Table 1.2.1

KEY SYSTEM DATA FOR HI-STAR 100
(See Chapter 2 for limitation on damaged fuel and fuel debris permitted in various MPC types)

	QUANTITY	NOTES
Types of MPCs included in this revision of the submittal	3~~2~~	2~~1~~ for PWR 1 for BWR
MPC storage capacity:	MPC-24	Up to 24 intact zircaloy or stainless steel clad PWR fuel assemblies
	MPC-68	Up to 68 intact zircaloy or intact stainless steel clad BWR fuel assemblies or damaged zircaloy clad fuel assemblies in damaged fuel containers in the MPC-68 or Up to 4 damaged fuel containers with zircaloy clad BWR fuel debris and the complement intact or damaged zircaloy clad BWR fuel assemblies within an MPC-68F.
	MPC-32	Up to 32 intact zircaloy ~~or stainless steel~~clad PWR fuel assemblies

1.4 GENERIC CASK ARRAYS

The only system required for storage of the HI-STAR 100 System is the loaded overpack itself. The HI-STAR 100 System is stored in a vertical orientation. A typical ISFSI storage pattern is illustrated in Figure 1.4.1, which shows an array in a rectangular layout pattern. The required center-to-center spacing between the modules (layout pitch), ~~is~~ guided by heat transfer considerations, is specified to be 12 feet in both orthogonal directions. ~~Table 1.4.1 provides the nominal pitch, determined by heat transfer calculations in Chapter 4.~~ The pitch may be increased to suit facility considerations. In horizontal storage, the lateral distance between cask centerlines shall not be less than 12 feet (3.7 m) to provide practical access for handling equipment. As described in Chapter 4, a site-specific spacing evaluation is required for horizontal casks. Such a site-specific evaluation shall use the same thermal; evaluation methodology as that described in Chapter 4 and shall satisfy the same acceptance criteria.

~~*(1058ºF) based on PNL-4835 [2.0.7]. Further, the MPC is backfilled with 99.995% pure helium at a pressure specified in Chapter 12 during canister sealing operations to promote heat transfer and prevent cladding degradation.*~~ *The thermal design and operation of the MPC in the HI-STAR 100 cask meets the intent of the review guidance contained in ISG-11, Revision 3 [2.0.5]. Specifically, the ISG-11 provisions that are explicitly invoked and satisfied are:*

i. The thermal acceptance criteria for all commercial spent fuel (CSF) authorized by the USNRC for operation in a commercial reactor are unified into one set of requirements.

ii. The maximum value of the calculated temperature for all CSF under long-term normal conditions of storage must remain below 400ºC (752ºF). For short-term operations, including canister drying, helium backfill, and on-site cask transport operations, the fuel cladding temperature must not exceed 400ºC (752ºF) for high burnup fuel (HBF) and 570ºC (1058ºF) for moderate burnup fuel. (Note, high burnup fuel is not included in the contents for the HI-STAR 100 storage system, see Tables 2.1.13, 2.1.14, and 2.1.15, and is therefore not allowed for storage in the HI-STAR 100.)

iii. The maximum fuel cladding temperature as a result of an off-normal or accident event must not exceed 570ºC (1058ºF).

iv. During loading operations, repeated thermal cycling may occur but are limited to cladding temperature variations that are less than 65ºC (~~117~~117ºF) each and the number of excursions to less than 10.

To achieve compliance with the above criteria, certain design and operational changes are necessary, as summarized below.

i. The peak fuel cladding temperature limit (PCT) for long term storage operations and short term operations is generally set at 400ºC (752ºF). However, for MPCs containing all moderate burnup fuel, the fuel cladding temperature limit for short-term operations is set at 570ºC (1058ºF) because the nominal fuel cladding stress is shown to be less than 90 MPa [2.0.6]. Appropriate analyses have been performed as discussed in Chapter 4 and operating restrictions have been added to ensure these limits are met.

ii. A method of drying, such as forced helium dehydration (FHD) is used if the above temperature limits for short-term operations cannot be met.‡

iii. The off-normal and accident condition PCT limit remains unchanged at 570 ºC (1058ºF).

The MPC cavity is dried, either with FHD or vacuum drying, and then it is backfilled with high purity helium to promote heat transfer and prevent cladding degradation.

The design temperatures for the structural steel components of the MPC are based on the temperature limits provided in ASME Section II, Part D, tables referenced in ASME Section III, Subsection NB

‡ Note: In this FSAR vacuum drying is always accepted. This statement is for future design changes, ~~if~~ when applicable.

~~*The non-mechanistic tip-over requirement does not apply if the cask is stored horizontally and fastened to a low profile wide base trailer which has been qualified by static equilibrium to maintain its horizontal configuration(i.e., no toppling) if subjected to the simultaneous ZPAs corresponding to the site's Design Basis Earthquake. The vertical ZPA shall be assumed to act to reduce the weight of the system to maximize the likelihood of predicted overturning.*~~

A tip-over analysis of a loaded HI-STAR 100 overpack stored on an ISFSI pad in the horizontal orientation is not required because:

i) the HI-STAR 100 is analyzed in Appendix 3.A for a 72" horizontal side drop onto an ISFSI pad that complies with the design parameters in Table 2.2.9;

ii) if the HI-STAR 100 overpack is stored horizontally the center of gravity height of the overpack above the ISFSI pad must not exceed 72 inches.

2.2.3.3 Fire

The possibility of a fire accident near an ISFSI site is considered to be extremely remote due to the absence of significant combustible materials. The only credible concern is related to a transport vehicle fuel tank fire engulfing ~~a~~ the loaded cask while it is being moved to the ISFSI.

The HI-STAR 100 System must withstand temperatures due to a fire event. The fire accident for storage is conservatively specified to be the result of the spillage and ignition of 50 gallons (18~~9~~90~~.3~~ liters) of combustible transporter fuel. The HI-STAR overpack surfaces are considered to receive an incident radiation and convection heat flux from the fire. Table ~~2.2.8 provides~~2.2.8 provides the fire duration based on the amount of flammable materials assumed. The temperature of the fire is assumed to be 1475 ^{o}F (80~~2~~0 ^{o}C) in accordance with 10CFR71.73.

The accident condition design temperatures for the HI-STAR 100 System, and the fuel rod cladding limits are specified in Table 2.2.3. The specified ~~accident condition~~ fuel cladding temperature limits ~~is the short-term temperature limit based on a PNL report [2.0.7]~~are based on the temperature limits specified in ISG-11, Rev 3 [2.0.5].

2.2.3.4 Partial Blockage of MPC Basket Vent Holes

The HI-STAR 100 System is designed to withstand reduction of flow area due to partial blockage of the MPC basket vent holes. As the MPC basket vent holes are internal to the confinement barrier, the only events that could partially block the vents are fuel cladding failure and debris associated with this failure, or the collection of ~~-~~crud at the base of the stored SNF assembly. The HI-STAR 100 System maintains the SNF in an inert environment with fuel rod cladding temperatures below accepted values (Table 2.2.3). Therefore, there is no credible mechanism for gross fuel cladding degradation during storage in the HI-STAR 100 System. For the storage of damaged BWR fuel assemblies or fuel debris, the assemblies and fuel debris will be placed in damaged fuel containers

Table 2.2.1

DESIGN PRESSURES

Pressure Location	**Condition**	**Pressure - ~~(~~psig~~)~~ (kPa)**
MPC Internal Pressure	Normal	100 (~~689.5~~689~~700690~~)
	Off-Normal	100 (~~689.569~~689~~700~~)
	Accident	~~125~~ 200 (1379~~8602~~)
MPC External Pressure/Overpack Internal Pressure	Normal	40 (275~~6~~)
	Off-Normal	40 (275~~6~~)
	Accident	60 (414~~0140~~)
Overpack External Pressure	Normal	(0) Ambient
	Off-Normal	(0) Ambient
	Accident	300 (20~~68~~0~~7~~0)
Overpack Neutron Shield Enclosure Internal Pressure	Normal	30 (~~206.8~~207~~07~~)
	Off-Normal	30 (~~206.8~~207~~07~~)
	Accident	N/A†

† The overpack neutron shield enclosure is equipped with two rupture disks which are set a relief pressure of 30 psig (~~206.8~~200~~7~~ kPa). Therefore, the pressure cannot exceed 30 psig (~~206.8~~20~~7~~0 kPa).

Finally, it is noted that the structural evaluations performed for the HI-STAR 100 System under normal and off-normal conditions are bounding regardless of whether the overpack is stored vertically or horizontally. This is because of the following considerations.

i) The normal design pressures and temperatures listed in Tables 2.2.1 and 2.2.3 of this FSAR, which are used as input to the structural evaluations, are bounding for both storage orientations (horizontal and vertical) as confirmed by the thermal evaluations in Chapter 4.

ii) As stated above, normal handling encompasses both vertical and horizontal orientations (regardless of the final storage configuration). Accordingly, the MPC fuel basket and enclosure shell are analyzed separately in Subparagraph 3.4.4.3.1 under axial and lateral loads due to normal handling. The evaluations for the HI-STAR lifting trunnion, pocket trunnions, and the various lid lifting points are not affected by the final storage configuration.

iii) As explained in Section 11.1 of this FSAR, the maximum component temperatures due to off-normal environmental conditions, when the HI-STAR 100 overpack is stored horizontally, are bounded by the results presented in Chapter 3 of the HI-STAR 100 SAR for normal conditions of transport. Moreover, the bounding component temperatures reported in Tables 3.4.10, 3.4.17, and 3.4.18 of the HI-STAR 100 SAR are less than the design temperatures listed in Table 2.2.3 of this FSAR for normal conditions of storage. In addition, the off-normal design pressures are the same as the normal design pressures, as shown in Table 2.2.1 and further discussed in Paragraph 11.1.1.3, so that the load combinations for normal and off-normal conditions are subsumed into a bounding set of load combinations.

In summary, the structural evaluations for the normal and off-normal load combinations are bounding for a loaded HI-STAR 100 overpack stored in the vertical or horizontal orientation.

3.1.2.2 Allowables

The important to safety components of the HI-STAR 100 System are listed in Table 2.2.6. Allowable stresses, as appropriate, are tabulated for these components for all service conditions in Tables 3.1.6 through 3.1.16.

In Subsection 2.2.5, the applicable service level from the ASME Code for determination of allowables is listed. Table 2.2.14 provides a tabulation of normal, off-normal, and accident conditions and the service levels defined in the ASME Code, along with the applicable loadings for each service condition.

Allowable stresses and stress intensities are calculated using the data provided in the ASME Code and Tables 2.2.10 through 2.2.12. Tables 3.1.6 through 3.1.16 contain numerical values of

and the ISFSI. The overturning moment due to a force F_T applied at height H^* is balanced by a restoring moment from the reaction to the cask buoyant force KW acting at radius D/2.

$$F_T\, H^* = KW \frac{D}{2}$$

or

$$F_T = \frac{K\,W\,D}{2\,H^*}$$

W is the minimum weight of the storage overpack with an empty MPC.

We have,

W = 189,000 lb~~.~~ (841 kN) (Table 3.2.1)

H^* = 103.2" (2,621~~590.8~~ mm) (maximum height of mass center per Table 3.2.2)

D = 83.25" (2,115~~4.55~~ mm) (Holtec Drawing 3913, Sheet 2~~1~~)

K = 0.719 (calculated)

so that

F_T = 54,811~~55,456~~ lb~~.~~ (244~~247~~ kN)

F_T is the horizontal drag force at incipient tip-over.

F = Cd A V^* = 11,080 lbs~~.~~ (49.3 kN) (drag force at 13 feet/sec (3.96 m/s))

The safety factor against overturning, SF_2, is given by

$$SF_2 = \frac{F_T}{F} = 4.95 > 1.1\,(\text{required})$$

The sliding and overturning evaluations performed above for the vertically oriented cask are also bounding for a loaded HI-STAR overpack stored on an ISFSI pad in the horizontal orientation for the following reasons.

The sliding evaluation depends on the mass and the projected (or impingement) area of the cask. These two quantities are quite similar whether the cask is stored horizontally or vertically. In the horizontal orientation, the cask rests on a low profile supporting structure, which adds slightly to the total mass and the impingement area, but these increases are offsetting with respect to the net

sliding force. Therefore, the sliding evaluation presented above is also bounding for the horizontal orientation.

With regard to the overturning evaluation, the cask is inherently more stable when it is stored horizontally because of its lower overall height above the ISFSI pad[1] and the added mass of the low profile supporting structure, both of which reduce the net overturning moment on the cask due the moving floodwater.

3.4.7 Seismic Event on HI-STAR 100 (Load Case C in Table 3.1.1)

3.4.7.1 Stability

3.4.7.1.1 Stability of Vertically Arrayed HI-STAR 100

The HI-STAR 100 System plus its contents are subject to the design basis seismic event consisting of three orthogonal statistically independent acceleration time-histories (orthogonal components). The HI-STAR 100 System can be considered as a rigid body subject to a net horizontal inertia force and a vertical inertia force for the purpose of performing a conservative analysis to determine the maximum ZPA that will not cause incipient tipping. The vertical seismic loading is conservatively assumed to act in the most unfavorable direction (upwards) at the same instant. The vertical seismic load is assumed to be equal to or less than the net horizontal load with ε being the ratio of vertical component to one of the horizontal components. Define D as the contact patch diameter, and H_{CG} as the height of the centroid of an empty HI-STAR 100 System (no fuel).

D = 83.25" (2,115~~4.55~~ mm) -(Drawing 3913, Sheet 2)

Tables 3.2.1 and 3.2.2 give HI-STAR 100 weight data and center-of-gravity heights.

The weights and center-of-gravity heights are reproduced here for calculation of the composite center of gravity height of the overpack together with an empty MPC.

Weight (pounds) (kN)	H of C.G. Height (Inches) (mm)
Overpack - W_o = 153,710 (684)	99.7 (2,532~~.38~~)
MPC-24 - W_{24} = 40,868 (182)	109.0 + 6 = 115.0† (2,769~~8.6~~+152~~.4~~ = 2,921)
MPC-32 - W_{32} = 34,507 (153)	113.2 + 6 = 119.2† (2,875~~.28~~+152~~.4~~ = 3,027~~7.68~~)
MPC-68 - W_{68} = 37,591 (167)	111.5 + 6 = 117.5† (2,832~~.1~~+152~~.4~~ = 2,984~~.5~~)

† MPC centroids reported in Section 3.2 are measured from the base of the MPC.

[1] Center of gravity height (i.e., centerline) of the cask must not exceed 72” above the ISFSI pad surface.

3.4.7.1.2 Stability of Horizontally Arrayed HI-STAR 100

~~Similar to vertically arrayed HI-STAR 100 System, it can be horizontally stored.~~ The stability of the horizontally stored HI-STAR 100 System is determined following the same approach as vertically arrayed HI-STAR 100 System. During the horizontal storage, the HI-STAR 100 may be staged on cribbing or other supporting structures. Define H_{CGH} as the height from the centroid of an empty HI-STAR 100 System (no fuel) to the base of the supporting structure, and B as the width of the supporting structure. Because the dimension of HI-STAR 100 System is greater in height than diameter~~direction than in base direction~~, the HI-STAR 100 System tends to overturn~~ing~~ about the longitudinal axis. Therefore, the horizontal seismic accelerations do not need to be combined vectorially, and the maximum of two horizontal seismic accelerations is applied as an overturning force at ~~H_{CGH}~~ the centroid height of the cask (H_{CGH}). The overturning moment is balanced by a vertical reaction force, acting at the outermost contact patch of the supporting structure location B. The overturning moment is:

$$WGH_{CGH}$$

The moment that resists “incipient tipping” is:

$$W\ (1\text{-}\varepsilon G)\ (B/2)$$

Equating the two moments to ensure equilibrium ~~of moments~~ yields

$$WGH_{CGH} = W\ (1\text{-}\varepsilon G)\ (B/2)$$

~~or~~and, after canceling W, ~~and having~~ the following relationship results

$$\frac{H_{CGH}}{B} \leq \frac{(1 - \varepsilon\, G)}{2G}$$

~~W~~where ε and G are defined in Subsection 3.4.7.1.1.

The HI-STAR 100 System is stable as long as the ~~left term in the above equation is less than or equal to the right~~above inequality is satisfied~~ term. A site specific analysis is required if this is not the case.~~ As an example, for H_{CGH} = 72” and B = 96”, the following results are obtained for different values of ε.

Acceptable Horizontal g-Level in Each of Two Orthogonal Directions	Vertical Acceleration Multiplier (ε)	Vectorial Sum of Acceptable Horizontal Accelerations (g)
0.40	1.0	0.566
0.44	0.75	0.622
0.46	0.667	0.651

0.50	0.50	0.707

If the above inequality cannot be satisfied for a particular site, then a 3-D time history analysis may be performed to demonstrate the stability of the HI-STAR 100 overpack in its horizontal storage configuration.

Lastly, it is noted that the supporting structure that is used for horizontal storage of a loaded HI-STAR overpack is an ancillary device, which is classified as not important to safety (NITS). The justification for its NITS designation is because the HI-STAR 100 overpack has been designed and qualified to withstand a horizontal drop from a height of 72” above the ISFSI pad surface (see Appendix 3.A). Therefore, by requiring that the center of gravity height of the HI-STAR 100 overpack above the ISFSI pad be limited to 72” or less when stored horizontally, a gross failure of the supporting structure does not pose any safety risk to the HI-STAR 100 System.

3.4.7.2 Primary Stresses in the HI-STAR 100 Structure

A simplified calculation to assess the flexural bending stress in the HI-STAR 100 structure under the limiting seismic event (at which tipping is incipient) is presented in the following:

From the acceptable acceleration table presented above, the maximum horizontal acceleration is 0.354g. The corresponding lateral seismic load, F, is given by F = 0.354 W. This load will be maximized if the upper bound HI-STAR 100 weight (W = 245,000 lbs. (1,090 kN), from Table 3.2.4) is used. Accordingly, F = (0.354) (245,000) = 86,730 lbs (F = (0.354) (1,090) = 385.9~~86~~ kN).~~.~~

$\mu =$ 0.25

For establishing the appropriate value of ω, reference is made to the USAEC publication TID-7024, "Nuclear Reactor and Earthquakes", page 35, 1963, which states that the significant energy of all seismic events in the U.S. essentially lies in the range of 0.4 to 10 Hz. Taking the mid-point value

$\omega = (2\pi)\ (0.5)\ (0.4+10) = 32.7$ rad/sec.

The numerical solution of the above equation yields the maximum displacement of the slider block x_{max} as 0.047 inches (1.19 mm), which is negligible compared to the spacing between casks.

Calculations performed at lower values of ω show an increase in x_{max} with reducing ω. At 1 Hz, for example, x_{max} = 1.236 inches (31.39 mm). It is apparent from the above that there is a large margin of safety against inter-module collision within the HI-STAR 100 arrays at an ISFSI, where the minimum installed spacing is approximately 4 feet (1.22 m) (Table 1.4.1).

3.4.8 Tornado Wind and Missile Impact (Load Case B in Table 3.1.1 and Load Case 06 in Table 3.1.5

During a tornado event, the HI-STAR 100 System is conservatively assumed to be subjected to a constant wind force. It is also subject to impacts by postulated missiles. The maximum wind speed is specified in Table 2.2.4 and the three missiles, designated as large, intermediate, and small, are described in Table 2.2.5.

The post impact response of the HI-STAR 100 System is required to assess stability.

~~Appendix 3.C~~Supplement 19 of [3.4.13] contains results for the post-impact response of the HI-STAR 100 where it is demonstrated there that the combination of tornado missile plus either steady tornado wind or instantaneous tornado pressure drop causes a rotation of the HI-STAR 100 to a maximum angle of inclination 18.23 degrees from vertical. This is less than the angle required to overturn the cask. The appropriate value for the drag coefficient used in the computation of the lateral force on the overpack from tornado wind is justified in Supplement 19 of [3.4.13]~~Appendix 3.C~~.

Supplement 19 of [3.4.13]~~Appendix 3.C~~ computes the maximum force acting on the projected area of the cask to be

F = 26,380 lbs~~.~~ (117.3 kN)

This is bounded by the seismic overturning force computed in Section 3.4.7. Therefore, the overpack stress analysis performed in Section 3.4.7 remains governing.

As discussed in Section 3.4.6 in relation to the flood event, the HI-STAR 100 overpack is inherently more stable when it is stored horizontally on an ISFSI pad. Therefore, the stability

evaluation summarized above for a freestanding HI-STAR 100 overpack in the vertical orientation, under the combined effects of tornado wind and a large missile impact, is also bounding for a HI-STAR 100 overpack stored horizontally.

The penetration potential of the missile strikes (Load Case 06 in Table 3.1.5) is examined in ~~Appendix 3.G~~Supplement 22 of [3.4.13]. It is shown in Supplement 22 of [3.4.13]~~Appendix 3.G~~ that there will be no penetration of the intermediate shells surrounding the inner shell of the overpack or penetration of the top closure plate. Therefore, there will be no radiological release associated with any missile strikes during a tornado. The following results summarize the work in Supplement 22 of [3.4.13]~~Appendix 3.G~~.

a. The small missile will dent any surface it impacts, but no significant puncture force is generated.

b. The following table summarizes the denting and penetration analysis performed for the intermediate missile in Supplement 22 of [3.4.13]~~Appendix 3.G~~. ~~-~~Denting is used to connote a local deformation mode encompassing material beyond the impacting missile envelope, while penetration is used to indicate a plug type penetration mechanism involving only the target material immediately under the impacting missile.

Intermediate Missile Strike – Denting and Penetration		
Location	Denting (in~~.~~) (mm)	Penetration
Outer Enclosure Shell	2.77 (70.4~~36~~)	Yes (> 0.5 in~~.~~) (> 12.7 mm)
Intermediate shells	2.81 (71.4~~37~~)	No (< 8.5 in~~.~~) (< 216~~5.9~~ mm)
Closure plate	3.00 (76.2)	No (< 6 in~~.~~) (< 152~~.4~~ mm)

Since the intermediate missile generates a large puncture force for a short duration, the effect of this puncture force on the overpack closure bolts is examined in ~~Appendix 3.F~~Supplement 21 of [3.4.13].

The primary stresses that arise due to an intermediate missile strike on the side of the overpack and in the center of the overpack top lid are also determined in Supplement 22 of [3.4.13]~~Appendix 3.G~~. It is demonstrated there that Level D stress limits are not exceeded in either the side shell or the top lid. The safety factor in the overpack inner shell, considered as a cantilever beam under tip load, is computed, as is the safety factor in the top lid, considered as a centrally loaded plate. The applied load, in each case, is the missile impact load. A summary of the results is given in the table below:

HI-STAR 100 Missile Impact - Global Stress Results (Load Case 06 in Table 3.1.5)			
Item	Value (ksi) (MPa)	Allowable (ksi) (MPa)	Safety Factor
Inner Shell - Side	12.6 (86.9)	48.2 (332~~.3~~)	3.83

Strike			
Intermediate Shell – Side Strike -	14.3 (98.6)	39.1 (270~~69.6~~)	2.73
~~Top Lid~~Closure Plate - (End Strike)	48.45 (334.1~~05~~)	64.6 (445~~.~~4)~~.~~	1.33

The above summary table does not include the circumferential fabrication stress since these have been designated as self-limiting, and therefore fall into the category of a secondary stress which need not be included in a Level D stress evaluation.

The calculated indentation and penetration results in the above tables are valid for both storage orientations. When the HI-STAR 100 overpack is stored horizontally, it is also vulnerable to a missile impact strike on the overpack bottom plate. However, the bottom plate is the same thickness, and is made from the same material, as the overpack closure plate, so the results for the closure plate can be extended to the bottom plate.

3.4.9 Non-Mechanistic Tip-over, Side and Vertical Drop Events

Pursuant to the provision in NUREG-1536, a non-mechanistic tip-over of a loaded HI-STAR 100 System on to the ISFSI pad is considered. Analyses are also performed to determine the maximum deceleration sustained by a side or vertical free fall of a loaded HI-STAR 100 System onto the ISFSI pad. The object of the analyses is to demonstrate that the plastic deformation in the fuel basket is sufficiently limited to permit the stored SNF to be retrieved by normal means and that there is no significant loss of radiation shielding in the system.

Ready retrievability of the fuel is presumed to be ensured if stress levels in the MPC structure remain below Level D limits during the postulated drop events.

Subsequent to the accident events, the overpack must be shown to contain the shielding so that unacceptable radiation levels do not result from the accident.

Appendix 3.A provides a description of the dynamic finite element analyses undertaken to establish the decelerations resulting from the postulated event. A non-mechanistic tip-over is considered together with a side and end drop of a loaded HI-STAR 100 System. A dynamic finite element analysis of each event is performed using a commercial finite element code well suited for such dynamic analyses with interface impact and non-linear material behavior. This code and methodology have been fully benchmarked against Lawrence Livermore Laboratories test data and correlation [3.4.12].

It is shown in Appendix 3.A that the peak deceleration is less than 60g's for tip-over. Table 3.A.3 shows that the maximum deceleration level at the top of the cask is 66.0~~52.8~~ g's, while the corresponding deceleration level at the top of the fuel basket is 59.8~~47.8~~ g's. For the case of a vertical drop from a height of 21" (533~~.4~~ mm), the bounding longitudinal deceleration is 52.3~~51.9~~ g's. Finally, for a side drop from a height of 72" (1,829~~8.8~~ mm), the maximum deceleration is 49.7~~2~~ g's.

4.3 SPECIFICATIONS FOR COMPONENTS

HI-STAR 100 System materials and components designated as "Important to Safety" (i.e., required to be maintained within their safe operating temperature ranges to ensure their intended function) which warrant special attention are summarized in Table 4.3.1. Long-term stability and continued neutron shielding ability of Holtite-A neutron shield material under normal storage conditions are ensured when material exposure temperatures are maintained below the maximum allowable limit. The integrity of the overpack helium retention boundary is assured by maintaining the temperature of the mechanical seals within the manufacturer's recommended operating temperature limits. Long-term integrity of SNF is ensured by the HI-STAR 100 System thermal performance, which demonstrates that fuel cladding temperatures are maintained below design basis limits. Neutron absorber~~Boral~~ materials used in MPC baskets for criticality control (~~a~~ composite materials composed of B_4C and aluminum) are~~is~~ stable up to 1000°F (538°C) for short-term and 850°F (454°C) for long-term dry storage†. However, for conservatism, a significantly lower maximum temperature limit is imposed.

Compliance to 10CFR72 requires, in part, identification and evaluation of short-term off-normal and severe hypothetical accident conditions. The inherent mechanical stability characteristics of cask materials and components ensure that no significant functional degradation is possible due to exposure to short-term temperature excursions outside the normal long-term temperature limits. For evaluation of HI-STAR 100 System thermal performance under off-normal or hypothetical accident conditions, material temperature limits for short-duration events are provided in Table 4.3.1. *Fuel temperature limits mandated by ISG-11 [4.3.8] are adopted for evaluation of cladding integrity under normal, short term operations, off-normal and accident conditions. These limits are applicable to all fuel types, burnup levels and cladding materials approved by the NRC for power generation.*

It is recognized that hydrides present in irradiated fuel rods (predominantly circumferentially oriented) dissolve at cladding temperatures above 400°C [4.3.9]. Upon cooling below a threshold temperature (T_p), the hydrides precipitate and reorient to an undesirable (radial) direction if cladding stresses at the hydride precipitation temperature T_p are excessive. For moderate burnup fuel, T_p is conservatively estimated as 350°C [4.3.9]. In a recent study, PNNL has evaluated a number of bounding fuel rods for reorientation under hydride precipitation temperatures for MBF [4.3.9]. The study concludes that hydride reorientation is not credible during short-term operations involving low to moderate burnup fuel (up to 45 GWD/MTU). Accordingly, the higher ISG-11 temperature limit is justified for moderate burnup fuel and is adopted for short-term operations for MBF fueled MPCs (see Table 4.3.1).

~~Demonstration of fuel cladding integrity against the potential for degradation and gross rupture throughout the entire dry cask storage period is mandated by the Code of Federal Regulations (Part 72, Section 72.72(h)). The specific criteria required to demonstrate fuel cladding integrity is set forth in the NUREG-1536 document as listed below.~~

~~A. The dry cask storage system shall ensure a less than 0.5 percent~~

† AAR Structures Boral thermophysical test data.

4.4.2 Maximum Temperatures

4.4.2.1 Maximum Temperatures Under Normal Storage Conditions

The ~~two~~ MPC basket designs developed for the HI-STAR 100 System have been analyzed to determine the temperature distribution under long-term normal storage conditions. The MPC baskets are considered to be loaded at design basis maximum heat loads with PWR or BWR fuel assemblies, as appropriate. The systems are considered to be arranged in an ISFSI array and subjected to design basis normal ambient conditions with insolation.

Applying the radiative blocking factor applicable for the worst case cask location, converged temperature contours are shown in Figures 4.4.17 and 4.4.18 for the MPC-24, and MPC-68 basket designs in a vertical orientation. The temperatures in these two figures are in degrees Kelvin. Analogous figures for horizontal orientation storage of the MPC-24 and MPC-68 are presented in Figures 3.4.16 and 3.4.17 of the HI-STAR 100 transportation SAR. The calculated temperatures presented in this chapter are based on an array of analyses that incorporate many conservatisms. As such, the calculated temperatures are upper bound values which would exceed actual temperatures.

The maximum fuel clad temperatures for zircaloy clad fuel assemblies in vertically-oriented casks are listed in Tables 4.4.10 and 4.4.11, which also summarize maximum calculated temperatures in different parts of the HI-STAR 100 System. Figures 4.4.21 and 4.4.22 show the axial temperature variation of the hottest fuel rod in the MPC-24 and MPC-68 basket designs, respectively. Figures 4.4.24 and 4.4.25 show the radial temperature profile in the MPC-24 and MPC-68 basket designs, respectively, in the horizontal plane where maximum fuel cladding temperature is indicated.

The thermal performance **trend** of an MPC-32 basket in a vertically-oriented HI-STAR 100 overpack can be ascertained by examining the performance of the three MPCs (i.e., MPC-24, MPC-32 and MPC-68) in a similar overpack **if the two casks are sufficiently similar**. To this end, the relative performance of the same three MPCs in the 125-ton HI-TRAC transfer cask is examined. Like the HI-STAR 100 cask, the 125-ton HI-TRAC transfer cask is a vertically-oriented non-ventilated overpack that encloses the MPC within a multi-layered body through which heat must be conducted to reach the cylindrical outer surface. Also like the HI-STAR 100 cask, the heat that reaches the cylindrical outer surface of the 125ton HI-TRAC transfer cask is rejected to the ambient via natural convection and thermal radiation. The 125-ton HI-TRAC thermal analysis described in Revision 2 of the HI-STORM 100 FSAR [4.4.10] used essentially the same thermal analysis methods as the thermal analyses described in this chapter. The following comparison of the HI-STAR 100 and 125-ton HI-TRAC overpacks shows the similarity between the two.

Parameter	HI-STAR 100 Overpack	125-ton HI-TRAC Overpack
Internal Cavity Diameter	68 3/4”	68 3/4”
Internal Cavity Length	191 1/8”	191 1/4”

Thickness of Cylindrical Metal Wall Inside Neutron Shield	8 1/2”	6 1/4”
Thickness of Cylindrical Neutron Shield	4.3”	5.36”
Thickness of Cylindrical Metal Shell Outside Neutron Shield	1/2”	1/2”
Overall Outside Diameter	93 3/4”	92.97”
Thickness of Closure Lid	6”	4 3/4”
Thickness of Base	6”	5 1/2”
Conduction Through Multi-Layered Overpack Wall?	Yes	Yes
Ventilated Overpack?	No	No
Surface Heat Rejection Via Natural Convection and Thermal Radiation	Yes	Yes

It is apparent from the above information that the characteristics, heat transfer mechanisms, and dimensions of the HI-STAR 100 and the 125-ton HI-TRAC are sufficiently similar to yield the same thermal performance trend.

As stated in Paragraph 4.5.2.1 of the applicable HI-STORM 100 FSAR [4.4.10], “A bounding steady-state analysis of the HI-TRAC transfer cask has been performed using the hottest MPC, the highest design-basis decay heat load … and design-basis insolation levels.” The MPC-68 bounds the MPC-32, both of which have nearly identical decay heat loads (28.19 kW for the MPC-68 and 28.74 for the MPC-32, a difference of less than 2%). These results from an analysis of a similar style cask using the same thermal analysis methods leads to the conclusion that the thermal performance of a vertically-oriented HI-STAR 100 System containing an MPC-32 basket will be bounded by the thermal performance of a vertically-oriented HI-STAR 100 System containing an MPC-68.

The maximum fuel clad temperatures for zircaloy clad fuel assemblies in horizontally-oriented casks are listed in Tables 3.4.10 and 3.4.11 of the HI-STAR 100 transportation SAR, which also summarize maximum calculated temperatures in different parts of the HI-STAR 100 System.

As discussed in Subsection 4.4.1.1.1, the thermal analysis is performed using a submodeling process where the results of an analysis on an individual component are incorporated into the analysis of a larger set of components. Specifically, the submodeling process yields directly computed fuel temperatures from which fuel basket temperatures are indirectly calculated. This modeling process differs from previous analytical approaches wherein the basket temperatures were evaluated first and then a basket-to-cladding temperature difference calculation by Wooten-Epstein or other means provided a basis for cladding temperatures. Subsection 4.4.1.1.2 describes the calculation of an effective fuel assembly thermal conductivity for an equivalent homogenous region. It is important to note that the result of this analysis is a function for thermal conductivity versus temperature. This function for fuel thermal conductivity is then input to the

[4.2.10] "Qualification of METAMIC for Spent-Fuel Storage Application", EPRI Report 1003137, (October 2001), EPRI, Palo Alto, CA.

[4.2.11] "Sourcebook for METAMIC Performance Assessment", Holtec Report HI-2043215, Holtec International, Marlton, NJ, 08053.

[4.3.1] Deleted.~~Levy, I.S., et al., "Recommended Temperature Limits for Dry Storage of Spent Light Water Reactor Zircaloy Clad Fuel Rods in Inert Gas", PNL-6189, (May 1987).~~

[4.3.2] Deleted.

[4.3.3] Deleted.~~"Characteristics of Spent Fuel High-Level Waste, and Other Radioactive Wastes Which May Require Long-Term Isolation", DOE/RW-0184, (December 1987).~~

[4.3.4] Deleted.~~Johnson, Jr., A.B. and Gilbert, E.R., "Technical Basis for Storage of Zircaloy-Clad Spent Fuel in Inert Gases", PNL-4835, (September 1983).~~

[4.3.5] Deleted.~~Cunningham et. al., "Evaluation of Expected Behavior of LWR Stainless Steel-Clad Fuel in Long-Term Dry Storage", EPRI TR-106440, (April 1996).~~

[4.3.6] Deleted.~~Schwartz, M.W., Witte, M.C., "Spent Fuel Cladding Integrity During Dry Storage", LLNL, UCID-22181 (September 1987).~~

[4.3.7] Deleted.~~"Temperature Limit Determination for the Inert Dry Storage of Spent Nuclear Fuel", EPRI TR-103949, (May 1994).~~

[4.3.8] "Cladding Considerations for the Transportation and Storage of Spent Fuel," Interim Staff Guidance – 11, Revision 3, USNRC, Washington, DC.

[4.3.9] Lanning and Beyer, "Estimated Maximum Cladding Stresses for Bounding PWR Fuel Rode During Short Term Operations for Dry Cask Storage," PNNL White Paper, (January 2004).

[4.4.1] Wooton, R.O. and Epstein, H.M., "Heat Transfer from a Parallel Rod Fuel Element in a Shipping Container", Battelle Memorial Institute, 1963.

[4.4.2] Rapp, D., "Solar Energy", Prentice-Hall, Inc., Englewood Cliffs, NJ, (1981).

[4.4.3] Deleted.

[4.4.4] Deleted.~~Holman, J.P., "Heat Transfer," 6th ed., McGraw Hill Book Co., (1986).~~

[4.4.5] Sanders et al., "A Method for Determining the Spent-Fuel Contribution to Transport Cask Containment Requirements," Sandia Report SAND90-2406-TTC-

in the MPC cavity is vaporized. At this point, the operation of the FHD system moves on to steadily lowering the relative humidity and bulk temperature of the circulating helium gas (Phase 2). The demoisturizer module, equipped with the facility to chill flowing helium, plays the principal role in the dehydration process in Phase 2.

4.A.2 Design Criteria

The design criteria set forth below are intended to ensure that design and operation of the FHD system will drive the partial pressure of the residual vapor in the MPC cavity to ≤ 3 torr if the temperature of helium exiting the demoisturizer has met the value and duration criteria provided in the HI-STAR 100 technical specifications. The FHD system shall be designed to ensure that during normal operation (i.e., excluding startup and shutdown ramps) the following criteria are met:

- *i. The temperature of helium gas in the MPC shall be at least 15ºF higher than the saturation temperature at coincident pressure.*
- *ii. The pressure in the MPC cavity space shall be less than or equal to 60.3 psig (75 psia).*
- *iii. The recirculation rate of helium shall be sufficiently high (minimum hourly throughput equal to ten times the nominal helium mass backfilled into the MPC for fuel storage operations) so as to produce a turbulated flow regime in the MPC cavity.*
- *iv. The partial pressure of the water vapor in the MPC cavity will not exceed 3 torr if the helium temperature at the demoisturer outlet is ≤ 21ºF for a period of 30 minutes.*

In addition to the above system design criteria, the individual modules shall be designed in accordance with the following criteria:

- *i. The condensing module shall be designed to de-vaporize the recirculating helium gas to a dew point of 120ºF or less.*
- *ii. The demoisturizer module shall be configured to be introduced into its helium conditioning function after the condensing module has been operated for the required length of time to assure that the bulk moisture vaporization in the MPC has been completed.*
- *iii. The helium circulator shall be sized to effect the minimum flow rate of circulation required by the system design criteria described above.*
- *iv. The pre-heater module shall be engineered to ensure that the temperature of the helium gas in the MPC meets the system design criteria described above.*

The MPC-24, MPC-32 and MPC-68 are qualified for storage of SNF with different combinations of maximum burnup levels and minimum cooling times. Tables 2.1.13 through 2.1.15 ~~Figure 2.1.6 specifies~~ specify the acceptable maximum burnup levels and minimum cooling times for storage of zircaloy clad fuel in the MPC-24, MPC-32 and the MPC-68 ~~(Appendix B to the Certificate of Compliance presents this data in tabular form)~~. Table 2.1.11 specifies the acceptable maximum burnup levels and minimum cooling times for storage of stainless steel clad fuel. The values in ~~Figure 2.1.6 and~~ Tables 2.1.11, and 2.1.13 through 2.1.15 were chosen based on an analysis of the maximum decay heat load that could be accommodated within each MPC. The shielding analyses presented in this chapter used the burnup and cooling time combinations listed below which are either equal to or conservatively bound the acceptable burnup levels and cooling times shown in Tables 2.1.11, and 2.1.13 through 2.1.15~~Figure 2.1.6 and Table 2.1.11~~.

Maximum Burnup and Minimum Cooling Times Analyzed		
Zircaloy Clad Fuel		
MPC-24 ~~& MPC-32~~	**MPC-32**	**MPC-68**
40,000 MWD/MTU 5 year cooling	40,000 MWD/MTU 8 year cooling	35,000 MWD/MTU 5 year cooling
47,500 MWD/MTU 8 year cooling	45,000 MWD/MTU 11 year cooling	45,000 MWD/MTU 9 year cooling
N/A	N/A	30,000 MWD/MTU 18 year cooling (6x6 intact, damaged and MOX fuel)
Stainless Steel Clad Fuel		
MPC-24		**MPC-68**
30,000 MWD/MTU 9 year cooling		22,500 MWD/MTU 10 year cooling
40,000 MWD/MTU 15 year cooling		N/A

~~It should be noted that the combinations of fuel assemblies' maximum burnup and minimum cooling time analyzed in this chapter conservatively bound those stated in Appendix B to the Certificate of Compliance.~~

~~Appendix B to the Certificate of Compliance requires that, in the MPC-24, for a minimum cooling time of 5 years, the maximum burnup is 28,700 MWD/MTU, and for 15 year cooling the maximum burnup is 42,100 MWD/MTU for PWR fuel assemblies without Burnable Poison Rod Assemblies (BPRAs). PWR fuel assemblies containing BPRAs are limited to 28,300~~

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

In accordance with ALARA practices, design objective dose rates are established for the HI-STAR 100 in Section 2.3.5.2 as: ~~125~~ 130 mrem/hour (1.3 mSv/h) on the radial surface of the overpack, and 375 mrem/hour (3.75 mSv/h) in areas above and below the neutron shield in the radial direction.

The dose rates presented in this section are calculated at 40,000 MWD/MTU and 5-year cooling for the MPC-24 ~~and~~, 40,000 MWD/MTU and 8-year cooling for MPC-32, and 35,000 MWD/MTU and 5-year cooling for the MPC-68. Based on a comparison of the normal condition dose rates at the fuel mid-plane for the various burnup and cooling time combinations analyzed, these were chosen as the worst case for the MPC-24, MPC-32 and the MPC-68. Section 5.4 provides a detailed list of dose rates at several cask locations for all burnup and cooling times analyzed.

Figure 5.1.1 identifies the locations of the dose points referenced in the summary tables. The bottom shield shown in this figure is temporary shielding which may be used during on-site horizontal handling operations. Dose Point #7 is located directly below the overpack bottom plate or directly below the bottom shield when it is attached. Dose Points #1, #3, and #4 are not contact doses, but rather, in-air doses at the locations shown. 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.

Tables 5.1.~~2~~ 1 ~~and~~ through 5.1.3 provide the maximum dose rates adjacent to the overpack during normal conditions for each of the MPCs. Tables 5.1.~~5~~ 4 ~~and~~ through 5.1.6 provide the maximum dose rates at one meter from the overpack.

The dose to any real individual at or beyond the controlled area boundary is required to be below 25 mrem (0.25 mSv) per year. The minimum distance to the controlled area boundary is 100 meters from the ISFSI. Only the MPC-24 and MPC-32 ~~was~~ were used in the calculation of the dose rates at the controlled area boundary. The MPC-24 was chosen because its dose rates are equivalent or greater than the dose rates from the MPC-68 as shown in Tables 5.1.2, 5.1.3, 5.1.5, and 5.1.6. Table 5.1.7 presents the annual dose to an individual from a single cask and various arrays of casks, assuming 100% occupancy (8760 hours). The minimum distance required for the corresponding dose is also listed. These values were calculated for the MPC-24 with a burnup of 40,000 MWD/MTU and a 5-year cooling time, and for the MPC-32 with a burnup of 40,000 MWD/MTU and a 8-year cooling time. It will be shown in Section 5.4.3 that ~~this~~ these burnup and cooling time combinations result~~s~~ in the highest offsite dose for the combinations of maximum burnup and minimum cooling time analyzed. It is noted that these data are provided for illustrative purposes only. A detailed site specific evaluation of dose at the controlled area boundary will be performed for each ISFSI in accordance with 10CFR72.212, as stated in Chapter 12, Operating Controls and Limits. The site specific evaluation will consider dose from other portions of the facility and will consider the specifics of the fuel being stored (burnup and

cooling time).

Figure 5.1.2 is an annual dose versus distance graph for the ~~cask~~ configurations of the cask with MPC-24 provided in Table 5.1.7. This curve, which is based on 100% occupancy, is provided for illustrative purposes only and will be re-evaluated on a site-specific basis.

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

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

Section 5.2.4 presents the Co-60 sources from the BPRAs and TPDs that are permitted for storage in the HI-STAR 100. Section 5.4.6 demonstrates that the maximum dose rates presented in this section bound the dose rates from fuel assemblies containing either BPRAs or TPDs.

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

Section 5.2.3 presents the gamma and neutron source for the design basis intact stainless steel clad fuel. The dose rates from this fuel are provided in Section 5.4.5.

The analyses summarized in this section demonstrate that the HI-STAR 100 System is in compliance with the 10CFR72.104 limits and ALARA practices.

5.1.2 Accident Conditions

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

> Any individual located on or beyond the nearest boundary of the controlled area may not receive from any design basis accident the more limiting of a total effective dose equivalent of 5 Rem (0.05 Sv), 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 (0.5 Sv). The lens dose equivalent shall not exceed 15 Rem (0.15 Sv) and the shallow dose equivalent to skin or to any extremity shall not exceed 50 rem (0.5 Sv). The

Table 5.1.1

DOSE RATES ADJACENT TO OVERPACK FOR NORMAL CONDITIONS
MPC-32 WITH DESIGN BASIS ZIRCALOY CLAD FUEL AT WORST CASE
BURNUP AND COOLING TIME
40,000 MWD/MTU AND ~~5~~8-YEAR COOLING
(The values in parentheses are in mSv/h)

Dose Point[†] Location	**Fuel Gammas[††] (mrem/hr) (mSv/h)**	**^{60}Co Gammas (mrem/hr) (mSv/h)**	**Neutrons (mrem/hr) (mSv/h)**	**Totals (mrem/hr) (mSv/h)**
1	5.59 (0.0559)	220.07 (2.2007)	84.52 (0.8452)	310.18 (3.1018)
2	49.16 (0.4916)	0.03 (0.0003)	23.37 (0.2337)	72.56 (0.7256)
3	1.95 (0.0195)	80.91 (0.8091)	67.31 (0.6731)	150.16 (1.5016)
4	0.97 (0.0097)	35.84 (0.3584)	39.15 (0.3915)	75.96 (0.7596)
5	0.25 (0.0025)	0.58 (0.0058)	58.78 (0.5878)	59.6 (0.596)
6 (dry MPC)[†††]	12.96 (0.1296)	258.78 (2.5878)	141.89 (1.4189)	413.62 (4.1362)
7 (no temp. shield)	64.04 (0.6404)	1476.27 (14.7627)	438.45 (4.3845)	1978.75 (19.7875)
7 (with temp. shield)	20.4 (0.204)	323.69 (3.2369)	20.57 (0.2057)	364.66 (3.6466)

~~DELETED~~

† Refer to Figure 5.1.1.

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

††† Overpack closure plate not present.

Table 5.1.4

DOSE RATES AT ONE METER FOR NORMAL CONDITIONS
MPC-32 WITH DESIGN BASIS ZIRCALOY CLAD FUEL AT WORST CASE
BURNUP AND COOLING TIME
40,000 MWD/MTU AND ~~5~~8-YEAR COOLING
(The values in parentheses are in mSv/h)

Dose Point† Location	**Fuel Gammas†† (mrem/hr) (~~mrem~~Sv/h)**	**^{60}Co Gammas (mrem/hr) (~~mrem~~Sv/h)**	**Neutrons (mrem/hr) (~~mrem~~Sv/h)**	**TOTALS (mrem/hr) (~~mrem~~Sv/h)**
1	5.07 (0.0507)	23.84 (0.2384)	9.24 (0.0924)	38.14 (0.3814)
2	21.42 (0.2142)	0.63 (0.0063)	8.62 (0.0862)	30.67 (0.3067)
3	3.58 (0.0358)	13.42 (0.1342)	8.74 (0.0874)	25.75 (0.2575)
4	2.24 (0.0224)	14.56 (0.1456)	8.97 (0.0897)	25.78 (0.2578)
5	0.09 (0.0009)	0.28 (0.0028)	16.72 (0.1672)	17.1 (0.171)
7 (no temp. shield)	31.07 (0.3107)	712.91 (7.1291)	119.75 (1.1975)	863.73 (8.6373)
7 (with temp. shield)	7.27 (0.0727)	129.28 (1.2928)	15.14 (0.1514)	151.69 (1.5169)

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† Refer to Figure 5.1.1.

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

Table 5.1.7

DOSE RATES FOR ARRAYS ~~OF MPC-24~~
WITH DESIGN BASIS ZIRCALOY CLAD FUEL
~~40,000 MWD/MTU AND 5-YEAR COOLING~~
(The values in parentheses are in mSv/year)

Array Configuration	1 cask	2x2	2x3	2x4	2x5
MPC-24 (40,000 MWD/MTU AND 5-YEAR COOLING)					
Annual Dose (mrem/year)† (mSv/year)	13.55 (0.1355)	18.60 (0.1860)	13.84 (0.1384)	18.45 (0.1845)	23.06 (0.2306)
Distance to Controlled Area Boundary (meters)††, †††	300	350	400	400	400
MPC-32 (40,000 MWD/MTU AND 8-YEAR COOLING)					
Annual Dose (mrem/year)† (mSv/year)	9.16 (0.0916)	13.96 (0.1396)	11.22 (0.1122)	14.96 (0.1496)	18.7 (0.1870)
Distance to Controlled Area Boundary (meters)††, †††	300	350	400	400	400

† 100% occupancy is assumed.

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

††† Actual controlled area boundary dose rates will be lower because the maximum permissible burnup, as specified in Tables 2.1.13 and 2.1.15, for the ~~5-year~~stated cooling time~~as specified in the Technical Specifications~~~~Table 2.1.13~~ is lower than the burnup analyzed for the design basis fuel used in this chapter.

Table 5.1.9

DOSE RATES AT ONE METER FOR ACCIDENT CONDITIONS
DESIGN BASIS ZIRCALOY CLAD FUEL
AT WORST CASE BURNUP AND COOLING TIME
(The values in parentheses provided ~~for MPC-32~~ are in mSv/h)

Dose Point[†] Location	Fuel Gammas[††] (mrem/hr) (mSv/h)	^{60}Co Gammas (mrem/hr) (mSv/h)	Neutrons (mrem/hr) (mSv/h)	Totals (mrem/hr) (mSv/h)
MPC-24 (40,000 MWD/MTU AND 5-YEAR COOLING)				
2 (Accident Condition)	100.98 (1.0098)	1.80 (0.0180)	388.94 (3.8894)	491.73 (4.9173)
2 (Normal Condition)	42.67 (0.4267)	1.06 (0.0106)	7.74 (0.0774)	51.47 (0.5147)
MPC-32 (40,000 MWD/MTU AND ~~5~~8-YEAR COOLING)				
2 (Accident Condition)	47.65 (0.4765)	1.6 (0.016)	421.85 (4.2185)	471.11 (4.7111)
2 (Normal Condition)	21.42 (0.2142)	0.63 (0.0063)	8.62 (0.0862)	30.67 (0.3067)
MPC-68 (35,000 MWD/MTU AND 5-YEAR COOLING)				
2 (Accident Condition)	98.28 (0.9828)	1.48 (0.0148)	360.93 (3.6093)	460.69 (4.6069)
2 (Normal Condition)	43.01 (0.4301)	0.60 (0.0060)	7.50 (0.0750)	51.11 (0.5111)

† Refer to Figure 5.1.1.

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

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

5.2.1 Gamma Source

Tables 5.2.~~4~~ 3 through 5.2.6 provide the gamma source in MeV/s and photons/s as calculated with SAS2H and ORIGEN-S for the design bases intact fuels for the MPC-32, MPC-24~~, MPC-32~~ and MPC-68, and the design basis damaged fuel. Table 5.2.16 provides the gamma source in MeV/s and photons/s for the design basis MOX fuel. NUREG-1536 [5.2.1] states that "only gammas with energies from approximately 0.8 to 2.5 MeV will contribute significantly to the dose rate." Conservatively, only energies in the range of 0.7 MeV-3.0 MeV are used in the shielding calculations. Photons with energies below 0.7 MeV are too weak to penetrate the steel of the overpack, and photons with energies above 3.0 MeV are too few to contribute significantly to the external dose. This section provides the radiation source for each of the burnup levels and cooling times evaluated.

The primary source of activity in the non-fuel regions of an assembly arise from the activation of ^{59}Co to ^{60}Co. The primary source of ^{59}Co in a fuel assembly is 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. As a conservative measure, the impurity level of ^{59}Co was assumed to be 1000 ppm or 1.0 gm/kg. Therefore, Inconel and stainless steel in the non-fuel regions are both conservatively assumed to have the 1.0 gm/kg impurity level.

The gamma source from the activation of the grid spacers is negligible in comparison to the source from the active fuel. In addition, in most fuel elements that obtain high burnups, the grid spacers are manufactured from zircaloy which does not activate to produce a gamma source. Therefore, for the PWR design basis fuel assembly, no contribution to the fuel region gamma source from activation of grid spacers is provided in the source term calculations. The BWR assembly grid spacers are zircaloy, however, some assembly designs have steel springs in conjunction with the grid spacers. The gamma source for the BWR fuel assembly includes the activation of these springs associated with the grid spacers.

The PWR fuel assemblies loaded into the MPC-32 basket may have non-zircaloy grid spacers. The calculations for PWR fuel assemblies with non-zircaloy grid spacers are provided in Subsection 5.4.9.

The non-fuel data listed in Table 5.2.1 was taken from References [5.2.2], [5.2.4], and [5.2.5].

The BWR masses are for an 8x8 fuel assembly. These masses are also appropriate for the 7x7 assembly since the masses of the non-fuel hardware from a 7x7 and an 8x8 are approximately the same. The masses listed are those of the steel components. The zircaloy in these regions was not included because zircaloy does not produce significant activation. These masses are larger than most other fuel assemblies from other manufactures. This, in combination with the conservative ^{59}Co impurity level, results in a conservative estimate of the ^{60}Co activity.

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

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

2. The activity calculated in Step 1 for the region of interest was modified by the appropriate scaling factors listed in Table 5.2.7. These scaling factors were taken from Reference [5.2.3].

Tables 5.2.~~9~~ 8 ~~and~~ through 5.2.10 provide the ^{60}Co activity utilized in the shielding calculations in the non-fuel regions of the assemblies for the MPC-32, MPC-24~~, MPC-32~~ and the MPC-68. The design basis damaged and MOX fuel assemblies are conservatively assumed to have the same ^{60}Co source strength as the BWR intact design basis fuel. This is a conservative assumption as the design basis damaged fuel and MOX fuel are limited to a significantly lower burnup and longer cooling time than the intact design basis fuel.

In addition to the two sources already mentioned, a third source arises from (n,γ) reactions in the material of the MPC and the overpack. This source of photons is properly accounted for in MCNP when a neutron calculation is performed in a coupled neutron-gamma mode.

5.2.2 Neutron Source

It is well known that the neutron source strength increases as enrichment decreases, for a constant burnup and decay time. This is due to the increase in Pu content in the fuel which increases the inventory of other transuranium nuclides such as Cm. The gamma source also varies with enrichment, although only slightly. Because of this effect and in order to obtain conservative source terms, low initial fuel enrichments were chosen for the BWR and PWR design basis fuel assemblies. The enrichments are appropriately varied as a function of burnup. Table 5.2.23 presents the ^{235}U initial enrichments for various burnup ranges from 20,000 - 50,000 MWD/MTU for PWR and BWR zircaloy clad fuel. These enrichments are based on Reference [5.2.6]. Table 8 of this reference presents average enrichments for burnup ranges. The initial enrichments chosen in Table 5.2.23 are approximately the average enrichments for the burnup range that are 5,000 MWD/MTU less than the ranges listed in Table 5.2.23. These

enrichments are below the enrichments typically required to achieve the burnups that were analyzed. Therefore, the source term calculations are conservative.

The neutron source calculated for the design basis intact fuel assemblies for the MPC-32, MPC-24~~, MPC-32~~ and MPC-68 and the design basis damaged fuel are listed in Tables 5.2.~~12~~ 11 through 5.2.14 in neutrons/s. Table 5.2.17 provides the neutron source in neutrons/sec for the design basis MOX fuel assembly. ^{244}Cm accounts for approximately 96% of the total number of neutrons produced, with slightly over 2% originating from (α,n) reactions within the UO_2 fuel. The remaining 2% derive from spontaneous fission in various Pu and Cm radionuclides. In addition, any neutrons generated from subcritical multiplication, (n,2n) or similar reactions are properly accounted for in the MCNP calculation.

5.2.3 Stainless Steel Clad Fuel Source

Table 5.2.18 lists the characteristics of the design basis stainless steel clad fuel. The fuel characteristics listed in this table are the input parameters that were used in the shielding calculations described in this chapter. The active fuel length listed in Table 5.2.18 is actually longer than the true active fuel length of 122 inches (310 cm) for the WE 15x15 and 83 inches (211cm) for the A/C 10x10. Since the true active fuel length is shorter than the design basis zircaloy clad active fuel length, it would be incorrect to calculate source terms for the stainless steel fuel using the correct fuel length and compare them directly to the zircaloy clad fuel source terms because this type of approach would not reflect the potential change in dose rates at the center of the cask (center of the active fuel). As an example, if it is assumed that the source strength for both the stainless steel and zircaloy fuel is 144 photons/s and that the active fuel lengths of the stainless steel fuel and zircaloy fuel are 83 inches (211 cm) and 144 inches (366 cm), respectively; the source strengths per inch of active fuel would be different for the two fuel types, 1.73 photons/s/inch (0.68 photons/s/cm) and 1 photons/s/inch (0.39 photons/s/cm) for the stainless steel and zircaloy fuel, respectively. The result would be a higher photon dose rate at the center of the cask with the stainless steel fuel than with the zircaloy clad fuel; a conclusion that would be overlooked by just comparing the source terms. This is an important consideration because the stainless steel clad fuel differs from the zircaloy clad in one important aspect: the stainless steel cladding will contain a significant photon source from Cobalt-60 which will be absent from the zircaloy clad fuel.

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

Tables 5.2.19 through 5.2.22 list the neutron and gamma source strengths for the design basis stainless steel clad fuel. It is obvious from these source terms that the neutron source strength for the stainless steel fuel is lower than for the zircaloy fuel. However, this is not true for all photon energy groups. The peak energy group is from 1.0 to 1.5 MeV which results from the large Cobalt activation in the cladding. Since some of the source strengths are higher for the stainless steel fuel, Section 5.4.5 presents the dose rates at the center of the overpack for the stainless steel fuel. The center dose location is the only location of concern since the end fittings are assumed to be the same mass as the end fittings for the zircaloy clad fuel. In addition, the burnup is lower and the cooling time is longer for the stainless steel fuel compared to the zircaloy clad fuel.

It should be noted that currently PWR spent fuel assemblies with stainless steel clad have not been qualified for storage in the MPC-32 basket.

5.2.4 Non-Fuel Hardware

Rod cluster control assemblies and axial power shaping rods are not permitted for storage in the HI-STAR 100 system. However, burnable poison rod assemblies (BPRAs) and thimble plug devices (TPDs) are permitted for storage in the HI-STAR 100 System as an integral part of a PWR fuel assembly.

5.2.4.1 BPRAs and TPDs

Burnable poison rod assemblies (BPRA) (including wet annular burnable absorbers and similarly designed devices with different names) and thimble plug devices (TPD) (including orifice rod assemblies, guide tube plugs, and similarly designed devices with different names) are an integral, yet removable, part of a large portion of PWR fuel. The TPDs are not used in all assemblies in a reactor core but are reused from cycle to cycle. Therefore, these devices can achieve very high burnups. In contrast, BPRAs are burned with a fuel assembly in core and are not reused. In fact, many BPRAs are removed after one or two cycles before the fuel assembly is discharged. Therefore, the achieved burnup for BPRAs is not significantly different than fuel assemblies.

TPDs are made of stainless steel and contain a small amount of inconel. These devices extend down into the plenum region of the fuel assembly but do not extend into the active fuel region with the exception of the W 14x14 water displacement guide tube plugs. Since these devices are made of stainless steel, there is a significant amount of cobalt-60 produced during irradiation. This is the only significant radiation source from the activation of steel and inconel.

BPRAs are made of stainless steel in the region above the active fuel zone and may contain a small amount of inconel in this region. Within the active fuel zone the BPRAs may contain 2-24 rodlets which are burnable absorbers clad in either zircaloy or stainless steel. The stainless steel

decreases as the burnup continues to increase. This is due to a decrease in the Cobalt-60 production rate as the initial Cobalt-59 impurity is being depleted. Conservatively, a constant cooling time has been specified for burnups from 180,000 to 630,000 MWD/MTU for the TPDs.

Section 5.4.6 demonstrates that the dose rates from fuel assemblies containing BPRAs or TPDs ~~is~~ are bounded by the dose rates presented in Section 5.1.1.

It should be noted that currently PWR spent fuel assemblies with BPRAs or TPDs have not been qualified for storage in the MPC-32 basket.

5.2.5 Choice of Design Basis Assembly

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

5.2.5.1 PWR Design Basis Assembly

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

Table 5.2.24 presents the characteristics of the fuel assemblies analyzed to determine the design basis zircaloy clad PWR fuel assembly. The fuel assembly listed for each class is the assembly with the highest UO_2 mass. The St. Lucie and Ft. Calhoun classes are not present in Table 5.2.24. These assemblies are shorter versions of the CE 16x16 and CE 14x14 assembly classes, respectively. Therefore, these assemblies are bounded by the CE 16x16 and CE 14x14 classes and were not explicitly analyzed. Since the Haddam Neck and San Onofre 1 classes are stainless steel clad fuel, these classes were analyzed separately and are discussed below. All fuel assemblies in Table 5.2.24 were analyzed at the same burnup and cooling time. The initial enrichment used in the analysis is consistent with Table 5.2.23. The results of the comparison are provided in Table 5.2.26. These results indicate that the B&W 15x15 fuel assembly has the highest radiation source term of the zircaloy clad fuel assembly classes considered in Table ~~2.1.1~~5.2.24. This fuel assembly also has the highest UO_2 mass (see Table 5.2.24) which confirms that, for a given initial enrichment, burnup, and cooling time, the assembly with the

Table 5.2.1

DESCRIPTION OF DESIGN BASIS INTACT ZIRCALOY CLAD FUEL
(The values in parentheses are in cm)

	PWR	BWR
Assembly type/class	B&W 15×15	GE 7×7
Active fuel length (in.) (cm)	144 (365.76)	144 (365.76)
No. of fuel rods	208	49
Rod pitch (in.) (cm)	0.568 (1.4427~~2~~)	0.738 (1.8745)
Cladding material	zircaloy-4	zircaloy-2
Rod diameter (in.) (cm)	0.428 (1.0871~~2~~)	0.570 (1.4478)
Cladding thickness (in.) (cm)	0.0230 (5.842E-2)	0.0355 (9.017E-2)
Pellet diameter (in.) (cm)	0.3742 (0.9504~~685~~)	0.488 (1.2395)
Pellet material	UO_2	UO_2
Pellet density (gm/cc)	10.412 (95% of theoretical)	10.412 (95% of theoretical)
Enrichment (w/o ^{235}U)	3.4 and 3.6	2.9 and 3.2
Burnup (MWD/MTU)†	40,000 and 47,500 (MPC-24) 40,000 and 45,000 (MPC-32)	35,000 and 45,000 (MPC-68)
Cooling Time (years)†	5 and 8 (MPC-24) 8 and 11 (MPC-32)	5 and 9 (MPC-68)
Specific power (MW/MTU)	40	30
Weight of UO_2 (kg)††	562.029	225.177
Weight of U (kg)††	495.485	198.516

Notes:
1. The B&W 15x15 is the design basis assembly for the ~~following~~ fuel assembly classes listed in Table ~~2.1.1~~5.2.24~~: B&W 15x15, B&W 17x17, CE 14x14, CE 16x16, WE 14x14, WE 15x15, WE 17x17, St. Lucie, and Ft. Calhoun~~.
2. The GE 7x7 is the design basis assembly for the ~~following~~ fuel assembly classes listed in

† Burnup and cooling time combinations conservatively bound the acceptable burnup and cooling times listed in ~~Appendix B to the Certificate of Compliance~~Section 2.1.

†† Derived from parameters in this table.

Table ~~2~~5.~~1~~2.~~2:~~25 ~~GE BWR/2-3, GE BWR/4-6, Humboldt Bay 7x7, and Dresden 1 8x8~~.

Table 5.2.3

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

Lower Energy	**Upper Energy**	**40,000 MWD/MTU 8-Year Cooling**		**45,000 MWD/MTU 11-Year Cooling**	
(MeV)	(MeV)	(MeV/s)	(Photons/s)	(MeV/s)	(Photons/s)
7.0e-01	1.0	2.38E+14	2.80E+14	1.25E+14	1.48E+14
1.0	1.5	7.61E+13	6.09E+13	6.03E+13	4.82E+13
1.5	2.0	3.72E+12	2.13E+12	3.05E+12	1.74E+12
2.0	2.5	5.91E+11	2.63E+11	6.73E+10	2.99E+10
2.5	3.0	3.47E+10	1.26E+10	5.55E+09	2.02E+09
Totals		3.18E+14	3.43E+14	1.89E+14	1.98E+14

~~DELETED~~

Table 5.2.8

CALCULATED MPC-32 ^{60}Co SOURCE PER ASSEMBLY
FOR DESIGN BASIS ZIRCALOY CLAD FUEL
AT VARYING BURNUP AND COOLING TIMES
(The values in parentheses are in TBq)

Location	**40,000 MWD/MTU 8-Year Cooling (curies) (TBq)**	**45,000 MWD/MTU 11-Year Cooling (curies) (TBq)**
Lower end fitting	104.44 (3.864)	75.87 (2.807)
Gas plenum springs	7.97 (0.295)	5.79 (0.214)
Gas plenum spacer	4.57 (0.169)	3.32 (0.123)
Expansion springs	N/A	N/A
Grid spacer springs	N/A	N/A
Upper end fitting	51.23 (1.895)	37.21 (1.377)
Handle	N/A	N/A

~~DELETED~~

Table 5.2.11

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

Lower Energy (MeV)	**Upper Energy (MeV)**	**40,000 MWD/MTU 8-Year Cooling (Neutrons/s)**	**45,000 MWD/MTU 11-Year Cooling (Neutrons/s)**
1.0e-01	4.0e-01	9.94E+06	1.30E+07
4.0e-01	9.0e-01	5.08E+07	6.63E+07
9.0e-01	1.4	4.65E+07	6.08E+07
1.4	1.85	3.44E+07	4.49E+07
1.85	3.0	6.10E+07	7.95E+07
3.0	6.43	5.51E+07	7.19E+07
6.43	20.0	4.87E+06	6.35E+06
Totals		2.63E+08	3.43E+08

~~DELETED~~

5.3 MODEL SPECIFICATIONS

The shielding analysis of the HI-STAR 100 System was performed with MCNP-4A [5.1.1]. MCNP is a Monte Carlo transport code that offers a full three-dimensional combinatorial geometry modeling capability including such complex surfaces as cones and tori. This means that no gross approximations were required to represent the HI-STAR 100 System in the shielding analysis. A sample input file for MCNP is provided in Appendix 5.C.

As discussed in Section 5.1.1, off-normal conditions do not have any implications for the shielding analysis. Therefore, the MCNP models and results developed for the normal conditions also represent the off-normal condition. Section 5.1.2 discussed the accident conditions and stated that the only accident that would impact the shielding analysis would be a loss of the neutron shield. Therefore, the MCNP models of the HI-STAR 100 System normal condition have the neutron shield in place while the accident condition replaces the neutron shield with void.

5.3.1 Description of the Radial and Axial Shielding Configuration

Section 1.5 provides the drawings that describe the HI-STAR 100 System. These drawings were used to create the MCNP models used in the radiation transport calculations. Figures 5.3.~~2~~1 ~~and~~ through 5.3.3 show cross sectional views of the HI-STAR 100 overpack and MPC as it was modeled in MCNP for each of the MPCs. These figures were created with the MCNP two-dimensional plotter and are drawn to scale. The figures clearly illustrate the radial steel fins and pocket trunnions in the neutron shield region. Since the fins and pocket trunnions were modeled explicitly, neutron streaming through these components is accounted for in the calculations of the dose adjacent to the overpack and 1 meter dose. In Section 5.4.1, the dose effect of localized streaming through these compartments is analyzed. Figure 5.3.4 shows the MCNP models of the MPC-32 fuel basket. Figures 5.3.5 and 5.3.6 show the MCNP models of the MPC-24 and MPC-68 fuel baskets including the as-modeled dimensions. Figure 5.3.9 shows a cross sectional view of the HI-STAR 100 overpack with the as-modeled thickness of the various materials. Figure 5.3.10 is an axial representation of the HI-STAR 100 overpack with the various as-modeled dimensions indicated. ~~As Figure 5.3.10 indicates~~It should be noted ~~,~~that the thickness of the MPC-68 lid is 10.0 inches (25.4 cm), and the thickness of the MPC-24 lid and MPC-32 lid ~~are 10.0 inches (25.4 cm) and~~is 9.5 inches (24.13 cm)~~, respectively~~. Correspondingly, the MPC-internal cavity height~~s~~ of MPC-68 differs by 0.5 inch (1.27 cm) compared to that of MPC-24 and MPC-32. In the MCNP models of the MPC-24, MPC-32 and MPC-68, the actual lid thickness and internal cavity height for that particular MPC was used.

Calculations were performed to determine the acceptability of homogenizing the fuel assembly versus explicit modeling. Based on these calculations it was concluded that it was acceptable to homogenize the fuel assembly without loss of accuracy. The PWR fuel assembly modeled was the design basis fuel assembly, the B&W 15x15. The width of this homogenized fuel assembly in MCNP is equal to 15 times the pitch. The BWR fuel assembly modeled was an 8x8 fuel assembly. This is different from the 7x7 design basis fuel assembly used for the source term

calculations. However, it is conservative to use an 8x8 fuel assembly in the MCNP model since it contains less fuel and therefore less shielding than the 7x7 fuel assembly. The width of the BWR homogenized fuel assembly is equal to 8 times the pitch. Homogenization of the fuel assemblies resulted in a noticeable decrease in run time.

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

1. The basket material in the top and bottom 0.9 inches (2.286 cm) where the MPC basket flow holes are located is not modeled. The length of the basket not modeled (0.9 inches (2.286 cm)) was determined by calculating the equivalent area removed by the flow holes. This method of approximation is conservative because no material for the basket shielding is provided in the 0.9 inch (2.286 cm) area at the top and bottom of the MPC basket.

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

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

4. The MPC-24 basket is fabricated from 5/16 inch (0.7938 cm) thick cell plates. It is conservatively assumed for modeling purposes that the structural portion of the MPC-24 basket is uniformly fabricated from 9/32 inch (0.7144 cm) thick steel. The Boral and sheathing are modeled explicitly. This is conservative since it removes steel which would provide a small amount of additional shielding.

5. In the modeling of the BWR fuel assemblies, the zircaloy flow channel was not represented. This was done because it cannot be guaranteed that all BWR fuel assemblies will have an associated flow channel when placed in the MPC. The flow channel does not contribute to the source, but does provide some small amount of shielding. However, no credit is taken for this additional shielding.

6. In the MPC-24, conservatively, all Boral panels on the periphery were modeled with a reduced width of 5 inches (12.7 cm) compared to 6.25 inches (15.875 cm) or 7.5 inches (19.05 cm).

7. The MPC-68 is designed for two lid thicknesses: 9.5 inches (24.13 cm) and 10 inches (25.4 cm). Conservatively, all calculations reported in this chapter were performed with

Sections 4.4 and 4.5 demonstrate that all materials used in the HI-STAR 100 System remain below their design temperatures as specified in Table 2.2.3 during all normal conditions. Therefore, the shielding analysis does not address changes in the material density or composition as a result of temperature changes.

Chapter 11 discusses the effect of the various accident conditions on the temperatures of the shielding materials and the resultant impact on their shielding effectiveness. As stated in Section 5.1.2, there is only one accident that has any significant impact on the shielding configuration. This accident is the loss of the neutron shield in the HI-STAR 100 System as a result of fire or other damage. The change in the neutron shield was conservatively analyzed by assuming that the entire volume of the neutron shield was replaced by void.

As discussed in Chapter 5 of HI-STORM 100 FSAR [5.3.1], approved by the NRC, *the MPCs can be manufactured with one of two possible neutron absorbing materials: Boral or Metamic. Both materials are made of aluminum and B_4C powder. The Boral contains an aluminum and B_4C powder mixture sandwiched between two aluminum plates while the Metamic is a single plate. The minimum ^{10}B areal density is the same for Boral and Metamic while the thicknesses are essentially the same. Therefore, the mass of Aluminum and B_4C are essentially equivalent and there is no distinction between the two materials from a shielding perspective. As a result,* Table 5.3.2 *identifies the composition for Boral and no explicit calculations were performed with Metamic.*

~~5.3.3 MPC-32 Dose Rate Calculations~~

~~The dose rates presented in this chapter for the MPC-32 are estimated by scaling the MPC-24 dose rates provided in Tables 5.4.2 through 5.4.4, and Table 5.4.8 for normal conditions, and Table 5.1.8 for the accident conditions. The approach is summarized below:~~

- ~~To estimate the MPC-32 total side dose rates, the MPC-24 neutron dose rates are scaled up by a factor of 32/24. The same MPC-24 fuel gamma and Co-60 dose rates are used. This is an acceptable approach since the fuel assemblies in the outer cells will shield the gamma radiation from the fuel assemblies in the inner cells.~~
- ~~To estimate the MPC-32 total top and bottom dose rates, the MPC-24 neutron, gamma and Co-60 dose rates are scaled up by a factor of 32/24.~~

~~Overall, this approach is conservative since same combinations of fuel assembly maximum burnup and minimum cooling time as those for MPC-24 are used to estimate MPC-32 dose rates. While as shown in Tables 2.1.13 and 2.1.15, for the same cooling time, the maximum burnup for fuel to be stored in MPC-32 is less than that for fuel to be stored in MPC-24.~~

~~Figure 5.3.1 shows the cross sectional view of the HI-STAR 100 overpack with MPC-32 basket. Figure 5.3.4 shows the MPC-32 basket.~~

Table 5.3.2

COMPOSITION OF THE MATERIALS IN THE HI-STAR 100 SYSTEM

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

gammas.

The MPC-24 and the MPC-68 were analyzed. Figure 5.4.1 shows a quarter of the HI-STAR 100 overpack with 91 azimuthal bins drawn. There is one bin per steel fin and 8 bins in each Holtite-A region. This azimuthal binning structure was used in an infinite height two-dimensional model of the MPC and overpack. The dose was calculated in each of these bins and then compared to the average dose calculated over the surface to determine a peak-to-average ratio for the dose in that bin. The location of the pocket trunnion is shown in Figure 5.4.1. The pocket trunnion was modeled as solid steel. During storage, a shield plug shall be placed in the pocket trunnion recess, and during handling operations a steel rotation trunnion shall be placed in the pocket trunnion recess. To conservatively evaluate the peak-to-average ratio, the pocket trunnion is assumed to be solid steel. The peak-to-average ratio was calculated for the entire pocket trunnion which would correspond to the first seven azimuthal bins.

Table 5.4.10 provides the peak-to-average ratios that were calculated for the various dose components and locations. The peak-to-average ratios were essentially the same for all evaluated MPCs, therefore, only one set of values is shown. The values presented for the pocket trunnions are very conservative since the two-dimensional model represented the trunnion as infinite in height whereas the actual height is approximately 12 inches (30.48 cm). In addition, the pocket trunnion was represented as being axially adjacent to the active fuel which is not completely accurate for the design basis fuel. The infinite two-dimensional model therefore does not represent any leakage out of the pocket trunnion in the axial direction which would reduce the peaking effect.

Table 5.4.11 presents the dose rates at Dose Point #2 (see Figure 5.1.1) and the adjusted dose rates at this point to account for the streaming effects. An additional dose point labeled 2a is listed in this table. This location is axially adjacent to the pocket trunnion and approximately 6 feet (182.88 cm) below Dose Point #2. Based on these results it can be concluded that the streaming effect is noticeable but is not of significant concern.

5.4.2 Damaged Fuel Post-Accident Shielding Evaluation

As discussed in Section 5.2.5.2, the analysis presented below, even though it is for damaged fuel, demonstrates the acceptability of storing intact Humboldt Bay 6x6 and intact Dresden 1 6x6 fuel assemblies.

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

Using the volume in conjunction with the damaged fuel container, the axial height of rubble is calculated to be 80 inches (203.2 cm).

Dividing the total fuel gamma source for damaged fuel in Table 5.2.6 by the 80 inch (203.2 cm) rubble height provides a gamma source per inch of 9.68e+10 photon/s. Dividing the total neutron source for damaged fuel in Table 5.2.14 by 80 inches (203.2 cm) provides a neutron source per inch of 2.75e+5 neutron/s. These values are both bounded by the BWR design basis fuel gamma source per inch and neutron source per inch values of 1.76e+12 photon/s and 5.60e+5 neutron/s. These BWR design basis values were calculated by dividing the total source strengths in Tables 5.2.5 and 5.2.13 by the active fuel length of 144 inches (365.76 cm). Therefore, the design basis damaged fuel assembly is bounded by the design basis intact BWR fuel assembly for accident conditions. No explicit analysis of the damaged fuel dose rates are provided as they are bounded by the intact fuel analysis.

5.4.3 Site Boundary Evaluation

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

As an example of the methodology, the dose from a single MPC-24 cask and various arrays of MPC-24 casks at a distance greater than 100 meters was evaluated with MCNP. In the model the casks were placed on an infinite slab of concrete to account for earth-shine effects. The atmosphere was represented as dry air at a uniform density corresponding to 20 degrees C. The height of air modeled was 800 meters. This is more than sufficient to properly account for skyshine effects.

The annual dose, assuming 100% occupancy (8760 hours), at 300 meters from one cask is presented in Table 5.4.12 for MPC-24 and in Table 5.4.22 for MPC-32 at the varying maximum burnup and minimum cooling times analyzed. This table indicates that the 40,000 MWD/MTU and 5-year cooling is the bounding case for the evaluated~~se~~ MPC-24 combinations, and the 40,000 MWD/MTU and 8-year cooling is the bounding case for the evaluated MPC-32 combinations.

~~This~~ These tables also indicate~~s~~ that the dose due to neutrons is 21% or more of the total dose. This is an important observation because it implies that simplistic analytical methods such as point kernel techniques may not properly account for the neutron transmission and could lead to low estimates of the site boundary dose.

One of the features of MCNP is the ability to calculate the dose from particles that have passed through certain geometrical regions (referred to as surface or cell flagging). This technique was

used to estimate the fraction of the dose at distance from particles, both neutron and gamma, passing through the upper flange region of the overpack. This region is referred to as 3 and 4 on Figure 5.1.1. It was found that, for one cask, approximately 9% of the dose comes from this upper flange region. This is a significant fraction of the total dose and one that is only accounted for using three-dimensional analysis, such as MCNP, which properly includes the effects of neutron and gamma skyshine.

Since the upper flange region is located at the top of the cask, it is reasonable to conclude that this contribution to total dose would be unaffected by placing the cask in an array configuration.

The annual dose, assuming 100% occupancy, at distance from an array of casks was calculated in three steps.

1. The annual dose from the radiation leaving the side of a single HI-STAR 100 overpack was calculated at the distance desired. The side of the HI-STAR 100 overpack is defined as any surface between the bottom of the bottom plate and the top of the closure plate including the upper flange area. Dose value = A.

2. The annual dose from the radiation leaving the top of a single HI-STAR 100 overpack was calculated at the distance desired. The top of the HI-STAR 100 overpack is defined as the top of the closure plate. Dose value = B.

3. The annual dose from the radiation leaving the side of a HI-STAR 100 overpack, when it is in the center of a 3x3 array of casks, was calculated at the distance desired. The casks in the array have a 12 foot pitch (3.6576 m). Dose value = C.

The annual dose calculated in each of these three steps was averaged over a cylindrical surface at various distances from the source cask for ease of calculation. In step 3, the dose at the cylindrical surface included contributions from radiation that traveled between the surrounding casks and from radiation that traveled above the surrounding casks and scattered in air to reach the dose location. Therefore, the average dose values from step 3 include all possible paths for radiation to reach the dose location. The values from step 3 represent the dose from a cask in the second row of an array which is shielded by casks in the front row.

The doses calculated in the steps above are listed in Table 5.4.13 for 40,000 MWD/MTU and 5-year cooling for MPC-2~~5~~4, and in Table 5.4.23 for 40,000 MWD/MTU and 8-year cooling for MPC-32. Using these values, the annual dose (at the center of the long side) from an arbitrary 2 by Z array of HI-STAR 100 overpacks can easily be calculated. The following formula describes the method.

Z = number of casks along long side

Dose = ZA + 2ZB + ZC

As an example, the dose from a 2x3 array at 250 meters is presented.

1. The annual dose from the side of a single cask: Dose A = 24.53
2. The annual dose from the top of a single cask: Dose B = 0.63
3. The annual dose from the side of a cask in the center of a 3x3 array: Dose C = 8.81

Using the formula shown above (Z=3) the total dose at 250 meters from a 2x3 array of filled HI-STAR 100 overpacks with MPC-24 is 103.80 mrem/year (1.0380 mSv/year), assuming 100% occupancy.

An important point to notice here is that the dose from the side of the back row of casks is approximately 25% of the total dose. This is a significant contribution and one that would probably not be accounted for properly by simpler methods of analysis.

The results for various arrays of filled HI-STAR 100 overpacks can be found in Section 5.1.1.

5.4.4 Mixed Oxide Fuel Evaluation

The source terms calculated for the Dresden 1 GE 6x6 MOX fuel assemblies can be compared to the design basis source terms for the GE 7x7 assemblies which demonstrates that the MOX fuel source terms are bounded by the design basis source terms and no additional shielding analysis is needed.

Since the active fuel length of the MOX fuel assemblies is shorter than the active fuel length of the design basis fuel, the source terms must be compared on a per inch basis. Dividing the total fuel gamma source for the MOX fuel in Table 5.2.16 by the 110 inch (279.4 cm) active fuel height provides a gamma source per inch of 6.97e+10 photons/s. Dividing the total neutron source for the MOX fuel assemblies in Table 5.2.17 by 110 inches (279.4 cm) provides a neutron source strength per inch of 3.06e+5 neutrons/s. These values are both bounded by the BWR design basis fuel gamma source per inch and neutron source per inch values of 1.76e+12 photons/s and 5.60e+5 neutrons/s. These BWR design basis values were calculated by dividing the total source strength in Tables 5.2.5 and 5.2.13 by the active fuel length of 144 inches (365.76 cm). This comparison shows that the MOX fuel source terms are bound by the design basis source terms. Therefore, no explicit analysis of dose rates is provided for MOX fuel.

Since the MOX fuel assemblies are Dresden 1 6x6 assemblies, they can also be considered as damaged fuel or fuel debris. Using the same methodology as described in Section 5.4.2, the source term for the MOX fuel is calculated on a per inch basis assuming a post accident rubble height of 80 inches (203.2 cm). The resulting gamma and neutron source strengths are 9.59e+10 photons/s and 4.21e+5 neutrons/s. These values are also bounded by the design basis fuel gamma source per inch and neutron source per inch. Therefore, no explicit analysis of dose rates is provided for MOX fuel in a post accident configuration.

would produce a neutron source of 603.2 neutrons/sec (1.04E+8 * 5.8E-6). The steady state amount of Sb-124 activated in the antimony was calculated to be 39.9 curies (1.476 TBq). This activity level would produce a neutron source of 4.63E+6 neutrons/sec (39.9 * 1.16E+5) or 6.0E+4 neutrons/sec/inch[†] ~~(4.63E+6/77.25)~~(2.36E+4 neutrons/sec/cm). These calculations conservatively neglect the reduction in antimony and beryllium which would have occurred while the neutron sources were in the core and being irradiated at full reactor power.

Since this is a localized source (77.25 inches (196.215 cm) in length) it is appropriate to compare the neutron source per inch from the design basis Dresden Unit 1 fuel assembly, 6x6, containing an Sb-Be neutron source to the design basis fuel neutron source per inch. This comparison, presented in Table 5.4.15, demonstrates that a Dresden Unit 1 fuel assembly containing an Sb-Be neutron source is bounded by the design basis fuel.

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

5.4.8 Thoria Rod Canister

Based on a comparison of the gamma spectra from Tables 5.2.32 and 5.2.6 for the thoria rod canister and design basis 6x6 fuel assembly, respectively, it is difficult to determine if the thoria rods will be bounded by the 6x6 fuel assemblies. However, it is obvious that the neutron spectra from the 6x6, Table 5.2.14, bounds the thoria rod neutron spectra, Table 5.2.33, with a significant margin. In order to demonstrate that the gamma spectrum from the single thoria rod canister is bounded by the gamma spectrum from the design basis 6x6 fuel assembly, the gamma dose rate on the outer radial surface of the overpack was estimated conservatively assuming an MPC full of thoria rod canisters. This gamma dose rate was compared to an estimate of the dose rate from an MPC full of design basis 6x6 fuel assemblies. The gamma dose rate from the 6x6 fuel was higher than the dose rate from an MPC full of thoria rod canisters. This in conjunction with the significant margin in neutron spectrum and the fact that there is only one thoria rod canister clearly demonstrates that the thoria rod canister is acceptable for storage in the MPC-68 or the MPC-68F.

5.4.9 PWR Fuel Assembly with Non-zircaloy Grid Spacers

PWR fuel assemblies with non-zircaloy (inconel or steel) grid spacers are qualified to be loaded to MPC-32 basket. Since the mass of the spacers is significant and since the cobalt impurity level

[†] 6.0E+4 = 4.63E+6 / 77.25

assumed for inconel is relatively high, the Cobalt-60 activity from the incore spacers may contribute significantly to the external dose rate. As a result, separate burnup and cooling times were developed for PWR assemblies that utilize zircaloy and non-zircaloy incore spacers. Since steel has a lower cobalt impurity level than inconel, any zircaloy clad PWR assemblies with stainless steel grid spacers are bounded by the analysis performed in this chapter utilizing inconel grid spacers.

The description of the analyzed PWR non-zircaloy grid spacers is provided in Table 5.4.25, including the grid spacer mass and cobalt impurity level in the inconel. These values are very conservative.

The analyzed fuel assembly burnup and compositions, along with surface and 1-m dose rates, are provided in Table 5.4.24. These representative burnup-cooling time combinations bound the MPC-32 loading patterns provided in Table 2.1.15 for non-zircaloy grid spacers (–i.e., for the same cooling time, the burnup of the analyzed combinations bound those is Table 2.1.15).

Table 5.4.12

ANNUAL DOSE AT 300 METERS FROM A SINGLE CASK†
MPC-24 DESIGN BASIS ZIRCALOY CLAD FUEL
(The values in parentheses are in mSv/~~h~~yr)

	40,000 MWD/MTU 5-Year Cooling (mrem/yr) (mSv/~~h~~yr)	**47,500 MWD/MTU 8-Year Cooling (mrem/yr) (mSv/~~h~~yr)**
Fuel gammas††	8.15 (0.0815)	4.34 (0.0434)
^{60}Co Gammas	2.46 (0.0246)	1.88 (0.0188)
Neutrons	2.94 (0.0294)	4.74 (0.0474)
Total	13.55 (0.1355)	10.96 (0.1096)

† 100% occupancy (8760 hours) is assumed.

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

Table 5.4.13

DOSE VALUES USED IN CALCULATING ANNUAL DOSE FROM
MPC-24 DESIGN BASIS ZIRCALOY CLAD FUEL
VARIOUS ISFSI CONFIGURATIONS
40,000 MWD/MTU AND 5-YEAR COOLING†
(The values in parentheses are in mSv/~~h~~yr)

	A **Side of Overpack** **(mrem/yr)** **(mSv/~~h~~yr)**	**B** **Top of Overpack** **(mrem/yr)** **(mSv/~~h~~yr)**	**C** **Side of Shielded Overpack** **(mrem/yr)** **(mSv/~~h~~yr)**
100 meters	337.58 (3.3758)	7.40 (0.0740)	110.31 (1.1031)
150 meters	115.93 (1.1593)	3.07 (0.0307)	40.56 (0.4056)
200 meters	51.52 (0.5152)	1.35 (0.0135)	17.54 (0.1754)
250 meters	24.53 (0.2453)	0.63 (0.0063)	8.81 (0.0881)
300 meters	13.28 (0.1328)	0.27 (0.0027)	4.15 (0.0415)
350 meters	6.76 (0.0676)	0.15 (0.0015)	2.23 (0.0223)
400 meters	3.28 (0.0328)	0.09 (0.0009)	1.16 (0.0116)

† 100% occupancy (8760 hours) is assumed.

Table 5.4.18

DOSE RATES FROM FUEL GAMMAS
DOSE LOCATION ADJACENT TO OVERPACK
NORMAL CONDITIONS
MPC-32 DESIGN BASIS ZIRCALOY CLAD FUEL AT VARYING BURNUP AND COOLING TIMES†
(The values in parentheses are in mSv/h)

Dose Point†† Location	**40,000 MWD/MTU ~~5~~8-Year Cooling (mrem/hr) (mSv/h)**	**4~~7~~5,~~5~~000 MWD/MTU ~~8~~11-Year Cooling (mrem/hr) (mSv/h)**
1	5.59 (0.0559)	4.33 (0.0433)
2	49.16 (0.4916)	38.88 (0.3888)
3	1.95 (0.0195)	1.66 (0.0166)
4	0.97 (0.0097)	0.81 (0.0081)
5	0.25 (0.0025)	0.3 (0.003)
6 (dry MPC)†††	12.96 (0.1296)	9.65 (0.0965)
7 (no temp. shield)	64.04 (0.6404)	45.37 (0.4537)
7 (with temp. shield)	20.4 (0.204)	17.75 (0.1775)

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

†† Refer to Figure 5.1.1.

††† Overpack closure plate not present.

Table 5.4.19

DOSE RATES FROM ^{60}Co GAMMAS
DOSE LOCATION ADJACENT TO OVERPACK
NORMAL CONDITIONS
MPC-32 DESIGN BASIS ZIRCALOY CLAD FUEL AT VARYING BURNUP AND COOLING TIMES
(The values in parentheses are in mSv/h)

Dose Point† Location	**40,000 MWD/MTU ~~5~~8-Year Cooling (mrem/hr) (mSv/h)**	**4~~7~~5,~~5~~000 MWD/MTU ~~8~~11-Year Cooling (mrem/hr) (mSv/h)**
1	220.07 (2.2007)	159.87 (1.5987)
2	0.03 (0.0003)	0.02 (0.0002)
3	80.91 (0.8091)	58.78 (0.5878)
4	35.84 (0.3584)	26.04 (0.2604)
5	0.58 (0.0058)	0.42 (0.0042)
6 (dry MPC)††	258.78 (2.5878)	187.99 (1.8799)
7 (no temp. shield)	1476.27 (14.7627)	1072.43 (10.7243)
7 (with temp. shield)	323.69 (3.2369)	235.15 (2.3515)

† Refer to Figure 5.1.1.

†† Overpack closure plate not present.

Table 5.4.20

DOSE RATES FROM NEUTRONS
DOSE LOCATION ADJACENT TO OVERPACK
NORMAL CONDITIONS
MPC-32 DESIGN BASIS ZIRCALOY CLAD FUEL AT VARYING BURNUP AND COOLING TIMES
(The values in parentheses are in mSv/h)

Dose Point[†] Location	**40,000 MWD/MTU ~~5~~8-Year Cooling (mrem/hr) (mSv/h)**	**4~~7~~5,~~5~~000 MWD/MTU ~~8~~11-Year Cooling (mrem/hr) (mSv/h)**
1	84.52 (0.8452)	110.33 (1.1033)
2	23.37 (0.2337)	30.51 (0.3051)
3	67.31 (0.6731)	87.86 (0.8786)
4	39.15 (0.3915)	51.1 (0.511)
5	58.78 (0.5878)	76.73 (0.7673)
6 (dry MPC)[††]	141.89 (1.4189)	~~141.89~~185.22 (~~1.4189~~1.8522)
7 (no temp. shield)	438.45 (4.3845)	572.34 (5.7234)
7 (with temp. shield)	20.57 (0.2057)	26.85 (0.2685)

† Refer to Figure 5.1.1.

†† Overpack closure plate not included.

Table 5.4.21

TOTAL DOSE RATES
DOSE LOCATION ADJACENT TO OVERPACK
NORMAL CONDITIONS
MPC-32 DESIGN BASIS ZIRCALOY CLAD FUEL AT VARYING BURNUP AND COOLING TIMES
(The values in parentheses are in mSv/h)

Dose Point† Location	**40,000 MWD/MTU ~~5~~8-Year Cooling (mrem/hr) (mSv/h)**	**4~~7~~5,~~5~~000 MWD/MTU ~~8~~11-Year Cooling (mrem/hr) (mSv/h)**
1	310.18 (3.1018)	274.53 (2.7453)
2	72.56 (0.7256)	69.41 (0.6941)
3	150.16 (1.5016)	148.3 (1.483)
4	75.96 (0.7596)	77.95 (0.7795)
5	59.6 (0.596)	77.45 (0.7745)
6 (dry MPC)††	413.62 (4.1362)	382.85 (3.8285)
7 (no temp. shield)	1978.75 (19.7875)	1690.14 (16.9014)
7 (with temp. shield)	364.66 (3.6466)	279.74 (2.7974)

† Refer to Figure 5.1.1.

†† Overpack closure plate not included.

Table 5.4.22

ANNUAL DOSE AT 300 METERS FROM A SINGLE CASK†
MPC-32 DESIGN BASIS ZIRCALOY CLAD FUEL
(The values in parentheses are in mSv/yr)

	40,000 MWD/MTU 8-Year Cooling (mrem/yr) (mSv/yr)	**45,000 MWD/MTU 11-Year Cooling (mrem/yr) (mSv/yr)**
Fuel gammas††	3.83 (0.0383)	3.04 (0.0304)
^{60}Co Gammas	2.25 (0.0225)	1.63 (0.0163)
Neutrons	3.08 (0.0308)	4.03 (0.0403)
Total	9.16 (0.0916)	8.7 (0.087)

† 100% occupancy (8760 hours) is assumed.

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

Table 5.4.23

DOSE VALUES USED IN CALCULATING ANNUAL DOSE FROM
MPC-32 DESIGN BASIS ZIRCALOY CLAD FUEL
VARIOUS ISFSI CONFIGURATIONS
40,000 MWD/MTU AND 8-YEAR COOLING†
(The values in parentheses are in mSv/yr)

	A **Side of Overpack (mrem/yr)** **(mSv/yr)**	**B** **Top of Overpack (mrem/yr)** **(mSv/yr)**	**C** **Side of Shielded Overpack (mrem/yr)** **(mSv/yr)**
100 meters	234.04 (2.3404)	7.24 (0.0724)	82.67 (0.8267)
150 meters	81.85 (0.8185)	2.94 (0.0294)	30 (0.3)
200 meters	35.65 (0.3565)	1.31 (0.0131)	13.29 (0.1329)
250 meters	17.27 (0.1727)	0.63 (0.0063)	6.37 (0.0637)
300 meters	8.84 (0.0884)	0.32 (0.0032)	3.38 (0.0338)
350 meters	4.85 (0.0485)	0.17 (0.0017)	1.79 (0.0179)
400 meters	2.63 (0.0263)	0.07 (0.0007)	0.97 (0.0097)

† 100% occupancy (8760 hours) is assumed.

Table 5.4.24

COMPARISON OF TOTAL DOSE RATES FOR DESIGN BASIS PWR FUEL TO PWR FUEL REPRESENTATIVE BURNUP-COOLING TIME COMBINATIONS MPC-32 NORMAL AND ACCIDENT CONDITIONS (The values in parentheses are in mSv/h)

Dose Point[†] Location	**40 GWD/MTU 8 year cooling DESIGN BASIS (mrem/hr) (mSv/h)**	**25 GWD/MTU 8 year cooling (mrem/hr) (mSv/h)**	**45 GWD/MTU 20 year cooling (mrem/hr) (mSv/h)**	**25 GWD/MTU 12 year cooling (mrem/hr) (mSv/h)**	**35 GWD/MTU 16 year cooling (mrem/hr) (mSv/h)**	**43 GWD/MTU 20 year cooling (mrem/hr) (mSv/h)**
Non-zircaloy Grid Spacer ?	No	No	No	Yes	Yes	Yes
SURFACE - NORMAL CONDITION						
1	310.18 (3.1018)	186.77 (1.8677)	130.26 (1.3026)	117.58 (1.1758)	114.3 (1.143)	123.81 (1.2381)
2	72.56 (0.7256)	29.16 (0.2916)	40.8 (0.408)	35.37 (0.3537)	39.99 (0.3999)	46.67 (0.4667)
3	150.16 (1.5016)	76.85 (0.7685)	81.95 (0.8195)	50.26 (0.5026)	59.48 (0.5948)	76.7 (0.767)
4	75.96 (0.7596)	36.24 (0.3624)	45.08 (0.4508)	24.19 (0.2419)	30.93 (0.3093)	42.05 (0.4205)
5	59.6 (0.596)	13.97 (0.1397)	55.39 (0.5539)	12.03 (0.1203)	28.77 (0.2877)	50.84 (0.5084)
7 (no temp. shield)	1978.75 (19.7875)	1238.1 (12.381)	758.4 (7.584)	782.41 (7.8241)	719.89 (7.1989)	730.86 (7.3086)
SURFACE - ACCIDENT CONDITION						
2	1355.35 (13.5535)	341.13 (3.4113)	1201.76 (12.0176)	320.62 (3.2062)	667.1 (6.671)	1125.03 (11.2503)

† Refer to Figure 5.1.1.

Table 5.4.24 (continued)

COMPARISON OF TOTAL DOSE RATES FOR DESIGN BASIS PWR FUEL
TO PWR FUEL REPRESENTATIVE BURNUP-COOLING TIME COMBINATIONS
MPC-32 NORMAL AND ACCIDENT CONDITIONS
(The values in parentheses are in mSv/h)

Dose Point† Location	**40 GWD/MTU 8 year cooling DESIGN BASIS (mrem/hr) (mSv/h)**	**25 GWD/MTU 8 year cooling (mrem/hr) (mSv/h)**	**45 GWD/MTU 20 year cooling (mrem/hr) (mSv/h)**	**25 GWD/MTU 12 year cooling (mrem/hr) (mSv/h)**	**35 GWD/MTU 16 year cooling (mrem/hr) (mSv/h)**	**43 GWD/MTU 20 year cooling (mrem/hr) (mSv/h)**
Non-zircaloy Grid Spacer ?	No	No	No	Yes	Yes	Yes
ONE METER - NORMAL CONDITION						
1	38.14 (0.3814)	22.4 (0.2240)	15.94 (0.1594)	15.53 (0.1553)	15.03 (0.1503)	15.92 (0.1592)
2	30.67 (0.3067)	13.04 (0.1304)	16.12 (0.1612)	15.62 (0.1562)	16.92 (0.1692)	18.93 (0.1893)
3	25.75 (0.2575)	13.77 (0.1377)	12.58 (0.1258)	9.87 (0.0987)	10.58 (0.1058)	12.4 (0.124)
4	25.78 (0.2578)	14.02 (0.1402)	12.55 (0.1255)	9.58 (0.0958)	10.28 (0.1028)	12.13 (0.1213)
5	17.1 (0.171)	4.07 (0.0407)	15.8 (0.1580)	3.48 (0.0348)	8.24 (0.0824)	14.51 (0.1451)
7 (no temp. shield)	863.73 (8.6373)	576.93 (5.7693)	280.28 (2.8028)	360.43 (3.6043)	303.86 (3.0386)	274.44 (2.7444)
ONE METER - ACCIDENT CONDITION						
2	471.11 (4.7111)	122.5 (1.2250)	409.61 (4.0961)	116.96 (1.1696)	232.74 (2.3274)	385.95 (3.8595)

† Refer to Figure 5.1.1.

Table 5.4.25

DESCRIPTION OF ANALYZED MPC-32 PWR FUEL NON-ZIRCALOY GRID SPACER

Description	**Value**
Inconel incore Grid Spacers (kg)	4.9
Co-59 Impurity Level (g/kg)	4.7

c. Insert the nozzle of the helium supply into the vent port recess to displace the oxygen.

Note:
Helium gas is required to be injected into the port recesses to ensure that the leakage test is valid.

~~d. Deleted.~~

d. Weld the cover plate and perform NDE with approved procedures. (See 9.1 and Table 2.2.15)

Note:
NDE personnel shall be qualified per the requirements of Section V of the ASME Code [8.1.3] or a site-specific program.

~~e. Deleted.~~

~~f. Deleted.~~

~~g. Deleted.~~

~~h. Deleted..~~

~~i. Deleted..~~

~~j.~~e. Repeat Steps 36~~36~~~~30~~.~~a~~ a through 36~~36jj~~.~~j~~d for the drain port cover plate.

~~38.~~37. Perform a leakage test of the MPC vent and drain port cover plates as follows:

Note:
The leakage detector may detect residual helium in the atmosphere from the helium injection process. If the leakage tests detects a leak, the area should be blown clear with compressed air or nitrogen and the location should be retested.

a. Flush the area around the vent and drain cover plates with compressed air or nitrogen to remove any residual helium gas.

b. Perform a helium leakage rate test of vent and drain cover plate welds in accordance with the Mass Spectrometer Leak Detector (MSLD) manufacturer's instructions and ANSI N14.5 [8.1.2]. The MPC helium leakage rate test acceptance criteria are specified in ~~LCO 2.1.1~~ the Technical Specification.

c. Repair any weld defects in accordance with the site's approved code weld repair procedures. Reperform the leakage test as required.

~~39.~~38. Weld the MPC closure ring as follows:

8.6 REFERENCES

[8.0.1] U.S. Nuclear Regulatory Commission, "Standard Review Plan for Dry Cask Storage Systems", NUREG-1536, Final Report, January 1997.

[8.0.2] *U.S. Code of Federal Regulations,* Title 10, “Energy”, Part 72, "Licensing Requirements for Independent Storage of Spent Nuclear Fuel and High-Level Radioactive Waste."

[8.0.3] American National Standard for Special Lifting Devices for Shipping Containers Weighing 10 000 Pounds (4500 KG) or More for Nuclear Materials, ANSI N14.6, 1993.

[8.0.4] U.S. Nuclear Regulatory Commission, "Control of Heavy Loads at Nuclear Power Plants," NUREG-0612.

[8.0.5] Holtec International, “Final Safety Analysis Report for the HI-STORM 100 System”, Report HI-2002444, latest revision.

[8.0.6] Holtec International, “Safety Analysis Report on HI-STAR 100 Cask System”, Report HI-951251, latest revision.

[8.1.2] American National Standards Institute, Institute for Nuclear Materials Management, "American National Standard for Radioactive Materials – Leakage Tests on Packages for Shipment," ANSI N14.5-1997.

[8.1.3] American Society of Mechanical Engineers, “Boiler and Pressure Vessel Code”.

[8.2.1] Holtec International, "Topical Safety Analysis Report for the HI-STORM 100 System", Report HI-951312, latest revision.

[8.4.1] *U.S. Code of Federal Regulations,* Title 10, “Energy”, Part 20, “Standards for Protection Against Radiation.”

[8.4.2] *U.S. Code of Federal Regulations,* Title 49, “Transportation”, Part 173, “Shippers – General Requirements for Shipments and Packages.”

$$\text{Fuel Spill Area (A)} = \frac{\pi}{4}(D_o^2 - D_i^2)$$
$$= 105.3\ ft^2\ (9.78\ m^2)$$

$$\text{Spill Depth (d)} = \frac{V}{A} = \frac{6.68}{105.3}$$
$$= 0.0634\ ft\ (0.761\text{” or } 1.93\ cm)$$

Fuel Consumption Rate (R) = 0.15 inch/min (0.38 cm/min) [11.2.7]~~)~~

$$\text{Fire Duration} = \frac{d}{R} = \frac{0.761}{0.15}$$
$$= 5.075\ min\ (305\ seconds)$$

It is noted that this duration is calculated for a vertically-oriented cask. This bounds the duration for a horizontally-oriented cask as follows. This design-basis fire is defined using the guidance of 10CFR71.73, which is endorsed by NUREG-1536. As described therein, the fuel puddle must extend horizontally at least 1 m (40 in) beyond any external surface of the cask. For a vertical HI-STAR 100 the external surface of the cask is defined by a 93 3/4” circle. For a horizontal HI-STAR 100 the external surface of the cask is defined by a 93 3/4” x 191 1/8” rectangle. The fuel puddle formed around the much larger rectangular area will also be much larger, and therefore shallower, than that formed around the vertical cask. A shallower puddle results in a shorter fire, as the diameter of the cask is less than its length and so a vertically-oriented cask results in a significantly smaller fuel puddle, which maximizes the puddle depth and corresponding fire duration. As a result the temperature response of the cask is evaluated using the duration for a vertically-oriented cask.

Within this time period, the cask outside surface and its contents will undergo a transient temperature rise due to the heat absorbed from the fire. Full effects of insolation before, during, and after the fire are included in the HI-STAR 100 System transient analysis. During the postulated fire event, the neutron shield material is exposed to high temperatures. Therefore, conservatively, an upper bound material thermal conductivity is assumed during the fire to maximize heat input to the cask. During the post-fire cooldown phase, no credit is taken for conduction through the neutron shield. The temperature history of a number of critical points in the HI-STAR 100 System transient fire analysis are tracked during the fire and the subsequent relaxation of temperature profiles during the post-fire cooldown phase. The impact of transient temperature excursions on HI-STAR 100 System materials is assessed in this section. During the fire, a cask surface emissivity specified in 10CFR71.73(b)(4) is applied to maximize radiant heat input. Destruction of the paint covering the external cask surfaces due to exposure to intense heat during fire is a credible possibility. Therefore, a lower emissivity of the exposed carbon steel surface is conservatively applied for post-fire cooldown analysis. This approach provides a conservatively bounding response of the HI-STAR 100 System to the fire accident condition.