TABLE 1.2.5 CRITICALITY AND SHIELDING SIGNIFICANT SYSTEM DATA		
Item	**Property**	**Value**
Metamic-HT Neutron Absorber	Nominal Thickness (mm)	10 (MPC-89) 15 (MPC-37) 15 (MPC-32ML)
	Minimum B_4C Weight %	10 (MPC-89) 10 (MPC-37) 10 (MPC-32ML)
Concrete in HI-STORM FW overpack body and lid	Installed Nominal Density (lb/ft^3)	150 (reference) 200 (maximum)
Holtite-A	Specific gravity (g/cm^3)	1.68
	Density (g/cm^3)	1.61
	Hydrogen Content (weight percent)	5.92
	B_4C content (weight percent)	1.0

Figure 1.2.7c: Figure Deleted

or post-weld cracking. The heat input and cooling rates in welding are important as they control ferrite to austenite transformation. Exceedingly low heat input may result in fusion zones and HAZ which are excessively ferritic (above 70%) [1.A.6]. Exceedingly high heat input increases the danger of forming intermetallic phases [1.A.6]. In both cases the impact toughness and corrosion resistance of the DSS will be seriously affected. Hence, heat input must be $0.6 - 2.6$ kJ/mm to retain the phase balance, limit the width of the HAZ, and obtain a sigma phase free product [1.A.5]. Further, cooling rate from the solution annealing temperature must exceed 0.3°C/s to avoid sigma phase and satisfy the generally accepted toughness requirements [1.A.8]. The maximum interpass temperature is limited to 150°C (302°F) [1.A.6].

DSS have chloride stress corrosion cracking (CSCC) resistance significantly greater than that of the austenitic stainless steels, but they are not completely immune. Experimental results indicate that DSS is prone to stress corrosion cracking at temperatures above 100°C [1.A.9]. Poor welding practice, a low pH, presence of Hydrogen in welds, and/or high ferrite (>70%) can contribute to failures at temperatures below 100°C.

Holtec will make sure that this material shall be used *only* if the metal temperature of the MPC shell can be assured to remain below the limit in Table 1.A.6 under all *normal operating* modes [1.A.3]. Likewise, under short term and accident conditions, such as the "inlet duct blockage" scenario, the maximum metal temperature of duplex stainless steel must be held below the limit in Table 1.A.6.

To confirm that the required properties are achieved in production, Holtec will implement a test program to ensure that the weldments are tested for the absence of detrimental intermetallic phases. The test program will comply with ASTM A923 and will use metallographic examination, impact testing and corrosion testing to demonstrate the absence of such detrimental phases. The test will be intended to determine the presence or absence of intermetallic phase to the extent that it is detrimental to the toughness and corrosion resistance of the material. The test *shall* be implemented to products during weld procedure qualification as well as during fabrication which will provide the assurance that the weldments are *free* from detrimental intermetallic phases, and *provide* the required corrosion resistance and fracture toughness [1.A.7].

For other stainless steels listed as members of Alloy X above, the design temperature limits in Table 2.2.3 remain unmodified.

This appendix defines the least favorable material properties of Alloy X.

1.A.2   Common Material Properties

Several material properties do not vary significantly from one Alloy X constituent to the next. These common material properties are as follows:

- density

2.1.4   Structural Parameters for Design Basis SNF

The main physical parameters of an SNF assembly applicable to the structural evaluation are the fuel assembly length, cross sectional dimensions, and weight. These parameters, which define the mechanical and structural design, are specified in Subsection 2.1.8. An appropriate axial clearance is provided to prevent interference due to the irradiation and thermal growth of the fuel assemblies.

2.1.5   Thermal Parameters for Design Basis SNF

The principal thermal design parameter for the stored fuel is the fuel's peak cladding temperature (PCT) which is a function of the maximum decay heat per assembly and the decay heat removal capabilities of the HI-STORM FW System.

To ensure the permissible PCT limits are not exceeded, Subsection 1.2 specifies the maximum allowable decay heat per assembly for each MPC model in the three-region configuration (see also Table 1.2.3 and 1.2.4).

The fuel cladding temperature is also affected by the heat transfer characteristics of the fuel assemblies. The design basis fuel assembly for thermal calculations for both PWR and BWR fuel is provided in Table 2.1.4.

Finally, the axial variation in the heat generation rate in the design basis fuel assembly is defined based on the axial burnup distribution. For this purpose, the data provided in references [2.1.3] and [2.1.4] are utilized and summarized in Table 2.1.5 and Figures 2.1.3 and 2.1.4. These distributions are representative of fuel assemblies with the design basis burnup levels considered. These distributions are used for analyses only, and do not provide a criteria for fuel assembly acceptability for storage in the HI-STORM FW System.

2.1.6   Radiological Parameters for Design Basis SNF

The principal radiological design criteria for the HI-STORM FW System are the 10CFR72 §104 and §106 operator-controlled boundary dose rate limits, and the requirement to maintain operational dose rates as low as reasonably achievable (ALARA). The radiation dose is directly affected by the gamma and neutron source terms of the assembly, which is a function of the assembly type, and the burnup, enrichment and cooling time of the assemblies. Dose rates are further directly affected by the size and arrangement of the ISFSI, and the specifics of the loading operations. All these parameters are site-dependent, and the compliance with the regulatory dose rate requirements are performed in site-specific calculations. The evaluations here are therefore performed with reference fuel assemblies, and with parameters that result in reasonably conservative dose rates. The reference assemblies given in Table 1.0.4 are the predominant assemblies used in the industry.

The design basis dose rates can be met by a variety of burnup levels and cooling times. Table 2.1.1 provides the acceptable ranges of burnup, enrichment and cooling time for all of the authorized fuel assembly array/classes. Table 2.1.5 and Figures 2.1.3 and 2.1.4 provide the axial distribution for the radiological source terms for PWR and BWR fuel assemblies based on the axial burnup distribution. The axial burnup distributions are representative of fuel assemblies with the design basis burnup levels considered. These distributions are used for analyses only, and do not provide a criterion for fuel assembly acceptability for storage in the HI-STORM FW System.

Non-fuel hardware, as defined in the Glossary, has been evaluated and is also authorized for storage in the PWR MPCs as specified in Table 2.1.1.

2.1.6.1   Radiological Parameters for Spent Fuel and Non-fuel Hardware in MPC-32ML, MPC-37 and MPC-89

MPC-32ML is authorized to store 16x16D spent fuel with burnup - cooling time combinations as given in Table 2.1.9. Spent fuel with burnup – cooling time combinations authorized for storage in MPC-37 and MPC-89 are given in Table 2.1.10.

The burnup and cooling time for every fuel assembly loaded into the MPC-32ML, MPC-37 and MPC-89 must satisfy the following equation:

$$Ct = A \cdot Bu^3 + B \cdot Bu^2 + C \cdot Bu + D$$

where,

Ct = Minimum cooling time (years),
Bu = Assembly-average burnup (MWd/mtU),
A, B, C, D = Polynomial coefficients listed in Table 2.1.9 or Table 2.1.10

2.1.7 Criticality Parameters for Design Basis SNF

Criticality control during loading of the MPC-37 is achieved through either meeting the soluble boron limits in Table 2.1.6 OR verifying that the assemblies meet the minimum burnup requirements in Table 2.1.7. Criticality control during loading of the MPC-32ML is achieved through meeting the soluble boron limits in Table 2.1.6.

For those spent fuel assemblies that need to meet the burnup requirements specified in Table 2.1.7, a burnup verification shall be performed in accordance with either Method A OR Method B described below.

Method A: Burnup Verification Through Quantitative Burnup Measurement

For each assembly in the MPC-37 where burnup credit is required, the minimum burnup is determined from the burnup requirement applicable to the loading configuration chosen for the cask (see Table 2.1.7). A measurement is then performed that confirms that the fuel assembly burnup exceeds this minimum burnup. The measurement technique may be calibrated to the reactor records for a representative set of assemblies. The assembly burnup value to be compared with the minimum required burnup should be the measured burnup value as adjusted by reducing the value by a combination of the uncertainties in the calibration method and the measurement itself.

Method B: Burnup Verification Through an Administrative Procedure and Qualitative Measurements

Depending on the location in the basket, assemblies loaded into a specific MPC-37 can either be fresh, or have to meet a single minimum burnup value. The assembly burnup value to be compared with the minimum required burnup should be the reactor record burnup value as adjusted by reducing the value by the uncertainties in the reactor record value. An administrative procedure shall be established that prescribes the following steps, which shall be performed for each cask loading:

- Based on a review of the reactor records, all assemblies in the spent fuel pool that have a burnup that is below the minimum required burnup of the loading curve for the cask to be loaded are identified.
- After the cask loading, but before the release for shipment of the cask, the presence and location of all those identified assemblies is verified, except for those assemblies that have been loaded as fresh assemblies into the cask.

Additionally, for all assemblies to be loaded that are required to meet a minimum burnup, a measurement shall be performed that verifies that the assembly is not a fresh assembly.

TABLE 2.1.9
BURNUP AND COOLING TIME FUEL QUALIFICATION REQUIREMENTS
FOR MPC-32ML (NOTE 1)

A	B	C	D
6.7667E-14	-3.6726E-09	8.1319E-05	2.7951E+00

Notes:
1. The burnup and cooling time for every fuel loaded into the MPC-32ML must satisfy the following equation:

$$Ct = A \cdot Bu^3 + B \cdot Bu^2 + C \cdot Bu + D$$

where,

Ct = Minimum cooling time (years)

Bu = Assembly-average burnup (MWd/mtU),

A, B, C, D = Polynomial coefficients listed above

Reference Decay Heat[1][2] (kW)	A	B	C	D
MPC-37				
0.85	1.68353E-13	-9.65193E-09	2.69692E-04	2.95915E-01
3.5	1.19409E-14	-1.53990E-09	9.56825E-05	-3.98326E-01
MPC-89				
0.32	1.65723E-13	-9.28339E-09	2.57533E-04	3.25897E-01
0.5	3.97779E-14	-2.80193E-09	1.36784E-04	3.04895E-01
0.75	1.44353E-14	-1.21525E-09	8.14851E-05	3.31914E-01
1.1	-7.45921E-15	1.09091E-09	-1.14219E-05	9.76224E-01
1.45	3.10800E-15	-7.92541E-11	1.56566E-05	6.47040E-01
1.6	-8.08081E-15	1.23810E-09	-3.48196E-05	1.11818E+00

Notes:

1. The maximum allowable decay heat load per fuel basket cell, i.e. a decay heat value that is equal to or greater than the appropriate uniform and regionalized decay heat load limits, is specified.

2. The burnup and cooling time for every fuel loaded into the MPC-37 or MPC-89 must satisfy the following equation:

$$Ct = A \cdot Bu^3 + B \cdot Bu^2 + C \cdot Bu + D$$

where,

Ct = Minimum cooling time (years)

---

[1] Decay heat per fuel assembly is presented.
[2] A decay heat value that is equal to or greater than the appropriate decay heat load limit.

Bu = Assembly-average burnup (MWd/mtU),

A, B, C, D = Polynomial coefficients listed above

TABLE 2.1.11
DAMAGED FUEL ISOLATOR CRITICAL CHARACTERISTICS

[Proprietary Information Withheld in Accordance with 10 CFR 2.390]

5.1   DISCUSSION AND RESULTS

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

- Gamma radiation originating from the following sources:

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

- Neutron radiation originating from the following sources

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

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

The shielding analyses were performed with MCNP5 [5.1.1] developed by Los Alamos National Laboratory (LANL). The source terms for the design basis fuels were calculated with the TRITON and ORIGAMI sequences from the SCALE 6.2.1 system [5.1.4]. Additional calculations in Section 5.4 were performed with the SAS2H and ORIGEN-S sequences of the SCALE 5 system [5.1.2, 5.1.3]. A detailed description of the MCNP models and the source term calculations are presented in Sections 5.3 and 5.2, respectively.

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

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

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

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

It is recognized that for a given heat load, an infinite number of burnup and cooling time combination could be selected, which would result in slightly different dose rate distributions around the cask. For a high burnup with a corresponding longer cooling time, dose locations with a high neutron contribution would show higher dose values, due to the non-linear relationship between burnup and neutron source term. At other locations dose rates are more dominated by contribution from the gamma sources. In these cases, short cooling time and lower burnup combinations with heat load comparable to the higher burnup and corresponding longer cooling time combinations would result in higher dose rates. However, in those cases, there would always be a compensatory effect, since for each dose location, higher neutron dose rates would be partly offset by lower gamma dose rates and vice versa. This is further complicated by the regionalized loading patterns qualified from a thermal perspective. These contain cells with substantially different heat load limits, and hence substantially different ranges of burnup, enrichment and cooling time combinations. The approach to cover all those variations in a conservative way is outlined below.

The uniform and regionalized loading curves for the fuel to be loaded in the MPC-37 or MPC-89 canisters are discussed in Subsection 5.2.7. For each loading curve, a number of burnup, enrichment and cooling time combinations are conservatively selected and listed in Tables 5.0.3, 5.0.4a, and 5.0.4b. Dose calculations are then performed for all combinations of the burnup, enrichment and cooling times for the different regions. The basket loading patterns are evaluated and the combination of region-specific dose rates that comes up with the maximum total dose rate are determined for each dose location together with the set of limiting burnup, enrichment and cooling times. Hence for each dose location, the set of the burnup, enrichment and cooling time combinations that form the basket loading pattern, which produces the bounding dose rates, may be different.

Based on this approach, the source terms used in the analyses of MPC-37 or MPC-89 are reasonably bounding for all realistically expected assemblies. All dose rates in this chapter are developed using this approach, unless noted otherwise. Also, as discussed in Section 5.2, the design basis BPRA activity is considered for MPC-37 in this chapter, unless noted otherwise.

All dose rates in Section 5.1 are developed using this regionalized approach. Some dose rates in Section 5.4 were retained from previous versions of the FSAR and that are based on a representative (while still conservative) uniform loading pattern, as discussed in that Section.

Table 5.0.3

SELECTED BURNUP, ENRICHMENT, COOLING TIME COMBINATIONS FOR THE MPC-37 LOADING PATTERNS BASED ON FIGURES 1.2.3 THROUGH 1.2.5 AND TABLE 2.1.10

Region	Burnup (MWd/mtU)	Enrichment (wt% ^{235}U)	Cooling Time (years)	Reference Decay Heat[1] (kW)
High Heat Load Basket Regions	5000	1.1	1.0	3.5
	10000	1.1	1.0	
	20000	1.6	1.0	
	30000	2.4	1.4	
	40000	3.0	1.6	
	50000	3.6	2.0	
	60000	3.9	2.2	
	70000	4.5	2.8	
Low Heat Load Basket Regions	5000	1.1	1.4	0.85
	10000	1.1	2.0	
	20000	1.6	3.0	
	30000	2.4	4.0	
	40000	3.0	6.0	
	50000	3.6	10.0	
	60000	3.9	18.0	
	70000	4.5	29.0	

[1] A decay heat value that is equal to or greater than the appropriate decay heat load limit.

Table 5.0.4a

SELECTED BURNUP, ENRICHMENT, COOLING TIME COMBINATIONS FOR THE MPC-89 LOADING PATTERNS BASED ON FIGURE 1.2.6 AND TABLE 2.1.10

Region	Burnup (MWd/mtU)	Enrichment (wt% ^{235}U)	Cooling Time (years)	Reference Decay Heat[1] (kW)
High Heat Load Basket Regions	5000	0.7	1.0	1.45
	10000	0.9	1.0	
	20000	1.6	1.0	
	30000	2.4	1.0	
	40000	3.0	1.2	
	50000	3.3	1.6	
	60000	3.7	1.8	
	70000	4.0	2.4	
Low Heat Load Basket Regions	5000	0.7	1.4	0.32
	10000	0.9	2.0	
	20000	1.6	3.0	
	30000	2.4	4.0	
	40000	3.0	6.0	
	50000	3.3	10.0	
	60000	3.7	18.0	
	70000	4.0	29.0	

---

[1] A decay heat value that is equal to or greater than the appropriate decay heat load limit.

Table 5.0.4b

SELECTED BURNUP, ENRICHMENT, COOLING TIME COMBINATIONS FOR THE MPC-89 LOADING PATTERNS BASED ON FIGURE 1.2.7 AND TABLE 2.1.10

Region	Burnup (MWd/mtU)	Enrichment (wt% ^{235}U)	Cooling Time (years)	Reference Decay Heat[1] (kW)
High Heat Load Basket Regions (Region 1)	5000	0.7	1.0	1.6
	10000	0.9	1.0	
	20000	1.6	1.0	
	30000	2.4	1.0	
	40000	3.0	1.0	
	50000	3.3	1.4	
	60000	3.7	1.6	
	70000	4.0	1.8	
High Heat Load Basket Regions (Region 2)	5000	0.7	1.0	1.1
	10000	0.9	1.0	
	20000	1.6	1.0	
	30000	2.4	1.4	
	40000	3.0	1.6	
	50000	3.3	2.2	
	60000	3.7	2.6	
	70000	4.0	2.8	
Low Heat Load Basket Regions (Region 3)	5000	0.7	1.0	0.75
	10000	0.9	1.0	
	20000	1.6	1.4	
	30000	2.4	2.0	
	40000	3.0	2.4	
	50000	3.3	3.0	
	60000	3.7	3.5	
	70000	4.0	5.0	

[1] A decay heat value that is equal to or greater than the appropriate decay heat load limit.

Table 5.0.4b (continued)

SELECTED BURNUP, ENRICHMENT, COOLING TIME COMBINATIONS FOR THE MPC-89 LOADING PATTERNS BASED ON FIGURE 1.2.7 AND TABLE 2.1.10

Region	Burnup (MWd/mtU)	Enrichment (wt% ^{235}U)	Cooling Time (years)	Reference Decay Heat[1] (kW)
Low Heat Load Basket Regions (Region 4)	5000	0.7	1.0	0.5
	10000	0.9	1.4	
	20000	1.6	2.2	
	30000	2.4	2.8	
	40000	3.0	3.5	
	50000	3.3	5.0	
	60000	3.7	7.0	
	70000	4.0	9.0	
Low Heat Load Basket Regions (Region 5)	5000	0.7	1.4	0.32
	10000	0.9	2.0	
	20000	1.6	3.0	
	30000	2.4	4.0	
	40000	3.0	6.0	
	50000	3.3	10.0	
	60000	3.7	18.0	
	70000	4.0	29.0	

[1] A decay heat value that is equal to or greater than the appropriate decay heat load limit.

equivalent to any individual organ or tissue (other than the lens of the eye) of 50 Rem. The lens dose equivalent shall not exceed 15 Rem and the shallow dose equivalent to skin or to any extremity shall not exceed 50 Rem. The minimum distance from the spent fuel or high-level radioactive waste handling and storage facilities to the nearest boundary of the controlled area shall be at least 100 meters.

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

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

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

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

The complete loss of the HI-TRAC neutron shield significantly affects the dose at mid-height (Dose Point #2) adjacent to the HI-TRAC. Loss of the neutron shield has a small effect on the dose at the other dose points. To illustrate the impact of the design basis accident, the dose rates at Dose Point #2 (see Figure 5.1.2) are provided in Table 5.1.4a with MPC-37 and Table 5.1.4b with MPC-89 for the HI-TRAC VW at a distance of 1 meter and at a distance of 100 meters. The normal condition dose rates are provided for reference. The dose for a period of 30 days is shown in Table 5.1.9, where 30 days is used to illustrate the radiological impact for a design basis accident. Based on this dose rate and the short duration of use for the loaded HI-TRAC transfer cask, it is evident that the dose as a result of the design basis accident cannot exceed 5 rem at the controlled area boundary for the short duration of the accident.

The HI-TRAC VW Version V2 shielding accident case where potentially the Holtite-A is lost from fire is bounded by the standard HI-TRAC VW for accident cases since the total radial

Table 5.1.1

MAXIMUM DOSE RATES FROM THE HI-TRAC VW FOR NORMAL CONDITIONS
MPC-37 DESIGN BASIS FUEL
REGIONALIZED LOADING BASED ON FIGURES 1.2.3 THROUGH 1.2.5

Dose Point Location	Fuel Gammas (mrem/hr)	(n,γ) Gammas (mrem/hr)	^{60}Co Gammas (mrem/hr)	Neutrons (mrem/hr)	Totals (mrem/hr)	Totals with BPRAs (mrem/hr)
ADJACENT TO THE HI-TRAC VW						
1	1137.9	17.2	755.5	46.2	1956.9	2115.8
2	3577.0	3.4	0.1	6.7	3587.3	4072.5
3	41.2	4.8	351.2	5.4	402.7	596.1
4	134.0	1.4	481.3	249.2	865.9	1177.7
5	710.0	2.9	1781.0	1122.0	3615.9	3734.7
ONE METER FROM THE HI-TRAC VW						
1	851.3	0.6	71.8	1.5	925.2	1025.6
2	1875.0	1.1	6.9	2.8	1885.8	2102.7
3	222.2	1.6	115.9	2.5	342.1	433.2
4	91.3	0.5	295.9	79.3	467.1	640.8
5	668.9	0.8	1145.1	269.0	2083.8	2157.1

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

Table 5.1.2a

MAXIMUM DOSE RATES FROM THE HI-TRAC VW FOR NORMAL CONDITIONS
MPC-89 DESIGN BASIS FUEL
REGIONALIZED LOADING BASED ON FIGURE 1.2.6

Dose Point Location	Fuel Gammas (mrem/hr)	(n,γ) Gammas (mrem/hr)	^{60}Co Gammas (mrem/hr)	Neutrons (mrem/hr)	Totals (mrem/hr)
ADJACENT TO THE HI-TRAC VW					
1	424.1	24.4	2212.0	54.1	2714.6
2	4906.7	12.8	<0.1	24.7	4944.2
3	19.1	5.0	661.0	5.7	690.8
4	66.7	1.4	477.8	214.8	760.7
5	216.8	3.0	1922.8	1061.7	3204.3
ONE METER FROM THE HI-TRAC VW					
1	720.0	9.5	247.5	19.7	996.8
2	2218.5	3.6	16.7	8.3	2247.2
3	205.7	4.8	300.8	6.1	517.3
4	40.5	0.5	334.9	66.4	442.3
5	174.7	0.9	1355.9	314.0	1845.6

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

Table 5.1.2b

MAXIMUM DOSE RATES FROM THE HI-TRAC VW FOR NORMAL CONDITIONS
MPC-89 DESIGN BASIS FUEL
REGIONALIZED LOADING BASED ON FIGURE 1.2.7

Dose Point Location	Fuel Gammas (mrem/hr)	(n,γ) Gammas (mrem/hr)	^{60}Co Gammas (mrem/hr)	Neutrons (mrem/hr)	Totals (mrem/hr)
ADJACENT TO THE HI-TRAC VW					
1	478.0	29.5	2419.7	66.1	2993.3
2	5754.5	50.5	<0.1	93.2	5898.2
3	23.0	5.6	716.8	6.3	751.6
4	48.2	2.6	434.7	403.0	888.4
5	166.3	5.7	1797.8	2093.5	4063.3
ONE METER FROM THE HI-TRAC VW					
1	855.9	12.8	259.4	26.7	1154.8
2	2629.5	4.9	17.8	11.4	2663.6
3	237.8	7.3	317.6	9.8	572.5
4	40.7	0.6	352.4	90.4	484.2
5	179.1	1.3	1425.8	431.8	2038.1

Notes:

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

Table 5.1.3

MAXIMUM DOSE RATES FOR ARRAYS OF HI-STORM FWs CONTAINING THE MPC-37
WITH REGIONALIZED LOADING
BASED ON FIGURES 1.2.3 THROUGH 1.2.5

Array Configuration	1 cask	2x2	2x3	2x4	2x5
HI-STORM FW Overpack					
Annual Dose (mrem/year)	9	22	12	16	19
Distance to Controlled Area Boundary (meters)	300	300	400	400	400

Notes:
- Values are rounded up to nearest integer.
- 8760 hour annual occupancy is assumed.
- Dose location is at the center of the long side of the array.
- The bounding regionalized loading source term, consistent with Table 5.1.7 for dose point location 2, is used.

Table 5.1.4a

MAXIMUM DOSE RATES FROM HI-TRAC VW WITH MPC-37
FOR ACCIDENT CONDITIONS AT REGIONALIZED LOADING BURNUP AND COOLING
TIMES
BASED ON FIGURES 1.2.3 THROUGH 1.2.5

Dose Point Location	Fuel Gammas (mrem/hr)	N, Gamma (mrem/hr)	Co-60 Gamma (mrem/hr)	Neutrons (mrem/hr)	Total (mrem/hr)
1 meter from HI-TRAC VW					
2 (Accident Condition)	2587.6	1.1	16.4	1073.7	4059.4
2 (Normal Condition)	1875.0	1.1	6.9	2.8	2102.7
100 meters from HI-TRAC VW					
2 (Accident Condition)	0.3	<0.1	<0.1	0.5	0.8

Notes:
- Refer to Figure 5.1.2 for dose locations.
- The "Fuel Gammas" category includes gammas from the spent fuel and ^{60}Co from the spacer grids.

Table 5.1.4b

MAXIMUM DOSE RATES FROM HI-TRAC VW WITH MPC-89
FOR ACCIDENT CONDITIONS AT REGIONALIZED LOADING BURNUP AND COOLING
TIMES
BASED ON FIGURE 1.2.6

Dose Point Location	Fuel Gammas (mrem/hr)	N, Gamma (mrem/hr)	Co-60 Gamma (mrem/hr)	Neutrons (mrem/hr)	Total (mrem/hr)
1 meter from HI-TRAC VW					
2 (Accident Condition)	3400.4	1.8	33.4	1617.4	5053.0
2 (Normal Condition)	2218.5	3.6	16.7	8.3	2247.2
100 meters from HI-TRAC VW					
2 (Accident Condition)	1.4	<0.1	0.1	<0.1	1.6

Notes:
- Refer to Figure 5.1.2 for dose locations.
- The "Fuel Gammas" category includes gammas from the spent fuel and ^{60}Co from the spacer grids.

Table 5.1.5

MAXIMUM DOSE RATES ADJACENT TO HI-STORM FW OVERPACK
FOR NORMAL CONDITIONS
MPC-37
REGIONALIZED BURNUP AND COOLING TIME BASED ON FIGURES 1.2.3 THROUGH
1.2.5

Dose Point Location	Fuel Gammas (mrem/hr)	(n,γ) Gammas (mrem/hr)	^{60}Co Gammas (mrem/hr)	Neutrons (mrem/hr)	Totals (mrem/hr)	Totals with BPRAs (mrem/hr)
1	386.7	<0.1	10.8	0.2	397.8	430.3
2	252.0	<0.1	<0.1	<0.1	252.2	269.0
3 (surface)	17.0	0.3	15.6	3.7	36.6	46.6
3 (overpack edge)	9.4	0.1	8.6	1.1	19.2	24.6
4 (center)	0.6	3.2	1.0	2.8	7.7	9.2
4 (mid)	16.2	0.9	7.7	3.1	27.9	34.8
4 (outer)	0.8	<0.1	0.5	<0.1	1.3	1.7

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

Table 5.1.6a

MAXIMUM DOSE RATES ADJACENT TO HI-STORM FW OVERPACK
FOR NORMAL CONDITIONS
MPC-89
REGIONALIZED BURNUP AND COOLING TIME BASED ON FIGURE 1.2.6

Dose Point Location	Fuel Gammas (mrem/hr)	(n,γ) Gammas (mrem/hr)	^{60}Co Gammas (mrem/hr)	Neutrons (mrem/hr)	Totals (mrem/hr)
1	239.3	0.2	21.9	0.4	261.8
2	204.3	0.3	<0.1	0.2	204.8
3 (surface)	8.8	0.2	19.8	3.2	32.1
3 (overpack edge)	3.7	0.1	9.8	1.1	14.7
4 (center)	0.4	2.2	1.8	1.9	6.4
4 (mid)	10.0	0.9	13.6	3.0	27.4
4 (outer)	0.6	<0.1	2.0	<0.1	2.6

Notes:

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

Table 5.1.6b

MAXIMUM DOSE RATES ADJACENT TO HI-STORM FW OVERPACK
FOR NORMAL CONDITIONS
MPC-89
REGIONALIZED BURNUP AND COOLING TIME BASED ON FIGURE 1.2.7

Dose Point Location	Fuel Gammas (mrem/hr)	(n,γ) Gammas (mrem/hr)	^{60}Co Gammas (mrem/hr)	Neutrons (mrem/hr)	Totals (mrem/hr)
1	293.1	0.3	23.5	0.5	317.4
2	238.1	0.4	<0.1	0.3	238.8
3 (surface)	9.9	0.3	21.3	4.2	35.7
3 (overpack edge)	4.3	0.1	10.6	1.2	16.2
4 (center)	0.4	3.2	2.0	2.7	8.4
4 (mid)	10.3	1.7	13.7	5.8	31.5
4 (outer)	0.6	<0.1	2.1	<0.1	2.8

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

Table 5.1.7

MAXIMUM DOSE RATES AT ONE METER FROM HI-STORM FW OVERPACK
FOR NORMAL CONDITIONS
MPC-37
REGIONALIZED BURNUP AND COOLING TIME
BASED ON FIGURES 1.2.3 THROUGH 1.2.5

Dose Point Location	Fuel Gammas (mrem/hr)	(n,γ) Gammas (mrem/hr)	^{60}Co Gammas (mrem/hr)	Neutrons (mrem/hr)	Totals (mrem/hr)	Totals with BPRAs (mrem/hr)
1	92.5	<0.1	2.3	<0.1	94.9	102.0
2	129.1	<0.1	0.5	<0.1	129.7	139.0
3	10.4	<0.1	2.3	<0.1	12.6	15.0
4 (center)	3.2	0.7	3.1	1.4	8.5	10.6

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

Table 5.1.8a

MAXIMUM DOSE RATES AT ONE METER FROM HI-STORM FW OVERPACK
FOR NORMAL CONDITIONS
MPC-89
REGIONALIZED BURNUP AND COOLING TIME BASED ON FIGURE 1.2.6

Dose Point Location	Fuel Gammas (mrem/hr)	(n,γ) Gammas (mrem/hr)	^{60}Co Gammas (mrem/hr)	Neutrons (mrem/hr)	Totals (mrem/hr)
1	68.6	<0.1	5.7	<0.1	74.5
2	106.8	0.1	0.7	<0.1	107.7
3	5.8	<0.1	3.3	<0.1	9.2
4 (center)	1.7	0.6	3.8	1.2	7.3

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

Table 5.1.8b

MAXIMUM DOSE RATES AT ONE METER FROM HI-STORM FW OVERPACK
FOR NORMAL CONDITIONS
MPC-89
REGIONALIZED BURNUP AND COOLING TIME BASED ON FIGURE 1.2.7

Dose Point Location	Fuel Gammas (mrem/hr)	(n,γ) Gammas (mrem/hr)	^{60}Co Gammas (mrem/hr)	Neutrons (mrem/hr)	Totals (mrem/hr)
1	81.0	<0.1	6.1	0.1	87.3
2	126.3	0.2	0.8	0.1	127.4
3	6.6	<0.1	3.7	0.1	10.4
4 (center)	1.6	1.2	3.7	2.3	8.8

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

Table 5.1.9

MAXIMUM DOSE FROM HI-TRAC VW
FOR ACCIDENT CONDITIONS
AT 100 METERS

Dose Point Location	Dose Rate (rem/hr)	Accident Duration (days)	Total Dose (rem)	Regulatory Limit (rem)	Time to Reach Regulatory Limit (days)
2 (Accident Condition)	1.6E-03	30	1.15	5	130

Notes:
- Refer to Figure 5.1.2 for dose locations.
- Values are rounded to nearest integer where appropriate.
- Dose rate used to evaluated "Total Dose (rem)" is the maximum from Tables 5.1.4a and 5.1.4b.
- Regulatory Limit is from 10CFR72.106.

| Table 5.1.10

DOSE RATES FROM THE HI-TRAC VW VERSION V2 FOR NORMAL CONDITIONS, DRY MPC-89 WITH NEUTRON SHIELD CYLINDER PRESENT, BASED ON FIGURE 1.2.7

Dose Point Location	Fuel Gammas (mrem/hr)	(n,γ) Gammas (mrem/hr)	^{60}Co Gammas (mrem/hr)	Neutrons (mrem/hr)	Totals (mrem/hr)
ADJACENT TO THE HI-TRAC VW					
1	1020	2	5794	362	7178
1*	104	2	675	384	1164
2	1412	19	< 1	154	1585
3	4	2	144	137	286
4	134	3	515	427	1078
5	174	6	1799	2123	4102
ONE METER FROM THE HI-TRAC VW					
1	345	3	63	31	441
1*	339	3	52	30	425
2	653	6	4	51	715
3	35	1	54	11	100
4	140	1	490	103	734
5	186	1	1430	462	2080

Notes:
* Location 1* uses a steel shield ring pedestal for the Neutron Shield Cylinder, which may be present for ALARA purposes. The critical shielding dimensions of the optional steel shield ring pedestal are as follows: Outer Diameter is 8 feet; radial thickness is 2.5 inches; Axial bottom of shield ring is 3 inches below MPC baseplate bottom surface; top of shield ring is in contact with Neutron Shield Cylinder.
- Refer to Figure 5.1.2 for dose locations.
- Values are rounded to nearest integer.
- Dose rates are based on no water within the MPC, an empty annulus, and the Neutron Shield Cylinder present. For the majority of the duration that the HI-TRAC bottom lid is installed, the MPC cavity will be flooded with water. The water within the MPC greatly reduces the dose rate.
- Streaming may occur through the annulus. However, during handling/operations the annulus is filled with water and lead snakes are typically present to reduce the streaming effects.
- The "Fuel Gammas" category includes gammas from the spent fuel and ^{60}Co from the spacer grids.
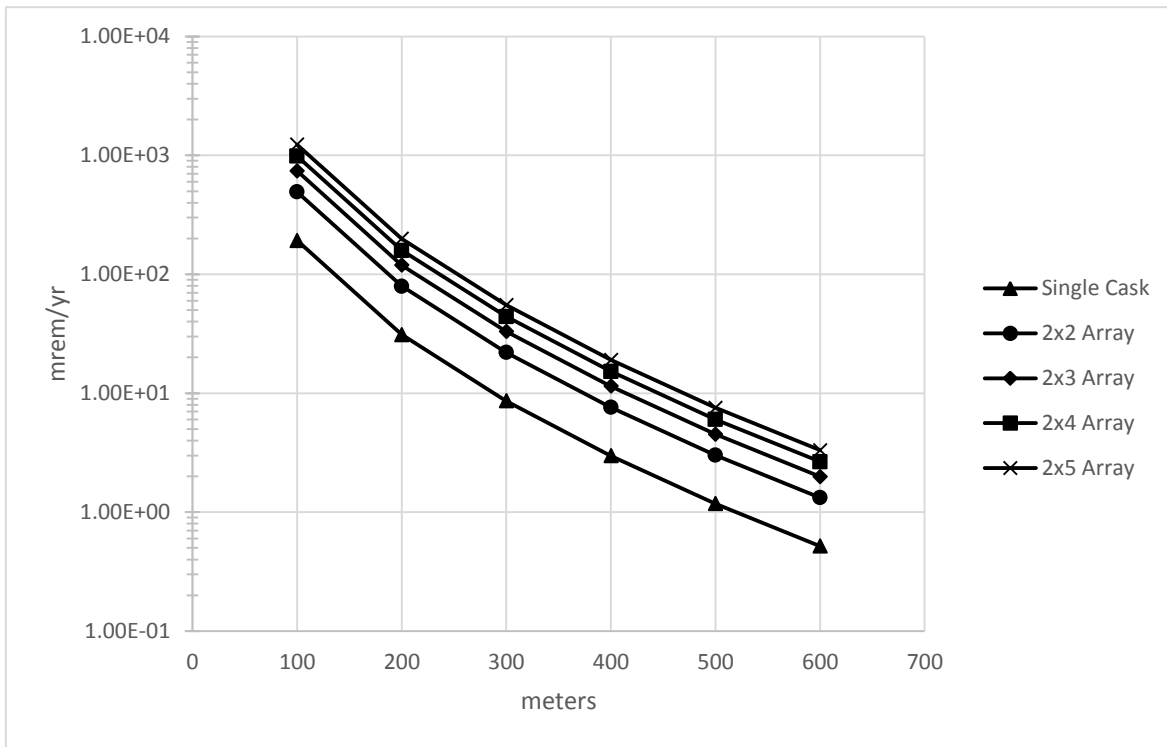
HOLTEC INTERNATIONAL COPYRIGHTED MATERIAL

Figure 5.1.3

ANNUAL DOSE VERSUS DISTANCE FOR VARIOUS CONFIGURATIONS OF THE MPC-37 FOR REGIONALIZED LOADING BASED ON FIGURES 1.2.3 THROUGH 1.2.5 (8760 HOUR OCCUPANCY ASSUMED)

Figure 5.1.4

FIGURE DELETED

5.2   SOURCE SPECIFICATION

The design basis neutron and gamma source terms, decay heat values, and quantities of radionuclides available for release were calculated with the TRITON and ORIGAMI sequences of the SCALE 6.2.1 system [5.1.4], which is consistent with other approved Holtec applications [5.2.18]. For some additional calculations presented in Section 5.4, the neutron and gamma source terms available were calculated with the SAS2H and ORIGEN-S modules of the SCALE 5 system [5.1.2, 5.1.3]. SAS2H has been extensively compared to experimental isotopic validations and decay heat measurements. References [5.2.8] through [5.2.12] and [5.2.15] present isotopic comparisons for PWR and BWR fuels for burnups ranging to 47 GWD/MTU and reference [5.2.13] presents results for BWR measurements to a burnup of 57 GWD/MTU. A comparison of calculated and measured decays heats is presented in reference [5.2.14]. All of these studies indicate good agreement between SAS2H and measured data.

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

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

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

5.2.1 Gamma Source

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

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

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

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

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

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

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

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

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

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

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

5.2.2 Neutron Source

It is well known that the neutron source strength increases as enrichment decreases, for a constant burnup and decay time. This is due to the increase in Pu content in the fuel, which increases the inventory of other transuranium nuclides such as Cm. The gamma source also varies with enrichment, although only slightly. Because of this effect and in order to obtain conservative source terms, low initial fuel enrichments were chosen for the BWR and PWR design basis fuel assemblies under normal and accident conditions, respectively, as discussed in Subsection 5.2.8.

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

5.2.3 Non-Fuel Hardware

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

5.2.3.1 BPRAs and TPDs

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

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

neutron sources (e.g. californium, americium-beryllium, plutonium-beryllium, polonium-beryllium, antimony-beryllium). These neutron sources are typically inserted into the water rod of a fuel assembly and are usually removable.

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

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

5.2.7 Design Basis Burnup and Cooling Times

For the fuel to be loaded into the HI-STORM FW system, the regionalized design basis loading curves (which specify burnup and cooling time combinations for each region of the cask) are provided in Table 2.1.10 using polynomial equation and corresponding polynomial coefficients.

In order to qualify the HI-STORM FW System with allowable burnup, cooling time combinations in Table 2.1.10, the considered range of burnup, enrichment and cooling time combinations is selected as follows:

- 5 GWD/MTU burnup and burnups from 10 GWD/MTU to 70 GWD/MTU, in increments of 10 GWD/MTU;
- The cooling time is calculated for each burnup using the equation and polynomial coefficients in Table 2.1.10. The determined cooling times are rounded down to the nearest available cooling time in the calculated source terms library, which provides a significant conservatism, especially, in the low cooling time area. The value of 1 year (minimum allowed cooling time) is used for all cooling times below 1 year;
- The appropriate burnup-specific lower bound enrichment is selected according to Table 5.2.17.

The final sets of the burnup, enrichment and cooling time combinations are provided in Tables 5.0.3 and 5.0.4.

5.2.8 Fuel Enrichment

As discussed in Subsection 5.2.2, enrichments have a significant impact on neutron dose rates, with lower enrichments resulting in higher dose rates at the same burnup. For assemblies with higher burnups (which result in high neutron source terms) and/or locations that are more neutron dominated, the enrichment would therefore be important in order to present dose rates in a conservative way. However, it would be impractical and excessively conservative to perform all

calculations at bounding low enrichment, since low enrichments are generally only found in lower burned assemblies. Therefore, a conservatively low enrichment value is selected based on the burnup. Specifically, based on industry information on more than 60,000 PWR assemblies, the lower bound enrichment that covers 99% of available data is determined for each selected burnup range.   The calculated and finally established lower bound enrichment values are summarized in Table 5.2.17.

Given that the considered baskets contain a relatively large number of available cells for fuel loading, selecting the minimum enrichment for all assemblies is considered reasonably conservative. The typical content of the basket would have most assemblies well above the lower bound enrichment assumed in the analyses, so even if a small number of assemblies would be below the assumed minimum, that would have a negligible effect or be essentially inconsequential for the dose rates around the cask. Furthermore, the site-specific shielding analysis shall consider actual or bounding fuel enrichment. Therefore, an explicit lower enrichment limit for the fuel assemblies is not considered necessary.

Table 5.2.2				
CALCULATED MPC-37 PWR FUEL GAMMA SOURCE PER ASSEMBLY FOR A SELECTED BURNUP AND COOLING TIME FOR NORMAL CONDITIONS				
Lower Energy	**Upper Energy**	**30,000 MWD/MTU 4-Year Cooling**	
(MeV)	(MeV)	(MeV/s)	(Photons/s)
0.45	0.7	1.44E+15	2.50E+15
0.7	1.0	6.08E+14	7.15E+14
1.0	1.5	1.44E+14	1.16E+14
1.5	2.0	1.19E+13	6.78E+12
2.0	2.5	1.34E+13	5.95E+12
2.5	3.0	1.07E+12	3.88E+11
Total		2.21E+15	3.34E+15

Table 5.2.3				
CALCULATED MPC-37 PWR FUEL GAMMA SOURCE PER ASSEMBLY FOR BURNUP AND COOLING TIME FOR ACCIDENT CONDITIONS				
Lower Energy	**Upper Energy**	**70,000 MWD/MTU 2.8-Year Cooling**	
(MeV)	(MeV)	(MeV/s)	(Photons/s)
0.45	0.7	4.83E+15	8.40E+15
0.7	1.0	2.91E+15	3.42E+15
1.0	1.5	5.49E+14	4.39E+14
1.5	2.0	4.12E+13	2.36E+13
2.0	2.5	4.49E+13	2.00E+13
2.5	3.0	3.80E+12	1.38E+12
Total		8.37E+15	1.23E+16

Table 5.2.4		
CALCULATED MPC-89 BWR FUEL GAMMA SOURCE PER ASSEMBLY FOR A SELECTED BURNUP AND COOLING TIME FOR NORMAL CONDITIONS		
Lower Energy	**Upper Energy**	**40,000 MWD/MTU 3.5-Year Cooling**	
(MeV)	(MeV)	(MeV/s)	(Photons/s)
0.45	0.7	8.07E+14	1.40E+15
0.7	1.0	3.72E+14	4.38E+14
1.0	1.5	8.11E+13	6.49E+13
1.5	2.0	6.36E+12	3.63E+12
2.0	2.5	6.88E+12	3.06E+12
2.5	3.0	5.69E+11	2.07E+11
Total		1.27E+15	1.91E+15

Table 5.2.5		
CALCULATED MPC-89 BWR FUEL GAMMA SOURCE PER ASSEMBLY FOR BURNUP AND COOLING TIME FOR ACCIDENT CONDITIONS		
Lower Energy	**Upper Energy**	**70,000 MWD/MTU 4-Year Cooling**	
(MeV)	(MeV)	(MeV/s)	(Photons/s)
0.45	0.7	1.91E+15	3.32E+15
0.7	1.0	1.13E+15	1.33E+15
1.0	1.5	2.10E+14	1.68E+14
1.5	2.0	1.70E+13	9.70E+12
2.0	2.5	1.80E+13	8.00E+12
2.5	3.0	1.60E+12	5.84E+11
Total		3.29E+15	4.84E+15

Table 5.2.7

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

Location	30,000 MWD/MTU and 4-Year Cooling (curies)
Lower End Fitting	73.84
Gas Plenum Springs	14.39
Gas Plenum Spacer	10.22
Expansion Springs	N/A
Incore Grid Spacers	306.62
Upper End Fitting	98.24
Handle	N/A

Table 5.2.8

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

Location	70,000 MWD/MTU and 2.8-Year Cooling (curies)
Lower End Fitting	133.42
Gas Plenum Springs	26.01
Gas Plenum Spacer	18.48
Expansion Springs	N/A
Incore Grid Spacers	554.05
Upper End Fitting	177.52
Handle	NA

Table 5.2.9

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

Location	40,000 MWD/MTU and 3.5-Year Cooling (curies)
Lower End Fitting	57.08
Gas Plenum Springs	17.44
Gas Plenum Spacer	N/A
Expansion Springs	3.17
Grid Spacer Springs	26.16
Upper End Fitting	15.86
Handle	1.98

Table 5.2.10

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

Location	70,000 MWD/MTU and 4-Year Cooling (curies)
Lower End Fitting	92.98
Gas Plenum Springs	28.41
Gas Plenum Spacer	N/A
Expansion Springs	5.17
Grid Spacer Springs	42.61
Upper End Fitting	25.83
Handle	3.23

Table 5.2.11

CALCULATED MPC-37 PWR NEUTRON SOURCE PER
ASSEMBLY FOR A SELECTED BURNUP AND COOLING TIME
FOR NORMAL CONDITIONS

Lower Energy (MeV)	Upper Energy (MeV)	30,000 MWD/MTU 4-Year Cooling (Neutrons/s)
1.0e-01	4.0e-01	1.13E+07
4.0e-01	9.0e-01	2.47E+07
9.0e-01	1.4	2.47E+07
1.4	1.85	1.97E+07
1.85	3.0	3.67E+07
3.0	6.43	3.34E+07
6.43	20.0	3.18E+06
Totals		1.54E+08

Table 5.2.12

CALCULATED MPC-37 PWR NEUTRON SOURCE PER
ASSEMBLY FOR BURNUP AND COOLING TIME FOR
ACCIDENT CONDITIONS

Lower Energy (MeV)	Upper Energy (MeV)	70,000 MWD/MTU 2.8-Year Cooling (Neutrons/s)
1.0e-01	4.0e-01	1.17E+08
4.0e-01	9.0e-01	2.55E+08
9.0e-01	1.4	2.55E+08
1.4	1.85	2.03E+08
1.85	3.0	3.77E+08
3.0	6.43	3.45E+08
6.43	20.0	3.36E+07
Totals		1.59E+09

Table 5.2.13 CALCULATED MPC-89 BWR NEUTRON SOURCE PER ASSEMBLY FOR DESIGN BASIS FUEL FOR A SELECTED BURNUP AND COOLING TIME FOR NORMAL CONDITIONS		
Lower Energy (MeV)	**Upper Energy (MeV)**	**40,000 MWD/MTU 3.5-Year Cooling (Neutrons/s)**
1.0e-01	4.0e-01	9.33E+06
4.0e-01	9.0e-01	2.03E+07
9.0e-01	1.4	2.03E+07
1.4	1.85	1.62E+07
1.85	3.0	3.01E+07
3.0	6.43	2.74E+07
6.43	20.0	2.63E+06
Totals		1.26E+08

Table 5.2.14
CALCULATED MPC-89 BWR NEUTRON SOURCE PER ASSEMBLY FOR DESIGN BASIS FUEL FOR BURNUP AND COOLING TIME FOR ACCIDENT CONDITIONS

Lower Energy (MeV)	Upper Energy (MeV)	70,000 MWD/MTU 4-Year Cooling (Neutrons/s)
1.0e-01	4.0e-01	5.43E+07
4.0e-01	9.0e-01	1.18E+08
9.0e-01	1.4	1.18E+08
1.4	1.85	9.43E+07
1.85	3.0	1.75E+08
3.0	6.43	1.61E+08
6.43	20.0	1.58E+07
Totals		7.37E+08

Table 5.2.17

LOWER BOUND INITIAL ENRICHMENTS USED IN THE SOURCE TERM CALCULATIONS[1]

Burnup Range[2] (MWD/MTU)	Initial Enrichment (wt.% ^{235}U)	
	PWR Fuel	BWR Fuel
0,000-5,000	0.7	0.7
5,000-10,000	1.1	0.7
10,000-15,000	1.1	0.9
15,000-20,000	1.1	1.5
20,000-25,000	1.6	1.6
25,000-30,000	2.0	2.0
30,000-35,000	2.4	2.4
35,000-40,000	2.6	2.7
40,000-45,000	3.0	3.0
45,000-50,000	3.3	3.2
50,000-55,000	3.6	3.3
55,000-60,000	3.6	3.7
60,000-65,000	3.9	3.7
65,000-70,000	4.2	3.7

Notes:
1. Burnup and initial enrichments listed in this table are used in source term calculations for the shielding evaluation of the loading patterns in Figures 1.2.3 through 1.2.7.
2. The burnup ranges do not overlap. Therefore, 20,000-25,000 MWD/MTU means 20,000-24,999.9 MWD/MTU, etc. This note does not apply to the maximum burnups of 65,000 and 70,000 MWD/MTU.

Table 5.2.18

Table Deleted

Figure 5.3.14 shows a cross sectional view of the HI-TRAC VW Version V2 with the Neutron Shield Cylinder and MPC-89, as it was modeled in MCNP. Figure 5.3.15 shows a cross sectional view of the HI-TRAC VW Version V2 with the MPC-89, in which the MPC and annulus between the MPC and HI-TRAC inner cavity are filled with water, as it was modeled in MCNP.

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

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

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

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

Conservatively, the zircaloy flow channels are not included in the modeling of the BWR assemblies.

Also, it should be noted that all dose calculations presented in this Chapter are performed with the HI-TRAC VW (standard) model unless otherwise noted. Site specific analysis of the HI-TRAC VW should consider the specific version of the HI-TRAC VW (for example, HI-TRAC VW (standard), HI-TRAC VW Version P, HI-TRAC VW Version V, HI-TRAC VW Version V2). Additionally, the HI-TRAC VW radial lead thickness, which is a site specific feature that is maximized to the extent possible without exceeding the site crane capacity or site dimensional constraints, is also considered in site specific shielding evaluations.

5.3.1.1   Fuel Configuration

As described earlier, the active fuel region is modeled as a homogenous zone. The end fittings and the plenum regions are also modeled as homogenous regions of steel. The masses of steel

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

Table 5.4.4 shows the corresponding dose rates adjacent to and one meter away from the HI-TRAC for the fully flooded MPC-89 condition with an empty water-jacket. Table 5.4.5 shows the dose rates adjacent to and one meter away from the HI-TRAC for the fully flooded MPC-89 condition with the water jacket filled with water. These results demonstrate that the dose rates on contact at the top and bottom of the HI-TRAC VW are somewhat higher in the MPC-37 case than in the MPC-89 case. The difference in dose rates between MPC-37 and MPC-89 is within approximately 30%. Therefore, the MPC-37 is sufficiently representative for the exposure calculations in Chapter 11 of the SAR.

Previous revisions of this FSAR with a smaller set of loading patterns used a representative (while still conservative) uniform loading conditions for dose evaluations, and analyses for the concrete overpack were performed with the standard lid. For reference purposes, those results are retained in this subsection of the chapter, in Tables 5.4.9 and 5.4.11 through 5.4.14. Table 5.4.9 shows the burnup, enrichment and cooling time combinations that were used. Tables 5.4.11 and 5.4.12 provide the dose rates adjacent to the HI-STORM overpack with the standard lid design during normal conditions for the MPC-37 and MPC-89, respectively. And Tables 5.4.13 and 5.4.14 provide the dose rates at one meter from the HI-STORM overpack with the standard lid design during normal conditions for the MPC-37 and MPC-89, respectively. The dose rates adjacent to and one meter from the HI-TRAC VW for normal conditions (i.e., dry MPC and full water jacket) and uniform loading source terms for the MPC-37 and MPC-89 are listed in Tables 5.4.15 and 5.4.16, respectively.

It should be noted that zircalow flow channels are included in the modeling of the BWR assemblies for HI-STORM FW with MPC-89 and the standard lid. The effect of this deviation is insignificant.

Table 5.4.17 shows the corresponding dose rates adjacent to and one meter away from the HI-TRAC VW Version V2 for the fully flooded MPC-89, flooded annulus between MPC and HI-TRAC Inner Cavity. Table 5.4.18 shows the dose rates adjacent to and one meter away from the HI-TRAC VW Version V2 for the fully flooded MPC-89, flooded annulus between MPC and

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

The representative annual dose, assuming 100% occupancy (8760 hours), at 300 meters from a single HI-STORM FW cask is presented in Table 5.4.6.

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

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

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

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

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

Z = number of casks along long side

Dose = ZA + 2ZB + ZC

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

5.4.4 Non-Fuel Hardware

As discussed in Subsection 5.2.3, non-fuel hardware in the form of BPRAs, TPDs, CRAs, and APSRs are permitted for storage, integral with a PWR fuel assembly, in the HI-STORM FW system. Since each device occupies the same location within an assembly, only one device will be present in a given assembly. ITTRs, which are installed after core discharge and do not

CRA is comparable to or higher than the value from the BPRA. The increase in the bottom dose rates due to the presence of CRAs is on the order of 10-15% (based on bounding configuration 1 in [5.2.17]). The dose rate out the top of the overpack is essentially 0. The latter is due to the fact that CRAs and APSRs do not achieve significant activation in the upper portion of the devices due to the manner in which they are utilized during normal reactor operations. In contrast, the dose rate out the bottom of the overpack is substantial due to these devices. However, these dose rates occur in an area (below the pool lid and transfer doors) which is not normally occupied.

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

5.4.5 Effect of Uncertainties

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

5.4.6 Dose Rate Evaluation for Fuel Assemblies with Irradiated Stainless Steel Replacement Rods or BLEU Fuel

Some fuel assemblies may contain irradiated stainless steel rods or BLEU fuel material. From shielding perspective, assemblies containing Blended Low Enriched Uranium (BLEU) fuel material are essentially identical to UO_2 fuel except for the presence of a higher quantity of cobalt impurity.

A dose rate evaluation for the HI-STORM FW containing the MPC-37 and the MPC-89 is performed to determine the impact of storing fuel assemblies with irradiated stainless steel replacement rods.

The stainless steel rods are irradiated in the same neutron flux and for the same time period as the design basis PWR and BWR UO_2 fuel rods. As an example, the dose rates at the same locations are evaluated assuming all 37 design basis PWR assemblies contain 4 irradiated stainless steel replacement rods and all 89 design basis BWR assemblies contain 2 irradiated stainless steel

Table 5.4.2
MAXIMUM DOSE RATES FOR THE HI-TRAC VW FOR THE FULLY FLOODED MPC CONDITION WITH AN EMPTY NEUTRON SHIELD
MPC-37 DESIGN BASIS ZIRCALOY CLAD FUEL
REGIONALIZED LOADING BASED ON FIGURES 1.2.3 THROUGH 1.2.5

Dose Point Location	Fuel Gammas (mrem/hr)	(n,γ) Gammas (mrem/hr)	^{60}Co Gammas (mrem/hr)	Neutrons (mrem/hr)	Totals (mrem/hr)	Totals with BPRAs (mrem/hr)
ADJACENT TO THE HI-TRAC VW						
1	804.4	<0.1	422.8	17.3	1244.6	1362.6
2	2627.9	0.7	<0.1	131.4	2760.1	3147.9
3	8.7	<0.1	129.2	2.8	140.7	213.7
4	26.9	<0.1	201.3	0.2	228.5	357.5
5 (bottom lid)	460.4	0.2	1509.1	70.4	2040.1	2103.2
ONE METER FROM THE HI-TRAC VW						
1	525.3	0.1	40.8	28.2	594.5	679.3
2	1170.7	0.3	3.5	60.3	1234.8	1431.3
3	133.8	<0.1	54.7	9.5	198.1	253.9
4	14.7	<0.1	128.0	0.2	142.9	215.5
5	388.8	<0.1	982.9	15.4	1387.2	1423.8

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

Table 5.4.3						
MAXIMUM DOSE RATES FOR THE HI-TRAC VW FOR THE FULLY FLOODED MPC CONDITION WITH A FULL NEUTRON SHIELD MPC-37 DESIGN BASIS ZIRCALOY CLAD FUEL REGIONALIZED LOADING BASED ON FIGURES 1.2.3 THROUGH 1.2.5						
Dose Point Location	**Fuel Gammas (mrem/hr)**	**(n,γ) Gammas (mrem/hr)**	**^{60}Co Gammas (mrem/hr)**	**Neutrons (mrem/hr)**	**Totals (mrem/hr)**	**Totals with BPRAs (mrem/hr)**
ADJACENT TO THE HI-TRAC VW						
1	513.4	0.1	254.2	0.7	768.5	837.8
2	1684.2	0.2	<0.1	1.1	1685.4	1910.3
3	3.5	<0.1	69.7	<0.1	73.3	112.3
4	26.9	<0.1	201.3	0.2	228.4	357.5
5 (bottom lid)	459.9	0.2	1508.9	70.2	2039.2	2102.0
ONE METER FROM THE HI-TRAC VW						
1	344.6	<0.1	21.3	0.1	366.0	406.7
2	770.9	<0.1	1.9	0.3	773.2	862.5
3	85.5	<0.1	23.4	0.2	109.0	140.0
4	14.7	<0.1	128.0	<0.1	142.8	215.4
5	388.4	<0.1	982.9	14.9	1386.3	1422.8

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

Table 5.4.4a					
MAXIMUM DOSE RATES FOR THE HI-TRAC VW FOR THE FULLY FLOODED MPC CONDITION WITH AN EMPTY NEUTRON SHIELD MPC-89 DESIGN BASIS ZIRCALOY CLAD FUEL REGIONALIZED LOADING BASED ON FIGURE 1.2.6					
Dose Point Location	**Fuel Gammas (mrem/hr)**	**(n,γ) Gammas (mrem/hr)**	**^{60}Co Gammas (mrem/hr)**	**Neutrons (mrem/hr)**	**Totals (mrem/hr)**
ADJACENT TO THE HI-TRAC VW					
1	350.5	0.2	1372.9	27.0	1750.6
2	4084.7	2.3	<0.1	397.0	4484.0
3	2.1	<0.1	370.0	2.8	375.0
4	4.7	<0.1	201.0	<0.1	205.8
5 (bottom lid)	76.5	<0.1	1366.8	4.1	1447.4
ONE METER FROM THE HI-TRAC VW					
1	637.9	0.2	140.2	29.9	808.3
2	1895.9	0.4	11.8	83.3	1991.5
3	154.9	0.1	143.4	18.7	317.1
4	2.5	<0.1	146.8	0.1	149.4
5	63.9	<0.1	971.4	2.5	1037.8

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

Dose Point Location	Fuel Gammas (mrem/hr)	(n,γ) Gammas (mrem/hr)	^{60}Co Gammas (mrem/hr)	Neutrons (mrem/hr)	Totals (mrem/hr)
Table 5.4.4b **MAXIMUM DOSE RATES FOR THE HI-TRAC VW FOR THE FULLY FLOODED MPC CONDITION WITH AN EMPTY NEUTRON SHIELD** **MPC-89 DESIGN BASIS ZIRCALOY CLAD FUEL** **REGIONALIZED LOADING BASED ON FIGURE 1.2.7**					
ADJACENT TO THE HI-TRAC VW					
1	373.8	0.3	1489.0	53.0	1916.2
2	4846.4	2.1	<0.1	364.6	5213.1
3	2.4	<0.1	403.4	3.4	409.2
4	4.4	<0.1	220.4	<0.1	224.9
5 (bottom lid)	84.1	<0.1	1484.3	7.9	1576.3
ONE METER FROM THE HI-TRAC VW					
1	756.5	0.3	147.5	41.5	945.7
2	2232.2	0.6	9.7	117.6	2360.1
3	174.0	0.2	159.6	24.1	357.9
4	3.1	<0.1	158.3	0.1	161.5
5	64.0	<0.1	1053.6	3.8	1121.5

Notes:

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

Table 5.4.5a

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

MPC-89 DESIGN BASIS ZIRCALOY CLAD FUEL

REGIONALIZED LOADING BASED ON FIGURE 1.2.6

Dose Point Location	Fuel Gammas (mrem/hr)	(n,γ) Gammas (mrem/hr)	^{60}Co Gammas (mrem/hr)	Neutrons (mrem/hr)	Totals (mrem/hr)
ADJACENT TO THE HI-TRAC VW					
1	205.2	0.3	851.4	1.4	1058.3
2	2602.2	0.8	<0.1	4.0	2606.9
3	0.6	<0.1	202.8	<0.1	203.5
4	4.7	<0.1	201.0	<0.1	205.8
5 (bottom lid)	76.5	<0.1	1366.9	4.2	1447.6
ONE METER FROM THE HI-TRAC VW					
1	397.3	<0.1	83.4	0.4	481.3
2	1179.3	0.2	4.5	1.4	1185.5
3	89.4	0.2	84.3	0.8	174.8
4	2.5	<0.1	146.8	<0.1	149.3
5	64.0	<0.1	971.3	2.3	1037.5

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

Dose Point Location	Fuel Gammas (mrem/hr)	(n,γ) Gammas (mrem/hr)	^{60}Co Gammas (mrem/hr)	Neutrons (mrem/hr)	Totals (mrem/hr)
ADJACENT TO THE HI-TRAC VW					
1	215.1	0.6	922.7	2.7	1141.1
2	3039.2	1.1	<0.1	6.0	3046.3
3	1.1	<0.1	220.8	<0.1	221.9
4	4.4	<0.1	220.4	<0.1	224.8
5 (bottom lid)	84.2	<0.1	1484.0	8.2	1576.5
ONE METER FROM THE HI-TRAC VW					
1	465.7	0.1	82.3	0.7	548.8
2	1394.8	0.3	4.8	2.1	1401.9
3	122.4	0.1	76.8	0.5	199.8
4	3.1	<0.1	158.3	<0.1	161.4
5	64.0	<0.1	1053.5	3.4	1121.0

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

Table 5.4.6

ANNUAL DOSE AT 300 METERS FROM A SINGLE
HI-STORM FW OVERPACK WITH THE XL LID DESIGN CONTAINING
AN MPC-37 WITH DESIGN BASIS
ZIRCALOY CLAD FUEL
FOR REPRESENTATIVE BURNUP AND COOLING TIME COMBINATION

Dose Component	45,000 MWD/MTU 4.5-Year Cooling (mrem/yr)
Fuel gammas	16.35
^{60}Co Gammas	1.11
Neutrons	0.25
Total	17.7

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

Distance	A Side of Overpack (mrem/yr)	B Top of Overpack (mrem/yr)	C Side of Shielded Overpack (mrem/yr)
100 meters	396.8	44.1	79.4
200 meters	60.9	6.8	12.2
300 meters	15.9	1.8	3.2
400 meters	5.2	0.6	1.0
500 meters	1.9	0.2	0.4
600 meters	0.8	0.1	0.2

Table 5.4.7

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

(REPRESENTATIVE BURNUP AND COOLING TIME COMBINATION)

Notes:
- 8760 hour annual occupancy is assumed.

Table 5.4.9

REPRESENTATIVE FUEL BURNUP, COOLING TIME AND ENRICHMENT FOR
NORMAL CONDITIONS

Representative Burnup and Cooling Times Uniform Loading	
MPC-37	**MPC-89**
45,000 MWD/MTU 4.5 Year Cooling 3.6 wt% U-235 Enrichment	45,000 MWD/MTU 5 Year Cooling 3.2 wt% U-235 Enrichment

Table 5.4.10

Table Deleted

Table 5.4.11

DOSE RATES ADJACENT TO HI-STORM FW OVERPACK WITH THE STANDARD LID
DESIGN
FOR NORMAL CONDITIONS
MPC-37
BURNUP AND COOLING TIME
45,000 MWD/MTU AND 4.5-YEAR COOLING
(REPRESENTATIVE BURNUP AND COOLING TIME COMBINATION)

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

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

Table 5.4.12

DOSE RATES ADJACENT TO HI-STORM FW OVERPACK WITH THE STANDARD LID
DESIGN
FOR NORMAL CONDITIONS
MPC-89
BURNUP AND COOLING TIME
45,000 MWD/MTU AND 5-YEAR COOLING
(REPRESENTATIVE BURNUP AND COOLING TIME COMBINATION)

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

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

Table 5.4.13

DOSE RATES AT ONE METER FROM HI-STORM FW OVERPACK WITH THE
STANDARD LID DESIGN
FOR NORMAL CONDITIONS
MPC-37
BURNUP AND COOLING TIME
45,000 MWD/MTU AND 4.5-YEAR COOLING
(REPRESENTATIVE BURNUP AND COOLING TIME COMBINATION)

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

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

Table 5.4.14

DOSE RATES AT ONE METER FROM HI-STORM FW OVERPACK WITH THE
STANDARD LID DESIGN
FOR NORMAL CONDITIONS
MPC-89
BURNUP AND COOLING TIME
45,000 MWD/MTU AND 5-YEAR COOLING
(REPRESENTATIVE BURNUP AND COOLING TIME COMBINATION)

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

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

Table 5.4.15

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

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

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

Table 5.4.16

DOSE RATES FROM THE HI-TRAC VW FOR NORMAL CONDITIONS
MPC-89 DESIGN BASIS FUEL
45,000 MWD/MTU AND 5-YEAR COOLING
(REPRESENTATIVE BURNUP AND COOLING TIME COMBINATION)

Dose Point Location	Fuel Gammas (mrem/hr)	(n,γ) Gammas (mrem/hr)	^{60}Co Gammas (mrem/hr)	Neutrons (mrem/hr)	Totals (mrem/hr)
ADJACENT TO THE HI-TRAC VW					
1	244	18	2247	40	2549
2	2466	107	<1	219	2793
3	3	3	581	4	591
4	25	<1	505	138	669
5	132	2	2135	720	2989
ONE METER FROM THE HI-TRAC VW					
1	411	13	291	29	744
2	1142	30	21	74	1267
3	119	5	280	8	412
4	16	<1	300	43	360
5	79	<1	1202	202	1484

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

<div align="center">

Table 5.4.17

DOSE RATES FOR THE HI-TRAC VW VERSION V2 FOR THE FULLY FLOODED
MPC AND FLOODED ANNULUS CONDITION WITHOUT NEUTRON SHIELD
CYLINDER PRESENT, BASED ON FIGURE 1.2.7

MPC-89 DESIGN BASIS ZIRCALOY CLAD FUEL

</div>

Dose Point Location	Fuel Gammas (mrem/hr)	(n,γ) Gammas (mrem/hr)	^{60}Co Gammas (mrem/hr)	Neutrons (mrem/hr)	Totals (mrem/hr)
ADJACENT TO THE HI-TRAC VW VERSION V2					
1	277.3	<1	3383.0	14.7	3675.1
2	4890.2	2.1	<1	227.7	5120.1
3	1.2	<1	331.3	<1	332.5
4	4.3	<1	253.8	<1	258.1
5 (bottom lid)	92.1	<1	1486.5	8.9	1587.5
ONE METER FROM THE HI-TRAC VW VERSION V2					
1	1175.5	<1	209.3	15.6	1400.6
2	2274.8	<1	9.7	31.8	2316.7
3	91.7	<1	106.4	6.3	204.5
4	3.1	<1	210.6	<1	213.8
5	65.9	<1	1060.2	3.3	1129.4

Notes:
- Refer to Figure 5.1.2 for dose point locations.
- Values are rounded to nearest integer.
- MPC internal water level is 5 inches below the MPC lid.
- Annulus water level is 2 inches above bottom surface of the MPC lid.
- The "Fuel Gammas" category includes gammas from the spent fuel and ^{60}Co from the spacer grids.

Table 5.4.18

DOSE RATES FOR THE HI-TRAC VW VERSION V2 FOR THE FULLY FLOODED MPC AND FLOODED ANNULUS WITH NEUTRON SHIELD CYLINDER PRESENT, BASED ON FIGURE 1.2.7
MPC-89 DESIGN BASIS ZIRCALOY CLAD FUEL

Dose Point Location	Fuel Gammas (mrem/hr)	(n,γ) Gammas (mrem/hr)	^{60}Co Gammas (mrem/hr)	Neutrons (mrem/hr)	Totals (mrem/hr)
ADJACENT TO THE HI-TRAC VW VERSION V2					
1	169.6	<1	2058.4	2.0	2230.0
1*	22.0	<1	255.1	1.3	278.4
2	678.2	<1	<1	2.7	681.0
3	<1	<1	41.4	<1	41.7
4	4.3	<1	253.4	<1	257.8
5 (bottom lid)	91.1	<1	1485.9	8.8	1585.9
ONE METER FROM THE HI-TRAC VW VERSION V2					
1	165.6	<1	21.2	<1	187.2
1*	164.8	<1	16.1	<1	181.4
2	320.9	<1	1.0	<1	322.9
3	15.4	<1	10.8	<1	26.3
4	3.1	<1	210.5	<1	213.6
5	66.2	<1	1060.2	3.4	1129.8

Notes:
* Location 1* uses a steel shield ring pedestal for the Neutron Shield Cylinder, which may be present for ALARA purposes. The critical shielding dimensions of the optional steel shield ring pedestal are as follows: Outer Diameter is 8 feet; radial thickness is 2.5 inches; Axial bottom of shield ring is 3 inches below MPC baseplate bottom surface; top of shield ring is in contact with Neutron Shield Cylinder.
- Refer to Figure 5.1.2 for dose point locations.
- Values are rounded to nearest integer.
- MPC internal water level is 5 inches below the MPC lid.
- Annulus water level is 2 inches above bottom surface of the MPC lid.
- The "Fuel Gammas" category includes gammas from the spent fuel and ^{60}Co from the spacer grids.

Table 5.4.19

Table Deleted

Table 5.4.20

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5.6 REFERENCES

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APPENDIX 5.A

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

[Proprietary Appendix Withheld in Accordance with 10 CCFR 2.390]

Code, Section III, Subsection NB, Article NB-5350 acceptance criteria. Any evidence of cracking or deformation shall be cause for rejection, or repair and retest, as applicable.

If a leak is discovered, the test pressure shall be reduced, the MPC cavity water level lowered, if applicable, the MPC cavity vented, and the weld shall be examined to determine the cause of the leakage and/or cracking. Repairs to the weld shall be performed in accordance with written and approved procedures prepared in accordance with the ASME Code, Section III, Article NB-4450.

The MPC confinement boundary pressure test shall be repeated until all required examinations are found to be acceptable. Test results shall be documented and maintained as part of the loaded MPC quality documentation package.

10.1.3   Materials Testing

The majority of materials used in the HI-TRAC transfer cask and a portion of the material in the HI-STORM overpack are ferritic steels. ASME Code, Section II and Section III require that certain materials be tested in order to assure that these materials are not subject to brittle fracture failures. Certain versions of the HI-TRAC include Holtite neutron shielding material.

Materials of the HI-TRAC transfer cask and HI-STORM overpack, as required, shall be Charpy V-notch tested in accordance with ASME Section IIA and/or ASME Section III, Subsection NF, Articles NF-2300, and NF-2430. The materials to be tested are identified in Table 3.1.9 and applicable weld materials. Table 3.1.9 provides the test temperatures and test acceptance criteria to be used when performing the material testing specified above.

For Holtite neutron shielding material, each manufactured lot of material shall be tested to verify the material composition (aluminum and hydrogen), boron concentration, and neutron shield density (or specific gravity) meet the requirements. A manufactured lot is defined as the total amount of material used to make any number of mixed batches comprised of constituent ingredients from the same lot/batch identification numbers supplied by the constituent manufacturer. Testing shall be performed in accordance with written and approved procedures and/or standards. Material composition, boron concentration, and density (or specific gravity) data for each manufactured lot of neutron shield material shall become part of the quality documentation package. The procedures shall ensure that mix ratios and mixing methods are controlled in order to achieve proper material composition, boron concentration and distribution, and that pours are controlled in order to prevent gaps from occurring in the material. Samples of each manufactured lot of neutron shield material shall be maintained by Holtec International as part of the quality record documentation package.

The concrete utilized in the construction of the HI-STORM overpack shall be mixed, poured, and tested as set down in Chapter 1.D of the HI-STORM 100 FSAR (Docket 72-1014) [10.1.6] in accordance with written and approved procedures. Testing shall verify the compressive strength and density meet design requirements. Tests required shall be performed at a frequency as defined in the applicable ACI code.