1875     $ 1 meter from 506 - upper and lower part overpack
523    cz     220.80875    $ 1 meter from 513 - outer steel
524    cz     303.1875     $ 2 meter from 506 - upper and lower part overpack
525    cz     320.80875    $ 2 meter from 513 - outer steel
526    cz     362.56       $ 2 meter from 521 - edge of impact limiters
527    cz     400.00
c
600    pz     0.0        $ bottom of overpack
610    pz     15.24      $ overpack baseplate   - 6 inches
620    pz     21.59      $ MPC baseplate        - 2.5 inches
630    pz     22.225     $ beginning of item 17 - 0.25 inches
635    pz     23.495     $ item 17              - 0.5 inches
640    pz     41.75125   $ hole in pocket trun  - 7.6875 inches from 630
645    pz     54.61      $ top of pocket trun   - 12.75 inches from 630
660    pz     455.6125   $ top of holtite       - 170.125 inches from 635
665    pz     460.6925   $ top of foam          - 2 inches
670    pz     461.9625   $ top of item 17 on top- 0.5 inches
```

```
675    pz     473.71      $ bottom of MPC in lid - 178 inches from 620
676    pz     476.5675    $ top of shear ring
677    pz     494.03      $ top of add steel
680    pz     499.11      $ top of MPC outer lid - 7.5 inches from 675
685    pz     500.6975    $ bot of overpack lid  - 5/8 inch
695    pz     515.9375    $ top of overpack lid  - 6 inches
c
c      tally segment surfaces
c
701    pz     -121.92
702    pz     -91.44
703    pz     -60.96
704    pz     -30.48
c      600    pz     0.0        $ bottom of overpack
c      630    pz     22.225     $ beginning of item 17 - 0.25 inches
705    pz     51.5408
706    pz     80.8567
707    pz     110.1725
708    pz     139.4883
709    pz     168.8042
710    pz     198.1200
711    pz     227.4358
712    pz     256.7517
713    pz     286.0675
714    pz     315.3833
715    pz     344.6992
716    pz     374.0150
717    pz     403.3308
718    pz     432.6467
c      670    pz     461.9625   $ top of item 17 on top- 0.5 inches
719    pz     488.3150
c      695    pz     514.6675   $ top of overpack lid  - 6 inches
720    pz     545.1475
721    pz     575.6275
722    pz     606.1075
723    pz     636.5875
c
801    pz     -2.54      $ steel disk
802    pz     -5.715     $ holtite
803    pz     -8.89      $ holtite
804    pz     -9.2075    $ cover over holtite
805    pz     -53.34     $ item 2 on impact limiters
810    pz     -100.0     $ 1 meter from surface overpack
811    pz     -105.7275 $ edge of impact limiter
812    pz     -200.0     $ 2 meter from surface overpack
813    pz     -305.7275 $ 2 meter from surface impact limiter
814    pz     -427.6475 $ 2 meter + 4 feet
815    pz     -488.6075 $ 2 meter + 6 feet
816    pz     -671.4875 $ 2 meter + 12 feet
817    pz     -700.00
c
821    pz     517.2075      $ steel disk
822    pz     520.3825      $ holtite
823    pz     523.5575      $ holtite
824    pz     523.875       $ cover over holtite
825    pz     568.0075      $ item 2 on impact limiters
830    pz     614.6675      $ 1 meter from surface overpack
831    pz     620.3950      $ edge of impact limiter
832    pz     714.6675      $ 2 meter from surface overpack
833    pz     820.395       $ 2 meter from surface impact limiter
834    pz     942.3150      $ 2 meter + 4 feet
835    pz     1003.275      $ 2 meter + 6 feet
836    pz     1186.155      $ 2 meter + 12 feet
837    pz     1200.00
c
851        py     1.5875
852        py    -1.5875
853    11  py     1.5875
854    11  py    -1.5875
```

```
855   12  py    1.5875
856   12  py   -1.5875
857   13  py    1.5875
858   13  py   -1.5875
859       px    1.5875
860       px   -1.5875
861   11  px    1.5875
862   11  px   -1.5875
863   12  px    1.5875
864   12  px   -1.5875
865   13  px    1.5875
866   13  px   -1.5875
c
870       cz    38.1
871       cz    41.91
c          steel spine and holtite cells
c
1000       px    0.0
1001       px   -0.635
1002       px    0.635
1011   1   px   -0.635
1012   1   px    0.635
1021   2   px   -0.635
1022   2   px    0.635
1031   3   px   -0.635
1032   3   px    0.635
1041   4   px   -0.635
1042   4   px    0.635
1051   5   px   -0.635
1052   5   px    0.635
1061   6   px   -0.635
1062   6   px    0.635
1071   7   px   -0.635
1072   7   px    0.635
1081   8   px   -0.635
1082   8   px    0.635
1091   9   px   -0.635
1092   9   px    0.635
c
1101    px    15.71625   $ pocket trunion
1102    px    8.09625    $ pocket trunion opening
1103    px   -8.09625    $ pocket trunion opening 6 3/8 inches thick
1104    px   -15.71625   $ pocket trunion - 9 3/8 inches thick
c
2000       py    0.0
2001       py   -0.635
2002       py    0.635
2011   1   py   -0.635
2012   1   py    0.635
2021   2   py   -0.635
2022   2   py    0.635
2031   3   py   -0.635
2032   3   py    0.635
2041   4   py   -0.635
2042   4   py    0.635
2051   5   py   -0.635
2052   5   py    0.635
2061   6   py   -0.635
2062   6   py    0.635
2071   7   py   -0.635
2072   7   py    0.635
2081   8   py   -0.635
2082   8   py    0.635
2091   9   py   -0.635
2092   9   py    0.635
c
2101    py    15.71625   $ pocket trunion
2102    py    8.09625    $ pocket trunion opening
2103    py   -8.09625    $ pocket trunion opening 6 3/8 inches thick
```

```
2104    py  -15.71625  $ pocket trunion - 9 3/8 inches thick
c
c      OVERPACK surfaces /\ /\ /\ /\ /\
c
c      BLANK LINE

c      BLANK LINE
c
*tr1     0 0 0  9 279 90  99  9 90 90 90 0
*tr2     0 0 0 18 288 90 108 18 90 90 90 0
*tr3     0 0 0 27 297 90 117 27 90 90 90 0
*tr4     0 0 0 36 306 90 126 36 90 90 90 0
*tr5     0 0 0 45 315 90 135 45 90 90 90 0
*tr6     0 0 0 54 324 90 144 54 90 90 90 0
*tr7     0 0 0 63 333 90 153 63 90 90 90 0
*tr8     0 0 0 72 342 90 162 72 90 90 90 0
*tr9     0 0 0 81 351 90 171 81 90 90 90 0
*tr11    0 0 0 22.5 292.5 90 112.5 22.5 90 90 90 0
*tr12    0 0 0 45.0 315.0 90 135.0 45.0 90 90 90 0
*tr13    0 0 0 67.5 337.5 90 157.5 67.5 90 90 90 0
c
c          PHOTON MATERIALS
c
c      fuel 3.4 w/o U235  10.412 gm/cc
c      m1              92235.01p   -0.029971
c                      92238.01p   -0.851529
c                      8016.01p    -0.1185
c      c     homogenized fuel  density 4.29251 gm/cc
c      m2              92235.01p   -0.024966
c                      92238.01p   -0.709315
c                       8016.01p   -0.098709
c                      40000.01p   -0.16701
c      c     zirconium 6.55 gm/cc
c      m3              40000.01p    1.          $ Zr Clad
c      c     stainless steel 7.92 gm/cc
c      m5              24000.01p   -0.19
c                      25055.01p   -0.02
c                      26000.01p   -0.695
c                      28000.01p   -0.095
c      c     boral  2.644 gm/cc
c      m6              5010.01p    -0.044226
c                      5011.01p    -0.201474
c                      13027.01p   -0.6861
c                      6000.01p    -0.0682
c      c     holtite 1.61 gm/cc
c      m7              6000.01p  -0.2766039
c                     13027.01p  -0.21285
c                      1001.01p  -0.0592
c                      8016.01p  -0.42372
c                      7014.01p  -0.0198
c                      5010.01p  -0.0014087
c                      5011.01p  -0.0064174
c      c     carbon steel 7.82 gm/cc
c      m8        6000.01p -0.005 26000.01p -0.995
c      c     air density 1.17e-3 gm/cc
c      m9        7014.01p 0.78 8016.01p 0.22
c      c
c      c         NEUTRON MATERIALS
c
c      fuel 3.4 w/o U235  10.412 gm/cc
m1           92235.50c   -0.029971
             92238.50c   -0.851529
             8016.50c    -0.1185
c      homogenized fuel  density 4.29251 gm/cc
m2           92235.50c   -0.024966
             92238.50c   -0.709315
              8016.50c   -0.098709
             40000.35c   -0.16701
c      zirconium 6.55 gm/cc
```

```
m3            40000.35c   1.          $ Zr Clad
c     stainless steel 7.92 gm/cc
m5            24000.50c   -0.19
              25055.50c   -0.02
              26000.55c   -0.695
              28000.50c   -0.095
c     boral  2.644 gm/cc
m6            5010.50c    -0.044226
              5011.56c    -0.201474
              13027.50c   -0.6861
              6000.50c    -0.0682
c     holtite 1.61 gm/cc
m7            6000.50c  -0.2766039
             13027.50c  -0.21285
              1001.50c  -0.0592
              8016.50c  -0.42372
              7014.50c  -0.0198
              5010.50c  -0.0014087
              5011.56c  -0.0064174
mt7    lwtr.01t
c     carbon steel 7.82 gm/cc
m8      6000.50c -0.005 26000.55c -0.995
c     air density 1.17e-3 gm/cc
m9      7014.50c 0.78 8016.50c 0.22
c
phys:n   20 0.0
phys:p   100 0
c     imp:n   1 228r 0
c     imp:p   1 228r 0
nps     500000
prdmp    j   -30   1   2
c     print   10 110 160 161 20 170
print
mode  n p
c
sdef  par=1  erg=d1 axs=0 0 1 x=d4 y=fx d5 z=d3
c
c    energy dist for gammas in the fuel
c
c    si1  h  0.7 1.0 1.5 2.0 2.5 3.0
c    sp1     0  0.31 0.31 0.15 0.15 0.08
c
c    energy dist for neutrons in the fuel
c
si1  h  0.1 0.4 0.9 1.4 1.85 3.0 6.43 20.0
sp1     0   0.03787 0.1935 0.1773 0.1310 0.2320 0.2098 0.01853
c
c    energy dist for Co60 gammas
c
c    si1  d 1.3325 1.1732
c    sp1    0.5    0.5
c
c    axial dist for neut and phot in fuel
c
si3  h 40.3479 55.5879 70.8279 101.3079 162.2679 223.2279
       284.1879 345.1479 375.6279 390.8679 406.1079
sp3    0 0.00005 0.00961 0.07031 0.23323 0.25719 0.22907
         0.16330 0.03309 0.00409 0.00005
sb3    0 0.009167 0.031667 0.086250 0.194583 0.199167
       0.193750 0.178750 0.072083 0.025833 0.009167
c
c    axial dist for Co60 - a zero prob is in the fuel
c
c    si3  h 21.59 40.3479 421.3479 445.4271 448.8561 457.3397 468.63
c    sp3    0      0.547   0.0      0.125     0.045     0.227     0.056
c
si4  s               15 16
               13 14 15 16 17 18
            12 13 14 15 16 17 18 19
```

```
              12 13 14 15 16 17 18 19
           11 12 13 14 15 16 17 18 19 20
           11 12 13 14 15 16 17 18 19 20
              12 13 14 15 16 17 18 19
              12 13 14 15 16 17 18 19
                 13 14 15 16 17 18
                       15 16
sp4   1 67r
c
ds5   s                30 30
                 29 29 29 29 29 29
              28 28 28 28 28 28 28 28
              27 27 27 27 27 27 27 27
           26 26 26 26 26 26 26 26 26 26
           25 25 25 25 25 25 25 25 25 25
              24 24 24 24 24 24 24 24
              23 23 23 23 23 23 23 23
                 22 22 22 22 22 22
                       21 21
c
si11   -80.74152 -67.61988
si12   -64.25692 -51.13528
si13   -47.77232 -34.65068
si14   -31.28772 -18.16608
si15   -14.80312 -1.68148
si16    1.68148   14.80312
si17    18.16608  31.28772
si18    34.65068  47.77232
si19    51.13528  64.25692
si20    67.61988  80.74152
c
si21   -80.74152 -67.61988
si22   -64.25692 -51.13528
si23   -47.77232 -34.65068
si24   -31.28772 -18.16608
si25   -14.80312 -1.68148
si26    1.68148   14.80312
si27    18.16608  31.28772
si28    34.65068  47.77232
si29    51.13528  64.25692
si30    67.61988  80.74152
sp11   0 1
sp12   0 1
sp13   0 1
sp14   0 1
sp15   0 1
sp16   0 1
sp17   0 1
sp18   0 1
sp19   0 1
sp20   0 1
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sp28   0 1
sp29   0 1
sp30   0 1
c
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c    2.5e-8   1.0e-7   1.0e-6   1.0e-5   1.0e-4   1.0e-3   1.0e-2    0.1
c        0.5      1.0      2.5      5.0      7.0      10.0      14.0     20.0
c    3.67e-6  3.67e-6  4.46e-6  4.54e-6  4.18e-6  3.76e-6  3.56e-6 2.17e-5
c        9.26e-5  1.32e-4  1.25e-4  1.56e-4  1.47e-4  1.47e-4  2.08e-4 2.27e-4
c
c       photon dose factors
c
c    0.01  0.03  0.05  0.07  0.1  0.15  0.2  0.25  0.3  0.35  0.4  0.45
c         0.5  0.55  0.6  0.65  0.7  0.8  1.0  1.4  1.8  2.2  2.6  2.8  3.25
c         3.75  4.25  4.75  5.0  5.25  5.75  6.25  6.75  7.5  9.0  11.0
c         13.0  15.0
c   3.96e-06  5.82e-07  2.90e-07  2.58e-07  2.83e-07  3.79e-07  5.01e-07
c        6.31e-07  7.59e-07  8.78e-07  9.85e-07  1.08e-06  1.17e-06  1.27e-06
c        1.36e-06  1.44e-06  1.52e-06  1.68e-06  1.98e-06  2.51e-06  2.99e-06
c        3.42e-06  3.82e-06  4.01e-06  4.41e-06  4.83e-06  5.23e-06  5.60e-06
c        5.80e-06  6.01e-06  6.37e-06  6.74e-06  7.11e-06  7.66e-06  8.77e-06
c        1.03e-05  1.18e-05  1.33e-05
c
c
f2:n    515 517 516 518
fc2     bot  shear top very-top
ft2     scx 1
de2    2.5e-8   1.0e-7   1.0e-6   1.0e-5   1.0e-4   1.0e-3   1.0e-2    0.1
       0.5      1.0      2.5      5.0      7.0      10.0      14.0     20.0
df2    3.67e-6  3.67e-6  4.46e-6  4.54e-6  4.18e-6  3.76e-6  3.56e-6 2.17e-5
       9.26e-5  1.32e-4  1.25e-4  1.56e-4  1.47e-4  1.47e-4  2.08e-4 2.27e-4
fq2    u s
c
f12:n    513 521 522 523 524 525 526
fs12     -702 -703 -704 -600 -630 -705 -706 -707 -708 -709 -710
         -711 -712 -713 -714 -715 -716 -717 -718 -670 -719 -695
         -720 -721 -722 t
fc12     1ft      1ft      1ft      1ft      8.75in    11.54in   11.54in 11.54in
         11.54in 11.54in 11.54in 11.54in 11.54in   11.54in   11.54in 11.54in
         11.54in 11.54in 11.54in 11.54in 10.375in 10.375in 1ft      1ft
         1ft      1ft
ft12     scx 1
de12     2.5e-8   1.0e-7   1.0e-6   1.0e-5   1.0e-4   1.0e-3   1.0e-2    0.1
       0.5      1.0      2.5      5.0      7.0      10.0      14.0     20.0
df12     3.67e-6  3.67e-6  4.46e-6  4.54e-6  4.18e-6  3.76e-6  3.56e-6 2.17e-5
       9.26e-5  1.32e-4  1.25e-4  1.56e-4  1.47e-4  1.47e-4  2.08e-4 2.27e-4
fq12     u s
c
c       PHOTON TALLIES
c
f102:p     515 517 516 518
fc102      bot  shear top very-top
ft102      scx 1
de102      0.01  0.03  0.05  0.07  0.1  0.15  0.2  0.25  0.3  0.35  0.4  0.45
           0.5  0.55  0.6  0.65  0.7  0.8  1.0  1.4  1.8  2.2  2.6  2.8  3.25
           3.75  4.25  4.75  5.0  5.25  5.75  6.25  6.75  7.5  9.0  11.0
           13.0  15.0
df102     3.96e-06  5.82e-07  2.90e-07  2.58e-07  2.83e-07  3.79e-07  5.01e-07
          6.31e-07  7.59e-07  8.78e-07  9.85e-07  1.08e-06  1.17e-06  1.27e-06
          1.36e-06  1.44e-06  1.52e-06  1.68e-06  1.98e-06  2.51e-06  2.99e-06
          3.42e-06  3.82e-06  4.01e-06  4.41e-06  4.83e-06  5.23e-06  5.60e-06
          5.80e-06  6.01e-06  6.37e-06  6.74e-06  7.11e-06  7.66e-06  8.77e-06
          1.03e-05  1.18e-05  1.33e-05
fq102     u s
c
f112:p     513 521 522 523 524 525 526
```

```
fs112    -702 -703 -704 -600 -630 -705 -706 -707 -708 -709 -710
        -711 -712 -713 -714 -715 -716 -717 -718 -670 -719 -695
        -720 -721 -722 t
fc112    1ft     1ft     1ft     1ft     8.75in   11.54in  11.54in 11.54in
        11.54in 11.54in 11.54in 11.54in 11.54in  11.54in  11.54in 11.54in
        11.54in 11.54in 11.54in 11.54in 10.375in 10.375in 1ft     1ft
        1ft     1ft
ft112    scx 1
de112    0.01  0.03  0.05  0.07  0.1  0.15  0.2  0.25  0.3  0.35  0.4  0.45
         0.5  0.55  0.6  0.65  0.7  0.8  1.0  1.4  1.8  2.2  2.6  2.8  3.25
         3.75  4.25  4.75  5.0  5.25  5.75  6.25  6.75  7.5  9.0  11.0
         13.0  15.0
df112   3.96e-06  5.82e-07  2.90e-07  2.58e-07  2.83e-07  3.79e-07  5.01e-07
        6.31e-07  7.59e-07  8.78e-07  9.85e-07  1.08e-06  1.17e-06  1.27e-06
        1.36e-06  1.44e-06  1.52e-06  1.68e-06  1.98e-06  2.51e-06  2.99e-06
        3.42e-06  3.82e-06  4.01e-06  4.41e-06  4.83e-06  5.23e-06  5.60e-06
        5.80e-06  6.01e-06  6.37e-06  6.74e-06  7.11e-06  7.66e-06  8.77e-06
        1.03e-05  1.18e-05  1.33e-05
fq112   u s
c
```