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 according to the alternative storage patterns shown in Figures 1.2.3 through 1.2.5 (MPC-37) and 1.2.6 through 1.2.7 (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

Minimum cooling time must also meet limits specified in Tables 2.1.1a and 2.1.1b. If the calculated *Ct* is less than the cooling time limits in Tables 2.1.1a or 2.1.1b, the minimum cooling time in table is used.

For MPC-37 and MPC-89, the coefficients for above equation for the assembly in an individual cell depend on the heat load limit in that cell, Table 2.1.10 lists the coefficients for several heat load limit ranges. Note that the heat load limits are only used for the lookup of the coefficients in that table, and do not imply any equivalency. Specifically, meeting heat load limits is not a substitute for meeting burnup and cooling time limits, and vice versa.

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 2.1.1b.

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.

<u>Method A: Burnup Verification Through Quantitative Burnup Measurement</u>

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.

<u>Method B: Burnup Verification Through an Administrative Procedure and Qualitative Measurements</u>

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:

Table 2.1.1a

MATERIAL TO BE STORED

PARAMETER	**VALUE**	
	MPC-37	MPC-89
Fuel Type	Uranium oxide undamaged fuel assemblies, damaged fuel assemblies, and fuel debris meeting the limits in Table 2.1.2 for the applicable array/class.	Uranium oxide undamaged fuel assemblies, damaged fuel assemblies, with or without channels, fuel debris meeting the limits in Table 2.1.3 for the applicable array/class.
Cladding Type	ZR (see Glossary for definition)	ZR (see Glossary for definition)
Maximum Initial Rod Enrichment	Depending on soluble boron levels or burnup credit and assembly array/class as specified in Table 2.1.6 and Table 2.1.7.	≤ 5.0 wt. % U-235
Post-irradiation cooling time and average burnup per assembly	Minimum Cooling Time: 1 years and meeting the equation in Subsection 2.1.6 Maximum Assembly Average Burnup: 68.2 GWd/mtU	Minimum Cooling Time: 1.2 years and meeting the equation in Subsection 2.1.6 Maximum Assembly Average Burnup: 65 GWd/mtU

Table 2.1.1a

MATERIAL TO BE STORED

PARAMETER	**VALUE**	
	MPC-37	MPC-89
Non-fuel hardware post-irradiation cooling time and burnup	Minimum Cooling Time: - BPRAs, WABAs, TPDs, water displacement guide tube plugs, orifice rod assemblies and vibration suppressors: 1 year - NSAs, APSRs, RCCAs, CRAs and CEAs: 2 years - ITTRs: not applicable Maximum Burnup†: - BPRAs, WABAs and vibration suppressors: 60 GWd/mtU - TPDs, water displacement guide tube plugs and orifice rod assemblies: 225 GWd/mtU - NSAs, APSRs, RCCAs, CRAs and CEAs: 630 GWd/mtU - ITTRs: not applicable	N/A
Decay heat per fuel storage location	Regionalized Loading: See Tables 1.2.3a and 1.2.3d	Regionalized Loading: See Tables 1.2.4a and 1.2.4b.

† Burnups for non-fuel hardware are to be determined based on the burnup and uranium mass of the fuel assemblies in which the component was inserted during reactor operation. Burnup not applicable for ITTRs since installed post-irradiation.

Table 2.1.1b

MATERIAL TO BE STORED

PARAMETER	**VALUE**
	MPC-32ML
Fuel Type	Uranium oxide undamaged fuel assemblies, damaged fuel assemblies, and fuel debris meeting the limits in Table 2.1.2 for the 16x16D array/class only.
Cladding Type	ZR (see Glossary for definition)
Maximum Initial Rod Enrichment	Depending on soluble boron levels and assembly array/class as specified in Table 2.1.6.
Post-irradiation cooling time and average burnup per assembly	Minimum Cooling Time: 3 years and meeting the equation in Subsection 2.1.6 Maximum Assembly Average Burnup: 68.2 GWd/mtU
Non-fuel hardware post-irradiation cooling time and burnup†	Minimum Cooling Time: 3 years Maximum Burnup: - BPRAs, WABAs and vibration suppressors: 60 GWd/mtU - TPDs, NSAs, APSRs, RCCAs, CRAs, CEAs, water displacement guide tube plugs and orifice rod assemblies: 630 GWd/mtU - ITTRs: not applicable
Decay heat per fuel storage location	Uniform Loading per Table 1.2.3b.
Fuel Assembly Nominal Length (in)	≤ 196.122 (including NFH and DFC)
Fuel Assembly Width (in)	≤ 9.04 (nominal design)
Fuel Assembly Weight (lb)	≤ 1860 (without NFH) ≤ 2120 (with NFH) ≤ 2200 (including DFC and NFH).

† Burnups for non-fuel hardware are to be determined based on the burnup and uranium mass of the fuel assemblies in which the component was inserted during reactor operation. Burnup not applicable for ITTRs since installed post-irradiation.

TABLE 2.1.9
BURNUP AND COOLING TIME FUEL QUALIFICATION REQUIREMENTS
FOR MPC-32ML

Polynomial Coefficients, see Paragraph 2.1.6.1			
A	**B**	**C**	**D**
6.7667E-14	-3.6726E-09	8.1319E-05	2.7951E+00

TABLE 2.1.10
BURNUP AND COOLING TIME FUEL QUALIFICATION REQUIREMENTS
FOR MPC-37 AND MPC-89

Cell Decay Heat Load Limit (kW)	Polynomial Coefficients, see Paragraph 2.1.6.1			
	A	B	C	D (Note 1)
MPC-37				
≤ 0.85	1.68353E-13	-9.65193E-09	2.69692E-04	2.95915E-01
0.85 < decay heat ≤ 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
0.32 < decay heat ≤ 0.5	3.97779E-14	-2.80193E-09	1.36784E-04	3.04895E-01
0.5 < decay heat ≤ 0.75	1.44353E-14	-1.21525E-09	8.14851E-05	3.31914E-01
0.75 < decay heat ≤ 1.1	-7.45921E-15	1.09091E-09	-1.14219E-05	9.76224E-01
1.1 < decay heat ≤ 1.45	3.10800E-15	-7.92541E-11	1.56566E-05	6.47040E-01
1.45 < decay heat ≤ 1.6	-8.08081E-15	1.23810E-09	-3.48196E-05	1.11818E+00

Notes:

1. For BLEU fuel, coefficient D is increased by 1.

Table 2.2.3
TEMPERATURE LIMITS

HI-STORM FW Component	Normal Condition and Design Temperature Limits (°F)	Short Term Events[††] Temperature Limits (°F)	Off-Normal and Accident Condition Temperature Limits [†] (°F)
MPC shell	600*	800*	800*
MPC basket	752	932	932
MPC basket shims	752	932	932
MPC lid	600*	800*	800*
MPC closure ring	500*	800*	800*
MPC baseplate	400*	800*	800*
HI-TRAC VW inner shell	-	600	700
HI-TRAC VW outer shell	-	500	700
HI-TRAC VW bottom lid	-	500	700
HI-TRAC VW water jacket shell	-	500	700**
HI-TRAC VW top flange	-	500	650
HI-TRAC VW bottom lid seals	-	400	N/A
HI-TRAC VW bottom lid bolts	-	400	800
HI-TRAC VW bottom flange	-	400	700
HI-TRAC VW radial neutron shield	-	311	N/A
HI-TRAC VW radial lead gamma shield	-	600	600
HI-TRAC VW Version V2 NSC steel	-	400	600

[††] Short term operations include, but are not limited to, MPC drying and onsite transport. The 1058°F temperature limit applies to MPCs containing all moderate burnup fuel. The limit for MPCs containing one or more high burnup fuel assemblies is 752°F.

[*] Temperature limits in Table 1.A.6 shall take precedence if duplex stainless steels are used for the fabrication of confinement boundary components, as described in Appendix 1.A.

[**] For fire accidents, the steel structure is required to remain physically stable similar to HI-STORM overpack.

[†] For accident conditions that involve heating of the steel structures and no mechanical loading (such as the blocked air duct accident), the permissible metal temperature of the steel parts is defined by Table 1A of ASME Section II (Part D) for Section III, Class 3 materials as 700°F. For the fire event, the structure is required to remain physically stable (no specific temperature limits apply)

Notes: 1.The normal condition temperature limits are used in the design basis structural evaluations for MPC and HI-STORM. The short-term condition temperature limits are used in the design basis structural evaluations for HI-TRAC. All other short-term, off-normal, and accident condition structural evaluations are based on bounding temperatures from thermal evaluations presented in Chapter 4.
2. The temperature limits provided for HI-TRAC VW are applicable to Version V and V2 unless otherwise specified.

Table 2.2.3
TEMPERATURE LIMITS

HI-STORM FW Component	Normal Condition and Design Temperature Limits (°F)	Short Term Events†† Temperature Limits (°F)	Off-Normal and Accident Condition Temperature Limits † (°F)
HI-TRAC VW Version V2 NSC Holtite-A	-	300	350
Fuel Cladding	752 (Storage)	752 or 1058 (Short Term Operations)††	1058 (Off-Normal and Accident Conditions)
Overpack concrete	300 (see HI-STORM 100 FSAR Appendix 1.D)	300	572
Overpack Lid Top and Bottom Plate	450	450	572
Remainder of overpack steel structure	350	350	700
Damaged Fuel Isolator	752	932	932

Table 3.2.2

LIMITING PARAMETERS

	Item	PWR	BWR
1.	Minimum fuel assembly length, inch	150	171
2.	Maximum fuel assembly length, inch	199.2	181.5[3]
3a.	Minimum thickness of the lead cylinder in the lowest weight HI-TRAC VW (standard and Version V), inch	2.75 (MPC-37)	2.50
3b.	Minimum thickness of the lead in the HI-TRAC VW (standard and Version V), inch	3.25 (MPC-32ML)	-
4.	Maximum thickness of the lead cylinder (all HI-TRAC VW versions), inch	4.25	4.25
5.	Nominal (radial) thickness of the water in the external jacket of HI-TRAC VW (standard and Version V), inch	4.75	4.75
6.	Minimum thickness of the lead cylinder in HI-TRAC VW Version V2, inch	2.625	2.625
7.	Minimum thickness of the Holtite cylinder in HI-TRAC VW Version V2, inch	4	4

[3] Maximum fuel assembly length for the BWR fuel assembly refers to the maximum fuel assembly length plus an additional 5” to account for a Damage Fuel Container (DFC).

If the peak cladding temperature cannot be maintained below the ISG-11, Revision 3 limit under a vacuum condition of infinite duration, cycles of vacuum drying resulting in heatup followed with cooling by helium are performed until drying criteria specified in Chapter 9 is achieved. The thermal model described in this section must be used to compute permissible time duration available to perform the heatup/cooldown cycles for a cask-specific decay heat loads. It must also be ensured per ISG-11 Rev 3 that the repeated thermal cycling is limited to less than 10 cycles, with cladding temperature variations less than 65^{o}C (117^{o}F) each. It must be noted that the permissible time for heatup/cooldown cycles is a function of cask specific heat loads. At lower heat loads the duration of vacuum drying cycles is increased and if the heat load is low enough, then the peak cladding temperature may remain below the ISG-11 limit under vacuum conditions indefinitely eliminating the need for cycling.

Following is a summary of the methodology and assumptions for multiple vacuum drying cycles:

i. The initial condition of the cask for vacuum drying is conservatively assumed to be at boiling temperature of water i.e. 100^{o}C (212^{o}F).

ii. Cask specific heat load and ambient conditions must be used.

iii. The cask bottom is assumed to be insulated.

iv. Cycle 1 (Heatup) – A transient thermal evaluation is performed for the cask-specific decay heat distribution with the cask cavity under vacuum condition. The time required for the fuel to heatup from an initial temperature of 100^{o}C (212^{o}F) to 380^{o}C (716^{o}F) for High Burnup Fuel (HBF) and 550^{o}C (1022^{o}F) for Moderate Burnup Fuel (MBF) is determined. If drying completion criteria is not met, then the cask cavity must be backfilled with helium for cooldown before it reaches the ISG-11 Rev 3 temperature limit of 400^{o}C (752^{o}F) for HBF or 570^{o}C (1058^{o}F) for MBF.

v. Cycle 1 (Cooldown) – The cask cavity is backfilled with helium to 1 atm absolute pressure. Fuel cooling under helium is evaluated until the fuel temperature decreases by 65^{o}C (117^{o}F) and the maximum permissible time is obtained from the transient evaluation.

vi. The drying process should return to vacuum drying again which now is the beginning of second cycle. Up to 9 additional multiple cycles of drying may be performed until the drying completion criteria is met.

vii. If a total of 10 drying cycles fail to meet drying criteria then other competent means to dry fuel (like FHD discussed in Subsection 4.5.4.2) must be used or the cask must be de-fueled.

As an example of cyclic drying, 3D thermal evaluation is performed using the methodology described above for a bounding condition of short MPC-37 with High Burnup Fuel under heat

load pattern A. The permissible time for each heatup and cooldown cycles are presented in Table 4.5.25. The variation of peak fuel cladding temperature under vacuum and helium conditions is graphed in Figure 4.5.1.

4.5.3 Maximum Time Limit During Wet Transfer Operations

In accordance with NUREG-1536, water inside the MPC cavity during wet transfer operations is not permitted to boil. This requirement is met by imposing time limits for fuel to remain submerged in water after a loaded HI-TRAC VW cask is removed from the pool.

Fuel loading operations are typically conducted with the HI-TRAC VW and its contents (water filled MPC) submerged in pool water. Under these conditions, the HI-TRAC VW is essentially at the pool water temperature. When the HI-TRAC VW transfer cask and the loaded MPC under water-flooded conditions is removed from the pool, the water, fuel, MPC and HI-TRAC VW metal absorb the decay heat emitted by the fuel assemblies. This results in a slow temperature rise of the HI-TRAC VW with time, starting from an initial (pool water) temperature. The rate of temperature rise is limited by the thermal inertia of the HI-TRAC VW system. To enable a bounding heat-up rate determination, the following conservative assumptions are utilized:

i. Heat loss by natural convection and radiation from the exposed HI-TRAC VW surfaces to ambient air is neglected (i.e., an adiabatic heat-up calculation is performed).

ii. Design maximum heat input from the loaded fuel assemblies is assumed.

iii. The shortest allowable HI-TRAC VW is credited in the analysis to impart the lowest thermal inertia on the system, which will result in the highest rate of temperature rise.

iv. The water mass is in the MPC cavity is understated.

Table 4.5.3 summarizes the weights and thermal inertias of several components in the loaded HI-TRAC VW transfer cask that corresponds to the shortest allowable fuel assembly. The rate of temperature rise of the HI-TRAC VW transfer cask and contents during an adiabatic heat-up is governed by the following equation:

$$\frac{dT}{dt} = \frac{Q}{C_h}$$

where:

Q = conservatively bounding heat load (Btu/hr)
C_h = thermal inertia of a loaded HI-TRAC VW (Btu/°F)
T = temperature of the HI-TRAC VW transfer cask (°F)

Table 4.5.25 PERMISSIBLE TIME FOR MULTIPLE VACUUM DRYING CYCLES FOR MPC-37	
Cycle	**Time (hours)**
Cycle 1 – Heat-Up (Vacuum Drying)	14.1
Cycle 1 – Cooldown (Helium)	7.75
Cycle 2 – Heat-Up (Vacuum Drying)	5.1
Note: The temperature versus time behavior in cycle 1 cooldown and cycle 2 heatup repeats itself in subsequent cycles.	

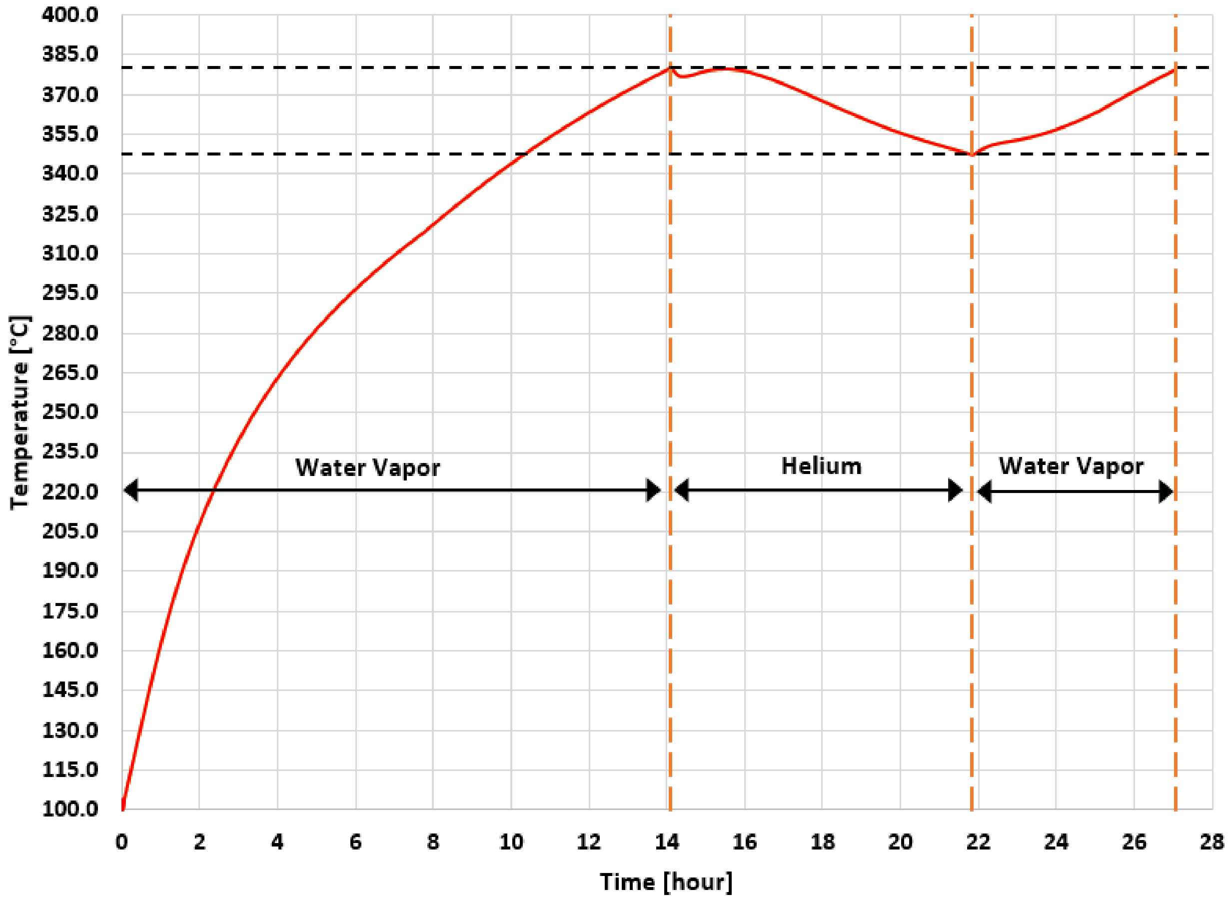

Figure 4.5.1: Variation of PCT with Time during Cyclic Vacuum Drying of MPC-37 with Short Fuel under Heat Load Pattern A

to block air flow through the bottom ducts, the lower region of the MPC will be submerged in the water. Although heat transport through air circulation is cut off in this scenario, the reduction is substantially offset by flood water cooling.

The MPCs are equipped with the thermosiphon capability, which brings the heat emitted by the fuel to the bottom region of the MPC as the circulating helium flows along the downcomer space around the basket. This places the heated helium in close thermal communication with the flood water, further enhancing convective cooling via the flood water.

The most adverse flood condition exists when the flood waters are high enough to block the inlet ducts but no higher. In this scenario, the MPC surface has minimum submergence in water and the ventilation air is completely blocked. In fact, as the flood water begins to accumulate on the ISFSI pad, the air passage size in the inlet vents is progressively reduced. Therefore, the rate of floodwater rise with time is necessary to analyze the thermal-hydraulic problem. For the reference design basis flood (DBF) analysis in this FSAR, the flood waters are assumed to rise instantaneously to the height to block the inlet vents and stay at that elevation for 32 hours. The consequences of the DBF event is bounded by the 100% blocked ducts events evaluated in Section 4.6.2.4. If the duration of the flood blockage exceeds the DBF blockage duration then a site specific evaluation shall be performed in accordance with the methodology presented in this Chapter and evaluated for compliance with Subsection 2.2.3 criteria.

4.6.2.7 100% Air Inlet Blockage of HI-TRAC VW Versions V and V2

As illustrated in the Licensing drawings of HI-TRAC VW Versions V and V2 listed in Section 1.5, the inlet flow passages in HI-TRAC ("the Cask") are not discrete vents; rather they are radially symmetric passages. The outlet is essentially an unhindered annular opening to ambient air above the cask as the design does not require a cover or lid that would otherwise restrict air exit. The ventilation action through the MPC/Cask annulus is entirely by natural convection. It is not credible to postulate that these circumferentially extant passages can be entirely blocked during the transport of the cask from the Fuel Building to the ISFSI pad. However, as a study to support a defense-in-depth approach, an evaluation is presented below assuming that inlet flow passages are 100% blocked. As a result of the considerable inertia of the storage overpack, a significant temperature rise is possible if the inlets are substantially blocked for extended durations. This accident condition is, however, a short duration event that is identified and corrected through scheduled periodic surveillance.

The FLUENT thermal model described in Section 4.5 is adopted to evaluate the temperature rise of the components of the HI-TRAC VW Versions V and V2 casks, by completely blocking the air flow at the inlet vents. A steady state evaluation of Version V is performed, and results are presented in Table 4.6.8 of the FSAR.

However, a steady state thermal evaluation of blockage of Version V2 inlet vents at design basis maximum heat loads result in component temperatures (like Holtite in NSC) exceeding its temperature limit. Therefore, a transient evaluation is instead performed such that all component

temperatures and MPC cavity pressure remain below their accident limits. The computed allowable time is presented in Table 4.6.9 and the associated temperature results are presented in Table 4.6.10.

The results of the CFD analysis of both versions demonstrate that all component temperatures are below their respective accident temperature limits. The MPC cavity pressures presented in Table 4.6.7 are also below the accident design pressure.

Table 4.6.7 OFF-NORMAL AND ACCIDENT CONDITION MAXIMUM MPC PRESSURES	
Condition	**Pressure (psig)**
Off-Normal Conditions	
Off-Normal Pressure[8]	110.0
Partial Blockage of Inlet Ducts	99.9
Accident Conditions	
HI-TRAC VW fire accident	103.3
Extreme Ambient Temperature	101.7
100% Blockage of Air Inlets	116.4
Burial Under Debris	130.8
HI-TRAC VW Jacket Water Loss	109.5
100% Blockage of HI-TRAC VW Version V2 Air Inlets	105.4
100% Blockage of HI-TRAC VW Version V Air Inlets	104.4

8 The off-normal pressure event defined in Section 4.6.1.1 bounds the off-normal ambient temperature event (Section 4.6.1.2)

Table 4.6.8 STEADY STATE TEMPERATURE RESULTS UNDER POSTULATED 100% BLOCKAGE OF HI-TRAC VW VERSION V INLET VENTS	
Component	**Maximum Temperature ºC (ºF)**
Fuel Cladding	410 (771)
MPC Basket	394 (741)
Basket Periphery	320 (608)
Aluminum Basket Shims	293 (560)
MPC Shell	266 (510)
MPC Lid Note 1	254 (489)
HI-TRAC Inner Shell	128 (263)
HI-TRAC Top Flange	124 (256)
HI-TRAC Bottom Flange	118 (245)
HI-TRAC Radial Lead Gamma Shield	127 (261)
Water Jacket Shell	120 (247)
Water Bulk Temperature in Water Jacket	117 (243)
Note 1: Maximum section average temperature is reported.	

Table 4.6.9 ALLOWABLE TIME UNDER 100% BLOCKAGE OF HI-TRAC VW VERSION V AND V2 INLET VENTS	
Design Type	Duration, hours
HI-TRAC VW Version V2	16
HI-TRAC VW Version V	72

Table 4.6.10
TEMPERATURE RESULTS UNDER POSTULATED 100% BLOCKAGE OF HI-TRAC VW VERSION V2 INLET VENTS AFTER 16 HOURS

Component	**Maximum Temperature °C (°F)**
Fuel Cladding	412 (773)
MPC Basket	397 (746)
Basket Periphery	327 (621)
Aluminum Basket Shims	313 (595)
MPC Shell	283 (541)
MPC Lid [Note1]	241 (466)
MPC Base Plate [Note1]	216 (421)
HI-TRAC V2 Inner Shell	203 (397)
HI-TRAC V2 Top Flange	135 (276)
HI-TRAC V2 Bottom Flange	176 (348)
HI-TRAC V2 Bottom Lid	215 (419)
HI-TRAC V2 Radial Lead Gamma Shield	202 (396)
NSC Inner Shell	163 (325)
NSC Top Flange	115 (239)
NSC Bottom Flange	112 (233)
NSC Outer Shell	115 (239)
NSC Holtite	161 (321)
Note 1: Maximum section average temperature is reported	

each device occupies the same location within a fuel assembly, a single PWR fuel assembly will not contain multiple devices, with the exception of instrument tube tie rods (ITTRs), which may be stored in the assembly along with other types of non-fuel hardware.

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

In order to offer the user more flexibility in fuel storage, the HI-STORM FW System offers several heat load patterns, each with two or more regions with different heat load limits. This is taken into consideration when calculating dose rates in this chapter. The regionalized storage patterns are guided by the considerations of minimizing occupational and site boundary dose to comply with ALARA principles.

Two different lids have been developed for the HI-STORM FW concrete overpack. The lid included in the initial application, referred to as "standard lid", and a revised design with overall improved shielding performance, referred to as "XL lid". Since by now essentially all installations utilize the "XL lid", all dose rates provided for MPC-37 and MPC-89 in this chapter are for that lid design, with the only exception being some tables in Section 5.4 which contain selected results for the "standard lid" from previous versions of this chapter for reference. The shielding analysis of HI-STORM FW with MPC-32ML is performed using a "standard lid" design. All references to the "lid" are to be understood to refer to the "XL lid", unless otherwise noted.

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

<u>Acceptance Criteria</u>

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

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

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 and shown in Figure 1.2.3 through Figure 1.2.5 for MPC-37 and Figures 1.2.6 and 1.2.7 for MPC-89. 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.

To prescribe radiological limits for the fuel to be loaded, loading curves are defined in Tables 2.1.9 and 2.1.10, where a loading curve specifies the minimum cooling time as a function of fuel burnup. Different loading curves are defined for the different heat load limits, so that the thermal and radiological requirements for the fuel in each cell are approximately aligned. However, it should be noted that thermal and radiological limits for each assembly are applied completely independent from each other. The uniform and regionalized loading curves for the fuel to be loaded in the MPC-37, MPC-32ML or MPC-89 canisters are discussed in Subsection 5.2.7.

[†] As it is discussed in Subsection 5.1.2, a site-specific shielding evaluation may be required for accident-condition of MPC-32ML.

To determine dose rates consistent with both the uniform and regionalized thermal loading, it is necessary to consider the ranges of burnup and cooling times from all loading curves. For that, 8 burnup values between 5 and 70 GWd/mtU are selected, and corresponding minimum required cooling times are established and used in the dose analyzes. The heat load patterns in Figures 1.2.3 through 1.2.7 contain from 5 to 20 regions each, i.e. from 5 to 20 principal locations with different heat load limit. Applying 8 burnup and cooling time combinations to each location would result up to 8^{20} = 1.15E+18 different burnup and cooling time loading arrangements per pattern. Analyzing and comparing those many arrangements would be excessive. Therefore, for the radiological evaluations, some regions and loading patterns (MPC-37) are combined using the highest heat load limit (source term) of each group. For MPC-37, the heat loads for each cell are based on the “Long” fuel heat loads in Figure 1.2.5a. The established bounding heat load limits are provided in Tables 5.0.3 and 5.0.4.

This then results in effectively only 2 or 5 regions to be independently varied for the considered bounding MPC-37 and MPC-89 patterns, and hence 8^2 = 64 or 8^5 = 32,768 different burnup and cooling time arrangements per pattern is to be analyzed, which is manageable. The selected burnup, enrichment and cooling time combinations for the uniform and regionalized loading patterns are listed in Tables 5.0.3, 5.0.4a, 5.0.4b and 5.0.5. The dose rates in the various important locations are calculated for each of these combination arrangements and the maximum is determined for each dose rate location. It should be noted that this maximum can be from a different loading arrangement in different locations.

Based on this approach, the source terms used in the analyses of MPC-37, MPC-32ML 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 activities are considered for MPC-37 and MPC-32ML in this chapter, unless noted otherwise.

All dose rates in Section 5.1 are developed using the approach discussed above. 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 (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	

NOTE:

To simplify the dose analyses in Chapter 5 that show bounding conditions, for some cells, burnup and cooling time combinations are selected for the dose analyses that may correspond to a higher decay heat than is permitted for that cell. The decay heat limits and burnup/cooling time limits remain independent of each other, so this does not impact the decay heat limit for a cell. The cell decay heat limits are given in Figures 1.2.3 through 1.2.5.

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 (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	

NOTE:

To simplify the dose analyses in Chapter 5 that show bounding conditions, for some cells, burnup and cooling time combinations are selected for the dose analyses that may correspond to a higher decay heat than is permitted for that cell. The decay heat limits and burnup/cooling time limits remain independent of each other, so this does not impact the decay heat limit for a cell. The cell decay heat limits are given in Figures 1.2.6.

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 (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	

NOTE:

To simplify the dose analyses in Chapter 5 that show bounding conditions, for some cells, burnup and cooling time combinations are selected for the dose analyses that may correspond to a higher decay heat than is permitted for that cell. The decay heat limits and burnup/cooling time limits remain independent of each other, so this does not impact the decay heat limit for a cell. The cell decay heat limits are given in Figures 1.2.7.

Table 5.0.5

SELECTED BURNUP, ENRICHMENT, COOLING TIME COMBINATIONS FOR MPC-32ML LOADING PATTERNS FOR NORMAL CONDITIONS

Burnup (MWD/MTU)	**Initial U-235 Enrichment (wt%)**	**Cooling Time (years)**	
		Calculated Using Combination Curve in Table 2.1.9	**Used in Shielding Analysis**
15000	1.1	3.42	3
20000	1.1	3.49	3
25000	1.6	3.59	3.5
30000	2	3.76	3.6
35000	2.4	4.04	4
40000	2.6	4.50	4.5
45000	3	5.18	5
50000	3.3	6.14	6
55000	3.6	7.42	7
60000	3.6	9.07	9
65000	3.9	11.15	11
70000	4.2	13.70	13

5.1.1 Normal and Off-Normal Operations

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

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

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

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

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

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

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

Figure 5.1.2 identifies the location of the dose points for the HI-TRAC VW transfer cask. Dose Point Locations #1 and #3 are situated below and above the water jacket, respectively. In the case of the HI-TRAC VW Version V2, Dose Point Locations #1 and #3 are situated below and above the neutron shield, respectively. Dose Point #4 is the dose location on the HI-TRAC VW lid and dose rates below the HI-TRAC VW are estimated with Dose Point #5. Dose Point Location #2 is situated on the side of the cask at the axial mid-height.

The total dose rates presented are presented for two cases: with and without BPRAs. The dose from the BPRAs was conservatively assumed to be the maximum calculated in Subsection 5.4.4.

Tables 5.1.1, 5.1.2 and 5.1.13 provide dose rates adjacent to and one meter from the HI-TRAC VW during normal conditions for the MPC-37, MPC-89 and MPC-32ML. The dose rates listed in Tables 5.1.1, 5.1.2 and 5.1.13 correspond to the normal condition in which the MPC is dry and the HI-TRAC water jacket is filled with water. It should be noted that the minimum lead thickness of HI-TRAC VW with MPC-32ML is more than that of HI-TRAC with MPC-37.

Tables 5.1.10 provides dose rates adjacent to and one meter from the HI-TRAC VW Version V2 during normal conditions for the MPC-89. The dose rates listed in Table 5.1.10 correspond to the normal condition in which the MPC is dry and the Gamma Shield Cylinder and Neutron Shield Cylinder are present.

Tables 5.1.5, 5.1.6 and 5.1.11 provide the design basis dose rates adjacent to the HI-STORM FW overpack during normal conditions for the MPC-37, MPC-89 and MPC-32ML. Tables 5.1.7, 5.1.8 and 5.1.12 provide the design basis dose rates at one meter from the HI-STORM FW overpack containing the MPC-37, MPC-89 and MPC-32ML, respectively.

The dose to any real individual at or beyond the controlled area boundary is required to be below 25 mrem per year. The minimum distance to the controlled area boundary is 100 meters from the ISFSI. Table 5.1.3 presents the annual dose to an individual from a single HI-STORM FW cask and various storage cask arrays, assuming an 8760 hour annual occupancy at the dose point location. The minimum distance required for the corresponding dose is also listed. It is noted that these data are provided for illustrative purposes only. A detailed site-specific evaluation of dose at the controlled area boundary must be performed for each ISFSI in accordance with 10CFR72.212. The site-specific evaluation will consider dose from other portions of the facility and will consider the actual conditions of the fuel being stored (burnup and cooling time).

Figure 5.1.3 is an annual dose versus distance graph for the HI-STORM FW cask array configurations provided in Table 5.1.3. Figure 5.1.4 is an annual dose versus distance graph for the HI-STORM FW cask array configurations provided in Table 5.4.21. These curves, which are based on an 8760 hour occupancy, are provided for illustrative purposes only and will be re-evaluated on a site-specific basis.

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

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

5.1.2 Accident Conditions

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

> Any individual located on or beyond the nearest boundary of the controlled area may not receive from any design basis accident the more limiting of a total effective dose

equivalent of 5 Rem, or the sum of the deep-dose equivalent and the committed dose equivalent to any individual organ or tissue (other than the lens of the eye) of 50 Rem. The lens dose equivalent shall not exceed 15 Rem and the shallow dose equivalent to skin or to any extremity shall not exceed 50 Rem. The minimum distance from the spent fuel or high-level radioactive waste handling and storage facilities to the nearest boundary of the controlled area shall be at least 100 meters.

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

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

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

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

The complete loss of the HI-TRAC neutron shield significantly affects the dose at mid-height (Dose Point #2) adjacent to the HI-TRAC. Loss of the neutron shield has a small effect on the dose at the other dose points. To illustrate the impact of the design basis accident, the dose rates at Dose Point #2 (see Figure 5.1.2) are provided in Table 5.1.4a with MPC-37 and Tables 5.1.4b and 5.1.4c 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	2333.2
2	3577.0	3.4	0.1	6.7	3587.3	4736.8
3	41.2	4.8	351.2	5.4	402.7	860.8
4	134.0	1.4	481.3	249.2	865.9	1604.3
5	710.0	2.9	1781.0	1122.0	3615.9	3897.4
ONE METER FROM THE HI-TRAC VW						
1	851.3	0.6	71.8	1.5	925.2	1163.1
2	1875.0	1.1	6.9	2.8	1885.8	2399.6
3	222.2	1.6	115.9	2.5	342.1	557.8
4	91.3	0.5	295.9	79.3	467.1	878.5
5	668.9	0.8	1145.1	269.0	2083.8	2257.4

Notes:

- Refer to Figure 5.1.2 for dose locations.
- Dose rates are based on no water within the MPC, an empty annulus, and a water jacket full of water. For the majority of the duration that the HI-TRAC bottom lid is installed, the MPC cavity will be flooded with water. The water within the MPC greatly reduces the dose rate.
- 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.
- ^{60}Co activities from BPRA at 1 year cooling are used.

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	23	12	16	20
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)	Totals with BPRAs (mrem/hr)
1 meter from HI-TRAC VW					
2 (Accident Condition)	2587.6	1.1	16.4	1073.7	4580.3
2 (Normal Condition)	1875.0	1.1	6.9	2.8	2399.6
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.
- ^{60}Co activities from BPRA at 1 year cooling are used.

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)	1.6	<0.1	0.2	0.8	2.7

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 5.1.4c

MAXIMUM DOSE RATES FROM HI-TRAC VW WITH MPC-89
FOR ACCIDENT CONDITIONS AT REGIONALIZED LOADING BURNUP AND COOLING TIMES
BASED ON FIGURE 1.2.7

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)	3231.4	3.8	35.9	3440.3	6711.4
2 (Normal Condition)	2629.5	4.9	17.8	11.4	2663.6
100 meters from HI-TRAC VW					
2 (Accident Condition)	1.5	<0.1	0.2	1.7	3.5

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	474.7
2	252.0	<0.1	<0.1	<0.1	252.2	292.1
3 (surface)	17.0	0.3	15.6	3.7	36.6	60.4
3 (overpack edge)	9.4	0.1	8.6	1.1	19.2	31.9
4 (center)	0.6	3.2	1.0	2.8	7.7	11.2
4 (mid)	16.2	0.9	7.7	3.1	27.9	44.2
4 (outer)	0.8	<0.1	0.5	<0.1	1.3	2.1

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.
- ^{60}Co activities from BPRA at 1 year cooling are used.

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	111.7
2	129.1	<0.1	0.5	<0.1	129.7	151.7
3	10.4	<0.1	2.3	<0.1	12.6	18.3
4 (center)	3.2	0.7	3.1	1.4	8.5	13.5

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 grids.
- ^{60}Co activities from BPRA at 1 year cooling are used.

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)	3.5E-03	30	2.52	5	59

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, 5.1.4b and 5.1.4c.
- Regulatory Limit is from 10CFR72.106.

Table 5.1.11

MAXIMUM DOSE RATES ADJACENT TO HI-STORM FW OVERPACK
FOR NORMAL CONDITIONS
MPC-32MLWITH 16X16D FUEL
LOADING PATTERNS (SEE TABLE 5.0.5)

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	201	1	78	1	280	280
2	172	<1	<1	<1	173	173
3 (surface)	16	1	16	2	35	45
3 (overpack edge)	16	<1	37	<1	53	77
4 (center)	< 1	1	<1	<1	2	2
4 (mid)	4	<1	1	<1	5	6
4 (outer)	10	<1	20	<1	30	43

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.
- ^{60}Co activities from BPRA at 3 year cooling are used.

Table 5.1.12

MAXIMUM DOSE RATES AT ONE METER FROM HI-STORM FW OVERPACK
FOR NORMAL CONDITIONS
MPC-32MLWITH 16X16D FUEL
LOADING PATTERNS (SEE TABLE 5.0.5)

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	52	<1	15	<1	66	67
2	91	<1	1	<1	92	93
3	8	<1	7	<1	16	20
4 (center)	1	<1	1	<1	2	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, ^{60}Co from the spacer grids, and ^{60}Co from the BPRAs in the active fuel region.
- ^{60}Co activities from BPRA at 3 year cooling are used.

Table 5.1.13

MAXIMUM DOSE RATES FROM THE HI-TRAC VW FOR NORMAL CONDITIONS
MPC-32MLWITH 16X16D FUEL
LOADING PATTERNS (SEE TABLE 5.0.5)

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	5	4	41	86	136	136
2	1627	14	<1	25	1666	1666
3	67	3	329	3	402	606
4	74	1	364	156	595	858
5	318	1	1887	527	2734	2734
ONE METER FROM THE HI-TRAC VW						
1	190	4	154	8	356	356
2	723	5	7	10	745	746
3	95	1	63	1	160	198
4	236	<1	229	25	490	634
5	168	<1	1028	133	1329	1329

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.
- ^{60}Co activities from BPRA at 3 year cooling are used.

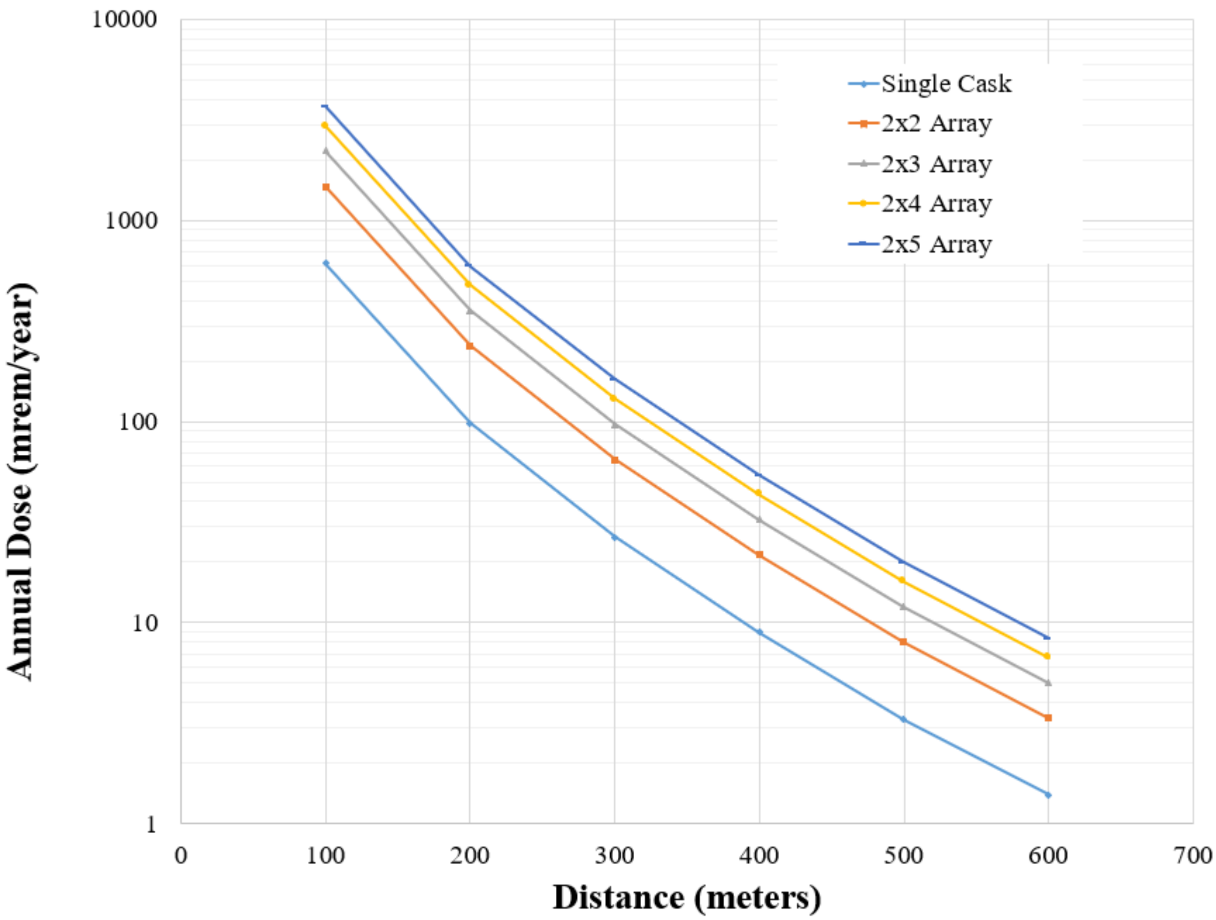

Figure 5.1.4

MAXIMUM ANNUAL DOSE VERSUS DISTANCE FOR VARIOUS CONFIGURATIONS OF THE MPC-32ML FOR BOUNDING UNIFORM PATTERNS (SEE TABLE 5.0.5) (8760-HOUR OCCUPANCY ASSUMED)

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 TRITON, 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 in MPC-37 and MPC-89 for the source term calculations is provided in Table 5.2.1, and in Table 5.2.18 for design basis fuel in MPC-32ML. 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 Tables 5.2.1 and 5.2.18 resulted in conservative source term calculations.

5.2.1 Gamma Source

Tables 5.2.2 through 5.2.5, and Tables 5.2.19 and 5.2.20 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 and Table 5.2.18 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 and Table 5.2.21. 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. Table 5.2.22 provide those data for the assemblies in the MPC-32ML.

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, and Table 5.2.23 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.

BPRAs are made of stainless steel in the region above the active fuel zone and may contain a small amount of inconel in this region. Within the active fuel zone the BPRAs may contain 2-24

rodlets which are burnable absorbers clad in either zircaloy or stainless steel. The stainless steel clad BPRAs create a significant radiation source (Co-60) while the zircaloy clad BPRAs create a negligible radiation source. Therefore, the stainless steel clad BPRAs are bounding.

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

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

Table 5.2.16a 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) at 3 years of cooling. These activities were used for shielding evaluations of MPC-37 and MPC-32ML. In order to qualify non-fuel hardware with the lower cooling time for MPC-37, the radiation source terms for BPRA and TPD with the cooling time of 1 year, independent of the burnup, have been additionally considered. The Co-60 activities that were calculated for BPRAs and TPDs at 1 year of cooling are presented in Table 5.2.16b. A burnup and cooling time, separate from the fuel assemblies, is used for BPRAs and TPDs. Tables 2.1.25 and 2.II.1.2 of the HI-STORM 100 [5.2.17] list the allowable burnups and cooling times for non-fuel hardware that corresponds to the BPRA. These burnup and cooling times assure that the Co-60 activity remains below the levels specified above. For specific site boundary evaluations, these levels/values can be used if they are bounding. Alternatively, more realistic values can be used.

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

It should be noted that 16x16D fuel assemblies may actually not use BPRAs, but the BPRAs with the minimum allowable cooling time are conservatively considered in MPC-32ML shielding calculations to demonstrate compliance with the applicable safety limits.

burnup and cooling time limits are derived based on the allowable decay heat limits.

5.2.6 Fuel Assembly Neutron Sources

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

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

For ease of implementation in the CoC, the restriction concerning the number of NSAs is being applied to all types of NSAs. In addition, conservatively NSAs are required to be stored in the inner region of the MPC basket as specified in Subsection 2.1. Further limitations allow for only one NSA to be stored in the MPC-37 (see Table 2.1.1a), or MPC-32ML (see Table 2.1.1b).

5.2.7 Design Basis Burnup and Cooling Times

For the fuel to be loaded into the HI-STORM FW system, the uniform and regionalized design basis loading curves (which specify burnup and cooling time combinations for each region of the cask) are provided in Tables 2.1.9 and 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 Tables 2.1.9 and 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 for MPC-37 and MPC-89, and burnups from 15 GWD/MTU to 70 GWD/MTU, in increments of 5 GWD/MTU for MPC-32ML;
- The cooling time is calculated for each burnup using the equation and polynomial coefficients in Tables 2.1.9 and 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. For MPC-37 and MPC-89, the value of 1 year (minimum allowed cooling time) is used for all cooling times below 1 year. For MPC-32ML, the value of 3 year (minimum allowed cooling time) is used for all cooling times below 3 years;
- 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 through 5.0.5.

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 130,000 PWR and 185,000 BWR assemblies, the fuel assemblies are distributed over different burnup range bins (0 5, 5-10 … 70-75 GWd/mtU). For instance, for a given burnup group of 5-10 GWd/mtU, the data array includes the enrichments for the fuel assemblies with the burnup from 5,000 MWd/mtU to 9,999 MWd/mtU. Then, in each burnup group, the array of enrichments is sorted from low to high, and the array index that precedes a fraction of 99% of the population is determined. The fuel enrichment under this array position represents the lower bound enrichment that conservatively bounds 99% of the fuel assembly population. 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 5.2.4

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 2.4-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	49.12
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	88.76
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 2.4-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 5.2.14

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 2.4-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 5.2.15 DESCRIPTION OF DESIGN BASIS BURNABLE POISON ROD ASSEMBLY AND THIMBLE PLUG DEVICE		
Region	**BPRA**	**TPD**
Upper End Fitting (kg of steel)	2.62	2.3
Upper End Fitting (kg of inconel)	0.42	0.42
Gas Plenum Spacer (kg of steel)	0.77488	1.71008
Gas Plenum Springs (kg of steel)	0.67512	1.48992
In-core (kg of steel)	13.2	N/A

Table 5.2.16a DESIGN BASIS COBALT-60 ACTIVITIES FOR BURNABLE POISON ROD ASSEMBLIES AND THIMBLE PLUG DEVICES AT 3-YEARS COOLING		
Region	**BPRA**	**TPD**
Upper End Fitting (curies Co-60)	32.7	25.21
Gas Plenum Spacer (curies Co-60)	5.0	9.04
Gas Plenum Springs (curies Co-60)	8.9	15.75
In-core (curies Co-60)	848.4	N/A

Table 5.2.16b DESIGN BASIS COBALT-60 ACTIVITIES FOR BURNABLE POISON ROD ASSEMBLIES AND THIMBLE PLUG DEVICES AT 1-YEAR COOLING		
Region	**BPRA**	**TPD**
Upper End Fitting (curies Co-60)	77	120
Gas Plenum Spacer (curies Co-60)	12	43
Gas Plenum Springs (curies Co-60)	21	75
In-core (curies Co-60)	2010	N/A

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
70,000-75,000	4.5	4.0

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 (MPC-37 and MPC-89) and uniform loading in Table 1.2.3b (MPC-32ML).
2. The burnup ranges do not overlap. Therefore, for MPC-37 and MPC-89, 20,000-25,000 MWD/MTU means 20,000-24,999.9 MWD/MTU, etc. This note does not apply to the maximum burnup of 75,000 MWD/MTU. For MPC-32ML, a lower enrichment value from a preceding burnup range is conservatively used for a transitional burnup, i.e. 20,000-25,000 MWD/MTU means 20,000.1-25,000 MWD/MTU, etc.

Table 5.2.18 DESCRIPTION OF 16X16D DESIGN BASIS CLAD FUEL	
	PWR (MPC-32ML)
Assembly type/class	16x16D
Active fuel length (cm)	390
No. of fuel rods	236
Rod pitch (cm)	1.43
Cladding material	Zircaloy-4
Rod diameter (cm)	1.075
Cladding thickness (cm)	0.068
Pellet diameter (cm)	0.911
Pellet material	UO_2
Pellet density (g/cc)	10.45 (95.3% of theoretical)
Enrichment (w/o ^{235}U)	3.6
Specific power (MW/MTU)	36.56
Weight of UO_2 (kg)††	624.651
Weight of U (kg)††	552.639
No. of Water Rods/ Guide Tubes	20
Water Rod/ Guide Tube O.D. (cm)	1.41
Water Rod/ Guide Tube Thickness (cm)	0.077

†† Derived from parameters in this table.

Table 5.2.18 (continued) DESCRIPTION OF 16X16D DESIGN BASIS FUEL	
	PWR (MPC-32ML)
Lower End Fitting (kg)	10.795 (steel/inconel)
Gas Plenum Springs (kg)	1.474 (steel/inconel)
Gas Plenum Spacer (kg)	1.692 (steel/inconel)
Upper End Fitting (kg)	12.344 (steel/inconel)
Incore Grid Spacers (kg)	12.67 (inconel)

Table 5.2.19

CALCULATED 16X16D (MPC-32ML) PWR FUEL GAMMA SOURCE PER ASSEMBLY FOR SELECTED DESIGN BASIS BURNUP AND COOLING TIME FOR NORMAL CONDITIONS

Lower Energy	**Upper Energy**	**45,000 MWD/MTU 5-Year Cooling**	
(MeV)	(MeV)	(MeV/s)	(Photons/s)
0.45	0.7	2.16E+15	3.76E+15
0.7	1.0	9.10E+14	1.07E+15
1.0	1.5	2.09E+14	1.67E+14
1.5	2.0	1.16E+13	6.62E+12
2.0	2.5	6.87E+12	3.05E+12
2.5	3.0	7.39E+11	2.69E+11
Total		3.30E+15	5.01E+15

Table 5.2.20

CALCULATED 16X16D (MPC-32ML) PWR FUEL GAMMA SOURCE PER ASSEMBLY FOR DESIGN BASIS BURNUP AND COOLING TIME FOR ACCIDENT CONDITIONS

Lower Energy	**Upper Energy**	**62,500 MWD/MTU 8-Year Cooling**	
(MeV)	(MeV)	(MeV/s)	(Photons/s)
0.45	0.7	2.07E+15	3.61E+15
0.7	1.0	5.30E+14	6.23E+14
1.0	1.5	1.67E+14	1.34E+14
1.5	2.0	7.19E+12	4.11E+12
2.0	2.5	6.65E+11	2.95E+11
2.5	3.0	1.00E+11	3.64E+10
Total		2.78E+15	4.37E+15

Table 5.2.21

SCALING FACTORS USED IN CALCULATING THE 16X16D (MPC-32ML) ^{60}Co SOURCE

Region	PWR (MPC-32ML)
Upper End Fitting	0.05
Gas Plenum Spacer	0.1
Gas Plenum Springs	0.2
Incore Grid Spacer	1.0
Lower End Fitting	0.2

Table 5.2.22

CALCULATED ^{60}Co SOURCE PER ASSEMBLY FOR 16X16D (MPC-32ML) AT SELECTED DESIGN BASIS BURNUP AND COOLING TIME COMBINATIONS FOR NORMAL AND ACCIDENT CONDITIONS

Location	**45,000 MWD/MTU and 5-Year Cooling (curies)**	**62,500 MWD/MTU and 8-Year Cooling (curies)**
Upper End Fitting	41.46	29.76
Gas Plenum Springs	11.37	8.16
Gas Plenum Spacer	19.80	14.21
Incore Grid Spacers	851.14	610.89
Lower End Fitting	145.04	104.10

Table 5.2.23

CALCULATED 16X16D (MPC-32ML) PWR NEUTRON SOURCE PER ASSEMBLY AT SELECTED DESIGN BASIS BURNUP AND COOLING TIME COMBINATIONS FOR NORMAL AND ACCIDENT CONDITIONS

Lower Energy (MeV)	**Upper Energy (MeV)**	**45,000 MWD/MTU 5-Year Cooling (Neutrons/s)**	**62,500 MWD/MTU 8-Year Cooling (Neutrons/s)**
1.0e-01	4.0e-01	4.75E+07	7.29E+07
4.0e-01	9.0e-01	1.04E+08	1.59E+08
9.0e-01	1.4	1.04E+08	1.59E+08
1.4	1.85	8.27E+07	1.27E+08
1.85	3.0	1.53E+08	2.36E+08
3.0	6.43	1.40E+08	2.15E+08
6.43	20.0	1.34E+07	2.06E+07
Totals		6.44E+08	9.89E+08

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 (in MPC-37) 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. The width of 16x16D fuel assembly in MCNP model of MPC-32ML is provided as a note under Table 5.3.1.

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.

3. The MPC cavity height, MPC height and HI-STORM FW cavity height for HI-STORM FW with MPC-32ML are calculated using the length of fuel without non-fuel hardware and/or DFC, and data provided in Table 3.2.1.

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

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 used in these regions are shown in Table 5.2.1 and Table 5.2.18. The axial description of the design basis fuel assemblies is provided in Table 5.3.1. Figures 5.3.10 and 5.3.11 graphically depict the location of the PWR and BWR fuel assemblies within the HI-STORM FW system. The axial locations of the basket, inlet vents, and outlet vents are shown in these figures.

5.3.1.2 Streaming Considerations

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

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

5.3.2 Regional Densities

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

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

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

Table 5.3.2 (continued)

COMPOSITION OF THE MATERIALS IN THE HI-STORM FW SYSTEM

Component	Density (g/cm^3)	Elements	Mass Fraction (%)
Water w/ 2000 ppm	0.958	B-10	0.036
		B-11	0.164
		H	11.17
		O	88.63
Concrete	2.4	H	1.0
		O	53.2
		Si	33.7
		Al	3.4
		Na	2.9
		Ca	4.4
		Fe	1.4
Holtite-A **Withheld in Accordance with 10 CFR 2.390**	**Withheld in Accordance with 10 CFR 2.390**		

[1] BWR fuel region mixture (no fuel channel) for dose rates based on the XL Lid Design.
[2] BWR fuel region mixture (fuel channel included) for dose rates based on the Standard Lid Design.

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

where,

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

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

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

Tables 5.1.5, 5.1.6 and 5.1.11 provide the design basis dose rates adjacent to the HI-STORM overpack during normal conditions for the MPC types in Table 1.0.1. Tables 5.1.7, 5.1.8 and 5.1.12 provide the design basis dose rates at one meter from the overpack. A detailed discussion of the normal, off-normal, and accident condition dose rates is provided in Subsections 5.1.1 and 5.1.2.

Table 5.4.2 shows the corresponding dose rates adjacent to and one meter away from the HI-TRAC for the fully flooded MPC-37 condition with an empty water-jacket (condition in which

assume BPRA in every assembly. Note that, even for calculations without NFH, the dose from the active region conservatively contains the contribution of the BPRA. This mainly affects dose location 1 and 2, and results for these locations are therefore identical in most tables, and don't show the dose rate difference indicated in Table 5.4.8.

The analyses in this chapter that consider presence of BPRAs assume that a full-length rod with burnable poison is present in all principal locations. In reality, many BPRAs contain full-length poison rods in some locations, and thimble rodlets in others. The burnup and cooling time combinations listed in Table 2.1.25 of HI-STORM 100 FSAR [5.2.17] for BPRAs and TPDs were selected to ensure the Co-60 activity of those devices is below the value of 895 Ci (BPRA) and 50 Ci (TPD) for the minimum cooling time of 3 years. The burnup and cooling time combinations listed in Table 2.II.1.2 of HI-STORM 100 FSAR [5.2.17] for BPRAs and TPDs were selected to ensure the Co-60 activity of those devices is below the value of 2120 Ci (BPRA) and 238 Ci (TPD) for the minimum cooling time reduced to 1 year. These activities are used in the dose evaluations presented in this chapter. Apart from the total activity, the axial distribution of the material in those devices is important for the dose rates. This axial distribution is shown in Table 5.2.15 (masses) and 5.2.16 (activities). It can be observed from Table 5.2.16, while TPDs have a lower overall activity, their activity in the gas plenum region of the assembly is higher compared to that of the BPRAs. These activities were used to calculate the dose rates in Table 5.4.8. The results in this table show that the maximum dose effect for BPRAs is at the side of the cask, while the maximum dose effect of TPDs is near and on the top of the cask. Nevertheless, Table 5.4.8 demonstrates that even near and on the top of the cask, the TPD doses are bounded by the BPRA doses. It is to be noted that BPRAs with several thimble plugs may result higher dose rate near and on the top of the cask than that reported in Table 5.4.8. However, the potential local increase in dose near and on the top of the cask due the presence of several thimble plug rodlets instead of full length BPRA rods would be more than compensated by the reduction of the dose from the side of the cask at larger distances. Therefore, using BPRAs with all burnable poison rods in the analyses that demonstrate compliance with the site boundary dose limits would be bounding, and hence the burnup and cooling time combinations for BPRAs in Tables 2.1.25 and 2.II.1.2 of the HI-STORM 100 FSAR [5.2.17] are conservative.

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

dose rate out the bottom of the overpack is substantial due to these devices. However, these dose rates occur in an area (below the pool lid and transfer doors) which is not normally occupied.

The effect of TPDs with the lower cooling time of 1 year as well as CRAs and APSRs with the lower cooling time of 2 years, independent of the burnup, was analyzed in the HI-STORM FSAR [5.2.17]. The results in Table 5.II.4.7 of [5.2.17] show that most of the increased Co-60 activities are well bounded by the activity increase of BPRA with 1 year cooling time, which is used in the dose rate calculations. The activity increase for CRAs and APSRs is higher, but the dose rates on the radial surface and at a distance from the overpack due to the storage of these devices is at least 16 times less than the dose rate from BPRAs, and the dose rate out the top of the overpack is essentially 0. Hence the Co-60 activity of BPRA with 1 year cooling time is considered bounding and used in the dose rate calculations in this chapter.

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 MPC-32ML Loading Pattern Dose Rates

The dose rates provided in Tables 5.1.11 and 5.1.12 are the maximum dose rates for HI-STORM FW with MPC-32ML for conservative loading patterns in Table 5.0.5. Table 5.4.19 and Table 5.4.20 provide adjacent and 1-m dose rates for all burnup-enrichment-cooling time combinations from Table 5.0.5.

The distance dose rates for arrays of HI-STORM FWs with MPC-32ML are provided in Table 5.4.21 for the most bounding loading pattern from Table 5.0.5.

The dose rates provided in Table 5.1.13 are the maximum dose rates for HI-TRAC VW with MPC-32ML for conservative loading patterns in Table 5.0.5. Table 5.4.22 provides adjacent and 1-m dose rates for all burnup-enrichment-cooling time combinations from Table 5.0.5.

Higher concrete density may be used in site specific shielding analysis to further lower the occupational dose rates.

5.4.7 Dose Rate Evaluation for Fuel Assemblies with Irradiated Stainless Steel Replacement Rods

Some fuel assemblies may contain irradiated stainless steel rods. 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 replacement rods. The dose rates with the 4 irradiated stainless steel replacement rods in the design basis PWR assembly are approximately 10% higher at the sides and top of the HI-STORM containing the MPC-37. The dose rates with the 2 irradiated stainless steel replacement rods in the design basis BWR assembly are approximately 21% higher at the sides and top of the HI-STORM containing the MPC-89.

Therefore, fuel assemblies containing irradiated stainless steel replacement rods are acceptable for storage and, if present in a fuel assembly, need to be considered in the site specific dose calculations.

5.4.8 Dose Rate Evaluation for BLEU Fuel

From shielding perspective, assemblies containing Blended Low Enriched Uranium (BLEU) fuel material are essentially identical to UO_2 fuel except for the presence of small amount of impurities. A source terms evaluation was performed to determine the impact of impurities in the BLEU fuel material in comparison with the design basis source terms. The results showed that only the increased cobalt impurity content in BLEU fuel can have an impact on the fuel gamma source terms. To compensate for increased gamma source terms due to increased cobalt impurity content, additional cooling time is applied to BLEU fuel (see Table 2.1.10) to maintain the dose rates below the maximum dose rates provided in this chapter.

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	1524.0
2	2627.9	0.7	<0.1	131.4	2760.1	3678.7
3	8.7	<0.1	129.2	2.8	140.7	313.6
4	26.9	<0.1	201.3	0.2	228.5	534.2
5 (bottom lid)	460.4	0.2	1509.1	70.4	2040.1	2189.5
ONE METER FROM THE HI-TRAC VW						
1	525.3	0.1	40.8	28.2	594.5	795.4
2	1170.7	0.3	3.5	60.3	1234.8	1700.3
3	133.8	<0.1	54.7	9.5	198.1	330.3
4	14.7	<0.1	128.0	0.2	142.9	314.9
5	388.8	<0.1	982.9	15.4	1387.2	1473.9

Notes:

- Refer to Figure 5.1.2 for dose point locations.
- 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.
- ^{60}Co activities from BPRA at 1 year cooling are used.

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	932.6
2	1684.2	0.2	<0.1	1.1	1685.4	2218.2
3	3.5	<0.1	69.7	<0.1	73.3	165.7
4	26.9	<0.1	201.3	0.2	228.4	534.1
5 (bottom lid)	459.9	0.2	1508.9	70.2	2039.2	2188.0
ONE METER FROM THE HI-TRAC VW						
1	334.3	<0.1	22.1	0.2	356.6	467.8
2	737.4	<0.1	2.0	0.4	739.9	1002.1
3	85.5	<0.1	23.4	0.2	109.0	182.3
4	14.7	<0.1	128.0	<0.1	142.8	314.8
5	388.4	<0.1	982.9	14.9	1386.3	1472.8

Notes:

- Refer to Figure 5.1.2 for dose point locations.
- 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.
- ^{60}Co activities from BPRA at 1 year cooling are used.

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 at 3 years cooling.
- The “Fuel Gammas” category includes gammas from the spent fuel, ^{60}Co from the spacer grids, and ^{60}Co from the BPRAs in the active fuel region.

Table 5.4.7

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

(REPRESENTATIVE BURNUP AND COOLING TIME COMBINATION)

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

Notes:

- 8760 hour annual occupancy is assumed.
- ^{60}Co activities from BPRA at 3 years cooling are used.

Table 5.4.8 DOSE RATES DUE TO BPRAs AND TPDs FROM THE HI-TRAC VW FOR NORMAL CONDITIONS (3 YEARS OF COOLING)		
Dose Point Location	**BPRAs (mrem/hr)**	**TPDs (mrem/hr)**
ADJACENT TO THE HI-TRAC VW		
1	159.09	0.0
2	509.04	0.0
3	192.78	165.31
4	304.15	275.53
5	137.27	0.0
ONE METER FROM THE HI-TRAC VW		
1	122.06	0.40
2	240.70	3.10
3	128.50	86.95
4	174.25	153.49
5	63.13	0.0

Notes:

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

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.
- ^{60}Co activities from BPRA at 3 years cooling are used.

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.
- ^{60}Co activities from BPRA at 3 years cooling are used.

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.
- ^{60}Co activities from BPRA at 3 years cooling are used.

Table 5.4.19

ADJACENT DOSE RATES FOR HI-STORM FW WITH MPC-32ML WITH 16X16D FUEL BURNUP-COOLING TIME COMBINATIONS (SEE TABLE 5.0.5 FOR LOADING PATTERNS)

Dose Point Location	**Totals + BPRA (mrem/hr)**					
	15,000 MWD/MTU 3-Year Cooling	**20,000 MWD/MTU 3-Year Cooling**	**25,000 MWD/MTU 3.5-Year Cooling**	**30,000 MWD/MTU 3.6-Year Cooling**	**35,000 MWD/MTU 4-Year Cooling**	**40,000 MWD/MTU 4.5-Year Cooling**
1	219	279	260	280	271	262
2	138	173	152	163	151	139
3 (surface)	34	42	41	44	44	45
3 (overpack edge)	66	77	74	77	75	75
4 (center)	1	1	1	1	1	1
4 (mid)	4	5	5	6	6	6
4 (outer)	36	43	41	43	42	42

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.
- ^{60}Co activities from BPRA at 3 years cooling are used.

Table 5.4.19 (continued)

ADJACENT DOSE RATES FOR HI-STORM FW WITH MPC-32ML WITH 16X16D FUEL BURNUP-COOLING TIME COMBINATIONS (SEE TABLE 5.0.5 FOR LOADING PATTERNS)

Dose Point Location	**Totals + BPRA (mrem/hr)**					
	45,000 MWD/MTU 5-Year Cooling	**50,000 MWD/MTU 6-Year Cooling**	**55,000 MWD/MTU 7-Year Cooling**	**60,000 MWD/MTU 9-Year Cooling**	**65,000 MWD/MTU 11-Year Cooling**	**70,000 MWD/MTU 13-Year Cooling**
1	250	219	197	162	136	118
2	129	109	97	80	70	63
3 (surface)	44	41	40	36	33	31
3 (overpack edge)	73	69	65	58	52	47
4 (center)	1	1	2	2	2	2
4 (mid)	6	5	5	5	4	4
4 (outer)	41	38	36	32	28	26

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.
- ^{60}Co activities from BPRA at 3 years cooling are used.

Table 5.4.20

1-METER DOSE RATES FOR HI-STORM FW WITH MPC-32ML WITH 16X16D FUEL BURNUP-COOLING TIME COMBINATIONS (SEE TABLE 5.0.5 FOR LOADING PATTERNS)

Dose Point Location	**Totals + BPRA (mrem/hr)**					
	15,000 MWD/MTU 3-Year Cooling	**20,000 MWD/MTU 3-Year Cooling**	**25,000 MWD/MTU 3.5-Year Cooling**	**30,000 MWD/MTU 3.6-Year Cooling**	**35,000 MWD/MTU 4-Year Cooling**	**40,000 MWD/MTU 4.5-Year Cooling**
1	53	67	60	64	60	57
2	74	93	81	87	81	74
3	17	20	19	20	20	19
4 (center)	2	2	2	2	2	2

Notes:
- Refer to Figure 5.1.1 for dose locations.
- Values are rounded to nearest integer where appropriate.
- ^{60}Co activities from BPRA at 3 years cooling are used.

Table 5.4.20 (continued)

1-METER DOSE RATES FOR HI-STORM FW WITH MPC-32ML WITH 16X16D FUEL BURNUP-COOLING TIME COMBINATIONS (SEE TABLE 5.0.5 FOR LOADING PATTERNS)

Dose Point Location	**Totals + BPRA (mrem/hr)**					
	45,000 MWD/MTU 5-Year Cooling	**50,000 MWD/MTU 6-Year Cooling**	**55,000 MWD/MTU 7-Year Cooling**	**60,000 MWD/MTU 9-Year Cooling**	**65,000 MWD/MTU 11-Year Cooling**	**70,000 MWD/MTU 13-Year Cooling**
1	53	46	41	34	28	25
2	69	58	51	42	36	33
3	19	17	16	14	13	12
4 (center)	2	2	2	2	2	2

Notes:
- Refer to Figure 5.1.1 for dose locations.
- Values are rounded to nearest integer where appropriate.
- ^{60}Co activities from BPRA at 3 years cooling are used.

Table 5.4.21

MAXIMUM DOSE RATES FOR ARRAYS OF HI-STORM FWs WITH MPC-32ML LOADING PATTERNS (SEE TABLE 5.0.5)

Array Configuration	**1 Cask**	**2x2**	**2x3**	**2x4**	**2x5**
HI-STORM FW Overpack					
Annual Dose (mrem/year)	9	22	12	16	20
Distance to Controlled Area Boundary (meters)	400	400	500	500	500

Table 5.4.22
DOSE RATES FOR HI-TRAC VW WITH MPC-32ML WITH SELECTED 16X16D FUEL BURNUP-COOLING TIME COMBINATIONS

Dose Point Location	Totals + BPRA (mrem/hr)					
	15,000 MWD/MTU 3-Year Cooling	**20,000 MWD/MTU 3-Year Cooling**	**25,000 MWD/MTU 3.5-Year Cooling**	**30,000 MWD/MTU 3.6-Year Cooling**	**35,000 MWD/MTU 4-Year Cooling**	**40,000 MWD/MTU 4.5-Year Cooling**
ADJACENT TO THE HI-TRAC VW						
1	61	82	83	92	99	110
2	1306	1666	1518	1654	1597	1553
3	516	598	585	606	602	604
4	641	757	751	795	810	846
5	1850	2389	2354	2539	2591	2719
ONE METER FROM THE HI-TRAC VW						
1	280	355	333	356	348	344
2	587	746	675	732	703	679
3	164	198	187	197	193	191
4	472	574	573	616	624	634
5	957	1223	1197	1278	1289	1329

Notes:

- Refer to Figure 5.1.2 for dose locations.
- Dose rates are based on no water within the MPC, an empty annulus, and a water jacket full of water. For the majority of the duration that the HI-TRAC bottom lid is installed, the MPC cavity will be flooded with water. The water within the MPC greatly reduces the dose rate.
- 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.
- ^{60}Co activities from BPRA at 3 years cooling are used.

Table 5.4.22 (continued)
DOSE RATES FOR HI-TRAC VW WITH MPC-32ML WITH SELECTED 16X16D FUEL BURNUP-COOLING TIME COMBINATIONS

Dose Point Location	**Totals + BPRA (mrem/hr)**					
	45,000 MWD/MTU 5-Year Cooling	**50,000 MWD/MTU 6-Year Cooling**	**55,000 MWD/MTU 7-Year Cooling**	**60,000 MWD/MTU 9-Year Cooling**	**65,000 MWD/MTU 11-Year Cooling**	**70,000 MWD/MTU 13-Year Cooling**
ADJACENT TO THE HI-TRAC VW						
1	117	121	127	136	135	135
2	1504	1361	1292	1185	1105	1058
3	591	559	531	481	433	395
4	858	852	854	852	821	800
5	2734	2649	2590	2470	2259	2103
ONE METER FROM THE HI-TRAC VW						
1	333	303	283	251	221	199
2	653	587	553	501	464	441
3	186	172	163	149	136	127
4	628	590	560	502	453	418
5	1316	1246	1188	1082	952	853

Notes:

- Refer to Figure 5.1.2 for dose locations.
- Dose rates are based on no water within the MPC, an empty annulus, and a water jacket full of water. For the majority of the duration that the HI-TRAC bottom lid is installed, the MPC cavity will be flooded with water. The water within the MPC greatly reduces the dose rate.
- 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.
- ^{60}Co activities from BPRA at 3 years cooling are used.

[5.2.8] O. W. Hermann, et al., "Validation of the Scale System for PWR Spent Fuel Isotopic Composition Analyses," ORNL/TM-12667, Oak Ridge National Laboratory, March 1995.

[5.2.9] M. D. DeHart and O. W. Hermann, "An Extension of the Validation of SCALE (SAS2H) Isotopic Predictions for PWR Spent Fuel," ORNL/TM-13317, Oak Ridge National Laboratory, September 1996.

[5.2.10] O. W. Hermann and M. D. DeHart, "Validation of SCALE (SAS2H) Isotopic Predictions for BWR Spent Fuel," ORNL/TM-13315, Oak Ridge National Laboratory, September 1998.

[5.2.11] "Summary Report of SNF Isotopic Comparisons for the Disposal Criticality Analysis Methodology," B00000000-01717-5705-00077 REV 00, CRWMS M&O, September 1997.

[5.2.12] "Isotopic and Criticality Validation of PWR Actinide-Only Burnup Credit," DOE/RW-0497, U.S. Department of Energy, May 1997.

[5.2.13] B. D. Murphy, "Prediction of the Isotopic Composition of UO_2 Fuel from a BWR: Analysis of the DU1 Sample from the Dodewaard Reactor," ORNL/TM-13687, Oak Ridge National Laboratory, October 1998.

[5.2.14] O. W. Hermann, et al., "Technical Support for a Proposed Decay Heat Guide Using SAS2H/ORIGEN-S Data," NUREG/CR-5625, ORNL-6698, Oak Ridge National Laboratory, September 1994.

[5.2.15] C. E. Sanders, I. C. Gauld, "Isotopic Analysis of High-Burnup PWR Spent Fuel Samples from the Takahama-3 Reactor," NUREG/CR-6798, ORNL/TM-2001/259, Oak Ridge National Laboratory, January 2003.

[5.2.16] Not Used.

[5.2.17] HI-2002444, Proposed Rev. 17A, "Final Safety Analysis Report for the HI-STORM 100 Cask System", USNRC Docket 72-1014, Submittal Letter 5014878, submitted on September 16, 2019.

[5.2.18] Safety Analysis Report on the HI-STAR 190 Package, Holtec International Report HI 2146214, Revision 3, USNRC Docket No 71-9373, Washington, DC.

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

APPENDIX 5.A

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

Proprietary Information Withheld in Accordance with 10 CFR 2.390

calculations. Instead, they are based on the values for the first scenarios, adjusted by suitable adjustment factors, as discussed below.

The factors are established by a comparison of the cask dose rates between the results in Section 5.1 and 5.4. Since most of the operations in the vicinity of the HI-TRAC overpack, such as installation/removal of equipment, welding, inspection, etc., are performed on scaffolding near the top end of the cask and near the MPC lid surface, the dose rate ratio determined near the top of the cask (i.e. cask surface and 1 m distance at the dose rate locations 3 and 4 in Figure 5.1.2 of the FSAR) is representative to determine the occupational dose for the workers near the HI-TRAC cask. Therefore, for the dose rates around the HI-TRAC, results near the top of the cask are used to derive this adjustment. For the dose rates from the HI-STORM, dose rates on the side of the cask are compared. The adjustment factors selected this way are also shown in Table 11.3.1.

It is to be noted that both scenarios represent rather extreme cases. Practical experience from the many casks loaded so far show that dose rates significantly below those shown here can be achieved. This not so much the result of a less bounding content of the canisters, but the result of operational improvements of the loading process that reduce durations of the presence of workers near the casks, that increase the distance to the cask areas with higher dose rates, and the use of temporary shielding.

Using Table 11.3.1 data, the dose data for fuel loading (wet to dry storage) is provided in Table 11.3.2.

For each step in Table 11.3.2, the task description, average number of personnel in direct radiation field, exposure duration in direct radiation field and average dose rate are identified. The relative locations refer to all HI-STORM FW overpacks. The dose rate location points around the transfer cask and overpack were selected based on actual experience in loading HI-STORM 100 Overpacks. Cask operators typically work with workers entering and exiting the immediate cask area. To account for this, an average number of workers and average dose rates are used. The tasks involved in each step presented in Table 11.3.2 are not provided in any specific order.

Table 11.3.1

ASSUMED PARAMETERS FOR DOSE ESTIMATE UNDER SHORT-TERM OPERATIONS AND UNDER LONG-TEM STORAGE

	Item	Value
1.	**MPC-Contents (MPC-37)[1]**	**Representative: 45,000 MWD/MTU and 4.5 years** **Adjustment Factors for bounding content:** **HI-TRAC 1.4** **HI-STORM 2.0**
2.	**Weight of HI-TRAC VW Full of Fuel and Water**	**125 tons**
3.	**HI-STORM Concrete Density**	**150 lb/cubic feet**

[1] The case of MPC-37 is used but similar results are expected for all MPC types.

TABLE 11.3.2: ESTIMATED PERSON-MREM DOSE FOR LOADING THE HI-STORM FW SYSTEM				
Task Description (See Chapter 9 for detailed description of operations)	**Average Number of Personnel in Direct Radiation Field**	**Exposure Duration in Direct Radiation Field (mins)**	**Average Dose Rate at worker location (mrem/hr)**	**Exposure Representative / Bounding (mrem)**
Fuel loading and removal of the transfer cask and MPC from the spent fuel pool (includes: fuel loading, fuel assembly identification check, MPC lid installation, Lift Yoke attachment to the HI-TRAC VW, HI-TRAC VW removal from the spent fuel pool, preliminary decontamination, HI-TRAC VW movement to the DAS, Lift Yoke removal and decontamination. Background radiation of 1 mrem/hr assumed.	3	800	1.0	40.0 / 40.0
MPC preparation for closure (includes: HI-TRAC VW and MPC decontamination, radiation surveys, partial MPC pump down, annulus seal removal, partial lowering of annulus water level, annulus shield ring installation, weld system installation); workers assumed to be on scaffolding near the top of the HI-TRAC.	3	30	55.7	83.5 / 116.9

TABLE 11.3.2: ESTIMATED PERSON-MREM DOSE FOR LOADING THE HI-STORM FW SYSTEM				
Task Description (See Chapter 9 for detailed description of operations)	**Average Number of Personnel in Direct Radiation Field**	**Exposure Duration in Direct Radiation Field (mins)**	**Average Dose Rate at worker location (mrem/hr)**	**Exposure Representative / Bounding (mrem)**
MPC Closure (includes MPC lid to shell welding, weld inspection). Assumes welding machine uses standard Holtec pedestal which provides additional shielding. Holtec auxiliary shielding methods and equipment assumed. Assumes operators are present for 10% of the total duration.	2	185	55.7	34.3 / 48.0
MPC Preparation for Storage (includes: MPC hydrostatic testing, draining, drying and backfill, vent and drain port cover plate installation, welding, weld inspection and leakage testing). Holtec auxiliary shielding methods and equipment assumed. Assumes operators are present for 20% of the total duration.	2	170	175.4	198.7 / 278.2
MPC Closure Ring Installation (includes: closure ring to MPC shell welding, weld inspection and leakage testing of the MPC primary closure). Holtec auxiliary shielding methods and equipment assumed (lead blankets, water shields, etc.) Assumes operators are present for 10% of the total duration.	2	80	229.4	61.2 / 85.6

TABLE 11.3.2: ESTIMATED PERSON-MREM DOSE FOR LOADING THE HI-STORM FW SYSTEM				
Task Description (See Chapter 9 for detailed description of operations)	**Average Number of Personnel in Direct Radiation Field**	**Exposure Duration in Direct Radiation Field (mins)**	**Average Dose Rate at worker location (mrem/hr)**	**Exposure Representative / Bounding (mrem)**
HI-STORM FW system preparation for receiving MPC (includes: HI-STORM FW overpack positioning at transfer location, HI-STORM lid removal, Mating Device installation on HI-STORM FW overpack).	3	160	0	0 / 0
MPC Transfer (attachment of MPC lifting device, movement of HI-TRAC VW to transfer location, placement of HI-TRAC VW in Mating Device, bottom lid removal, MPC lowering, HI-TRAC VW removal, MPC lift device removal). Holtec auxiliary shielding methods and equipment assumed. Assumes operators are present for 10% of the total duration.	3	120	148	88.8 / 124.3
HI-STORM FW overpack movement to the ISFSI (will include: movement of the HI-STORM FW overpack from the fuel building to placement of the HI-STORM FW overpack on the ISFSI pad, disconnecting transporter, attachment of HI-STORM FW lid, attachment of thermal monitoring system). Holtec auxiliary shielding methods and equipment assumed. Assumes operators are present for 50% of the total duration.	3	220	37.3	205.2 / 410.4

TABLE 11.3.2: ESTIMATED PERSON-MREM DOSE FOR LOADING THE HI-STORM FW SYSTEM				
Task Description (See Chapter 9 for detailed description of operations)	**Average Number of Personnel in Direct Radiation Field**	**Exposure Duration in Direct Radiation Field (mins)**	**Average Dose Rate at worker location (mrem/hr)**	**Exposure Representative / Bounding (mrem)**
TOTAL EXPOSURE (person-mrem)				711.6 / 1103.4

BASES

ACTIONS (continued)	A.1 The heat removal system is inoperable. The blockage should be cleared within 8 hours for Version V or Version V2 to remove the obstructions in the air flow path and maintain fuel temperature and TRANSFER CASK component materials temperatures within design limits. The Completion Time is consistent with the thermal analyses of this event, which show that fuel temperature and all TRANSFER CASK component materials temperatures remain below their accident temperature limits indefinitely for Version V and up to 16 hours after event initiation for Version V2.
	B.1 If the heat removal system cannot be restored to operable status within the 64 hours for Version V or 8 hours for Version V2, the fuel or the TRANSFER CASK materials, may experience elevated temperatures. The fuel temperature accident limits will not be exceeded; however, the TRANSFER CASK component materials temperature limits may be exceeded with the heat removal system inoperable. Efforts must continue to restore the heat removal system to operable status by removing the air flow obstructions.
	B.2 In addition to Required Action B.1, Supplemental Cooling to the TRANSFER CASK may be provided to protect the integrity of the TRANSFER CASK component materials. If the completion time has been exceeded, an engineering evaluation must be performed to determine if deterioration of the TRANSFER CASK component materials, which prevents it from performing its design function has occurred. If the evaluation is successful and the air flow obstructions have been cleared, the TRANSFER CASK heat removal system may be considered operable. Transfer of the MPC into an OVERPACK removes the TRANSFER CASK from the LCO Applicability.

BASES	
SURVEILLANCE REQUIREMENTS	SR 3.1.4 The short-term integrity of the fuel in the HI-TRAC VW Version V or V2 is dependent on the ability of the TRANSFER CASK to reject heat from the MPC to the environment. Visual observation of all inlet and outlet air ducts are unobstructed ensures that air flow past the MPC is occurring and heat transfer is taking place. Any amount of blockage renders the heat removal system inoperable and this LCO is not met. The Frequency of 8 hours is reasonable, based on the time necessary for fuel cladding and TRANSFER CASK components to heat up to unacceptable temperatures, assuming design basis heat loads, and allowing for corrective actions to take place upon discovery of blockage of air ducts.
REFERENCES	1. FSAR Chapter 4 2. ANSI/ANS 57.9-1992