CHAPTER 4† THERMAL EVALUATION

4.0 OVERVIEW

The HI-STORM System is designed for long-term storage of spent nuclear fuel (SNF) in a vertical orientation. An array of HI-STORM Systems laid out in a rectilinear pattern will be stored on a concrete ISFSI pad in an open environment. In this section, compliance of the HI-STORM thermal performance to 10CFR72 requirements for outdoor storage at an ISFSI is established. Safe thermal performance during on-site loading, unloading and transfer operations utilizing the HI-TRAC transfer cask is also demonstrated. The analysis considers passive rejection of decay heat from the stored SNF assemblies to the environment under the most severe design basis ambient conditions. Effects of solar radiation (insolation) and partial radiation blockage due to the presence of neighboring casks at an ISFSI site are included in the analyses. Finally, the thermal margins of safety for long-term storage of both moderate burnup (up to 45,000 MWD/MTU) and high burnup spent nuclear fuel (greater than 45,000 MWD/MTU) in the HI-STORM 100 System are quantified.

The *HI-STORM thermal evaluation adopts* ~~guidelines presented in~~ NUREG-1536 [4.4.10] *and the ISG-11 [4.1.4] guidelines to demonstrate safe storage of Commercial Spent Fuel (CSF)*.* ~~include eight specific acceptance criteria that should be fulfilled by the cask thermal design~~. These ~~eight criteria are summarized here as follows~~ *guidelines are stated below*:

1. The fuel cladding temperature *for long term storage and short-term operations shall be limited to 400°C (752°F).* ~~at the beginning of dry cask storage should generally be below the anticipated damage threshold temperatures for normal conditions and a minimum of 20 years of cask storage.~~

2. The fuel cladding temperature should ~~generally~~ be maintained below ~~570°C~~ *570°C* (~~1058°F~~ *1058°F*) for accident~~,~~ *and* off-normal *event*~~, and fuel transfer~~ conditions.

3. The maximum internal pressure of the cask should remain within its design pressures for normal (1% rod rupture), off-normal (10% rod rupture), and accident (100% rod rupture) conditions.

---

† This chapter has been prepared in the format and section organization set forth in Regulatory Guide 3.61. However, the material content of this chapter also fulfills the requirements of NUREG-1536. Pagination and numbering of sections, figures, and tables are consistent with the convention set down in Chapter 1, Section 1.0, herein. Finally, all terms-of-art used in this chapter are consistent with the terminology of the glossary (Table 1.0.1) and component nomenclature of the Bill-of-Materials (Section 1.5).

* *Defined as nuclear fuel that is used to produce energy in a commercial nuclear reactor (See Table 1.0.1).*

4. The cask and fuel materials should be maintained within their minimum and maximum temperature criteria for normal, off-normal, and accident conditions.

5. For fuel assemblies proposed for storage, the cask system should ensure a very low probability of cladding breach during long-term storage.

~~6. Fuel cladding damage resulting from creep cavitation should be limited to 15% of the original cladding cross sectional area.~~

~~7.~~ *6.* The ~~cask~~ *HI-STORM* system should be passively cooled.

~~8.~~ *7.* The thermal performance of the cask should be within the allowable design criteria specified in FSAR Chapters 2 and 3 for normal, off-normal, and accident conditions.

As demonstrated in this chapter (see Subsections 4.4.6 and 4.5.6), the HI-STORM System is designed to comply with all ~~eight~~ *of the* criteria listed above. All thermal analyses to evaluate normal conditions of storage in a HI-STORM storage module are described in Section 4.4. All thermal analyses to evaluate normal handling and on-site transfer in a HI-TRAC transfer cask are described in Section 4.5. All analyses for off-normal conditions are described in Section 11.1. All analyses for accident conditions are described in Section 11.2. Sections 4.1 through 4.3 describe thermal analyses and input data that are common to all conditions. This FSAR chapter is in full compliance with NUREG-1536 requirements, subject to the exceptions and clarifications discussed in Chapter 1, Table 1.0.3 *and to ISG-11 requirements (no exceptions).*

~~—~~This revision to the HI-STORM *Final* Safety Analysis Report~~, the first since the HI-STORM 100 System was issued a Part 72 Certificate of Compliance,~~ incorporates several features into the thermal analysis to respond to the changing needs of the U.S. nuclear power generation industry *and revisions to NRC regulations*. The most significant changes are:

- *The thermal analysis is revised to comply with recently issued NRC staff guidance ("Cladding Considerations for the Transportation and Storage of Spent Fuel", ISG 11, Rev. 2).*

- *The Aluminum Heat Conduction Elements (AHCEs), optional under Amendment 1 of CoC 1014, are removed from the design. Removing the AHCEs from the MPC eliminates the constriction of the downcomer flow (Figure 4.0.1) and thus further enhances the thermal performance of the MPC.*

- *The whole spectrum of regionalized storage of Spent Nuclear Fuel (SNF) for each MPC type has been analyzed to allow flexibility in Region 1 (core region of the basket) and Region 2 (the outer*

region) heat loads. The heat loads flexibility afforded to the ISFSI owner by the analyses documented in this FSAR permits MPCs to be loaded in the most effective manner to minimize the aggregate dose emitted from the totality of the casks arrayed on the pad.

- *Certain elements of excessive conservatism in the mathematical model have been relaxed to retain a moderate level of conservatism. Subsection 4.4.6 documents conservatisms that apply to the thermal solution. A quantitative estimate of the consequences of the elements of conservatism is provided in Appendix 4.B.*

- *The helium backfill pressure is increased (Table 4.4.38) to facilitate increased heat dissipation from the MPC through the classical thermosiphon action (Figure 4.0.1).*

- *The design maximum decay heat load (Q_d) for the HI-STORM System is revised (Table 4.4.39).*

~~- Post-core decay time (PCDT) limitations on high burnup fuel (burnup > 45,000 MWD/MTU) have been computed. The allowable cladding temperatures for high burnup PWR and BWR fuel, required to establish PCDT limits, are computed using a methodology consistent with ISG-11.~~

~~- Both uniform and regionalized storage are permitted, the latter being particularly valuable in mitigating the dose emitted by the MPC by restricting "cold and old" SNF in the locations surrounding the core region of the basket (where the "hot and new" fuel is stored).~~

~~- The effect of convective heat transfer in the MPC, originally included in the analysis but subsequently neglected to enable the NRC to make a more considered assessment of gravity-driven convective heat transfer in honeycomb basket equipped MPCs, is now reintroduced.~~

~~- In the absence of the credit for convective (thermosiphon) effect, the previous analysis relied on the conduction heat transfer through the clearance between the basket and the MPC enclosure vessel. The conduction heat flow path was provided by the Aluminum Heat Conduction Elements (AHCE). The AHCE hardware is retained in the MPC and credit for ACHE heat dissipation is eliminated in the thermal analyses to maintain a solid margin of conservatism in the computed results. In a similar spirit of conservatism, the heat transfer in narrow cavities (the Rayleigh effect), approved by the SFPO in the previous analysis, is neglected in this revision.~~

~~Aside from the above mentioned changes, this revision of this chapter is essentially identical to its predecessor~~

In this chapter, the maximum HI-STORM System temperatures and pressures for normal conditions of storage are established. The normal storage conditions are defined below:

Condition	*Value*
Decay Heat	*Values Specified in Table 4.4.39*
MPC Helium Fill Pressure	*Minimum Specified in Table 4.4.38*
Ambient Temperature	*Annual Average Temperature (Table 2.2.2)*

The HI-STORM maximum temperatures are provided in Tables 4.4.9, 4.4.10 and 4.4.26 for MPC-24/24E, MPC-68 and MPC-32 respectively and in Table 4.4.36 for the overpack temperatures. The maximum MPC pressures are provided in Table 4.4.14.

4.0.1 HI-STORM Modeling Overview

The HI-STORM 100 System is a vertical ventilated overpack having an internal cavity for emplacement of a stainless steel Multi-Purpose Canister (MPC) containing Spent Nuclear Fuel (SNF). Prior to its emplacement, the MPC internal space is pressurized with helium. The HI-STORM Overpack is equipped with four large ducts at each of its bottom and top extremities. The design of the system provides for an annular gap between the MPC and overpack. The ducted overpack construction, together with an engineered annular space between the MPC and the HI-STORM overpack enables cooling of the MPC external surfaces by ventilation action. The ventilation cooling of the HI-STORM system is illustrated in Figure 4.0.2. As shown in this figure, ambient air is drawn into the annulus from the bottom inlet ducts. The air is heated during its upward movement through the vertical annulus and exits through the top outlet ducts.

In Figure 4.0.1, a cutaway view of an MPC is shown. The fuel basket that resides in the MPC is constructed as an array of square shaped fuel cells in a honeycomb structure for storing SNF. The fuel basket (and it's stored fuel) is enclosed in an all welded pressure vessel formed by the MPC baseplate, top lid and cylindrical shell. The MPC interior space is filled with pressurized helium. The MPC design incorporates top and bottom plenums formed by elongated semi-circular holes at the base and top of fuel cell walls. Between the fuel basket and the MPC shell is an open downcomer space engineered to connect the top and bottom plenums. In this manner, the MPCs feature a fully connected helium space consisting of the fuel basket cells, top and bottom plenums and a peripheral downcomer gap.

It is heuristically apparent from the geometry of the MPC that the fuel basket metal, the fuel assemblies and the contained helium will be at their peak temperature at or near the longitudinal axis of the MPC. As a result of conduction along the metal fuel basket walls and radiant heat exchange from the fuel assemblies to the MPC fuel basket, the temperatures will attenuate with increasing radial distance from the center, reaching their lowest values in the downcomer space. As a result, the bulk temperatures of the helium columns in the fuel basket are elevated above the bulk temperature of the downcomer space. Since two fluid columns with different temperatures in communicative contact cannot remain in static equilibrium, the non-isotropic temperature field guarantees the incipience of heat transfer by internal convection. This mode of MPC heat dissipation, "thermosiphon action," is depicted in Figure 4.0.1.

The thermosiphon action is initiated by the heating of helium residing in the fuel storage cells by the stored nuclear fuel. The heated helium expands, moving upward through the stored fuel assemblies due to buoyancy forces. The upward moving helium exits into the top plenum space turning 90° as shown in Figure 4.0.1. The helium flows through the top plenum towards the MPC shell, turns 90° and flows down in the downcomer space. In this space, the helium is progressively cooled as it flows down, rejecting its heat to the MPC shell. At the bottom of the downcomer, the helium turns 90° into the bottom plenum space to supply the fuel basket cells with cooled helium. In this manner a circulation of helium is sustained inside the MPC by the heat from the stored fuel.

From the discussion above, it is apparent that the thermal model of the HI-STORM system must include the following elements:

i) HI-STORM overpack
ii) MPC with it's enclosed fuel basket
iii) Air-flow in the HI-STORM annular space
iv) Helium circulation inside the MPC

A thermal model of the HI-STORM system incorporating the elements listed above is shown in Figure 4.0.3. In addition, as we explain next, the model recognizes the "coupled" nature of the heat transfer process in the HI-STORM system wherein the internal circulation of helium inside the MPC occurs with the upward flow of air outside of it. The MPC shell, also shown in this figure, is the interface boundary that separates the two fluids – helium and air. At this interface boundary air is moving upward on the outside of the MPC shell and helium is moving downward. Heat is exchanged from hot fluid (helium) to cold fluid (air) across a metal boundary, which is similar to that of a counter-flow heat exchanger. The heat exchange progressively lowers the helium temperature in the down-flow direction and progressively elevates that of air in the up-flow direction. It is apparent that the internal circulation of helium influences the heat input profile to the annulus air. The helium circulation removes heat from the fuel cells and rejects a portion of it to the upper elevations of the MPC shell and to the MPC lid. As a result, heat input to the air is skewed towards the upper portion of the MPC shell. Based on classical chimney operating principles, adding more heat towards the top of an air stack is less efficient for air circulation relative to adding heat at lower elevations.

In order to properly simulate the interaction of the helium circulation and the annulus air flow, a "coupled fluid" thermal model is constructed for the HI-STORM thermal analyses. The coupled fluid thermal model includes both fluids (helium and air) in one unified model. The model is constructed on FLUENT version 6.1 computer code from FLUENT Inc., a software company based in Lebanon, NH. The modeling methodology is described in Subsections 4.4.1.1.2 through 4.4.1.1.9. The results of the analysis are presented in Subsection 4.4.2 and evaluated in Subsection 4.4.6.

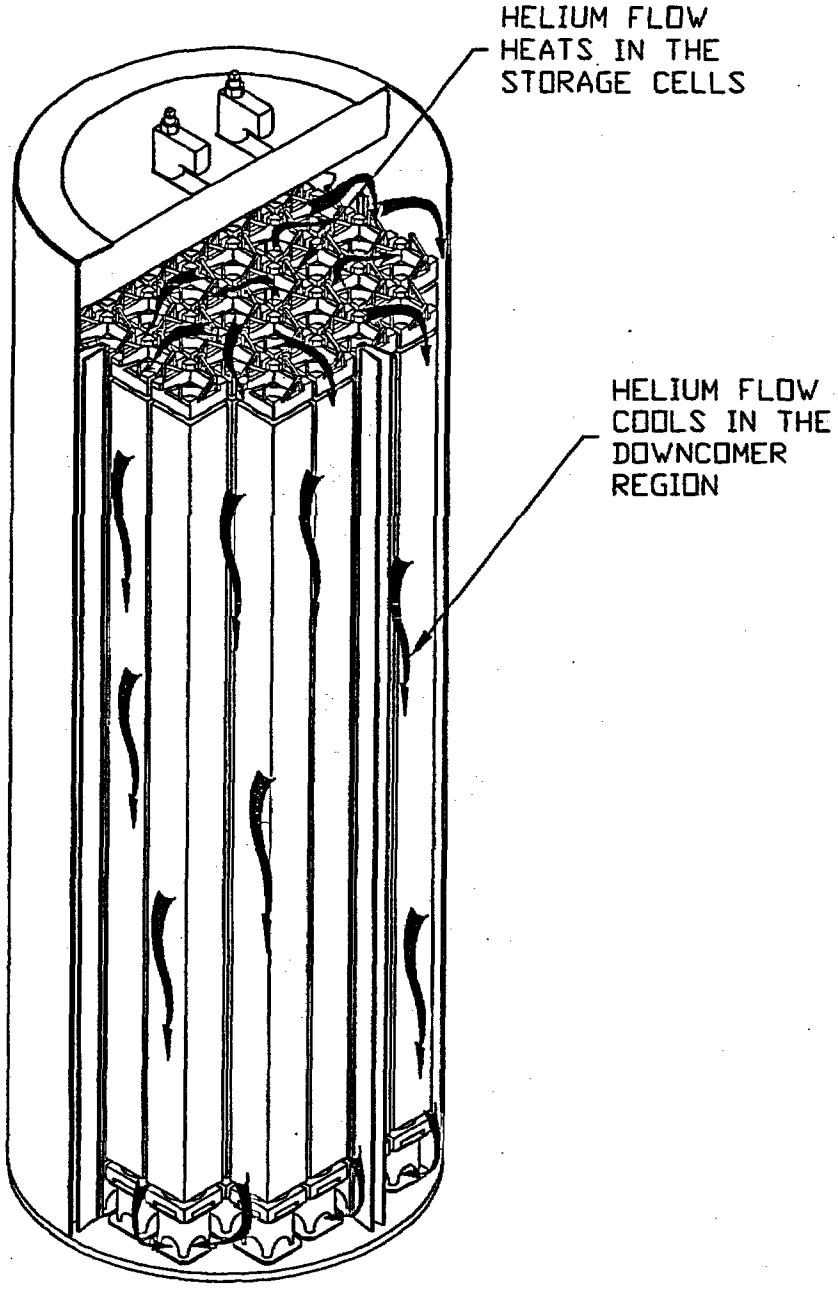


FIGURE 4.0.1: MPC INTERNAL HELIUM CIRCULATION

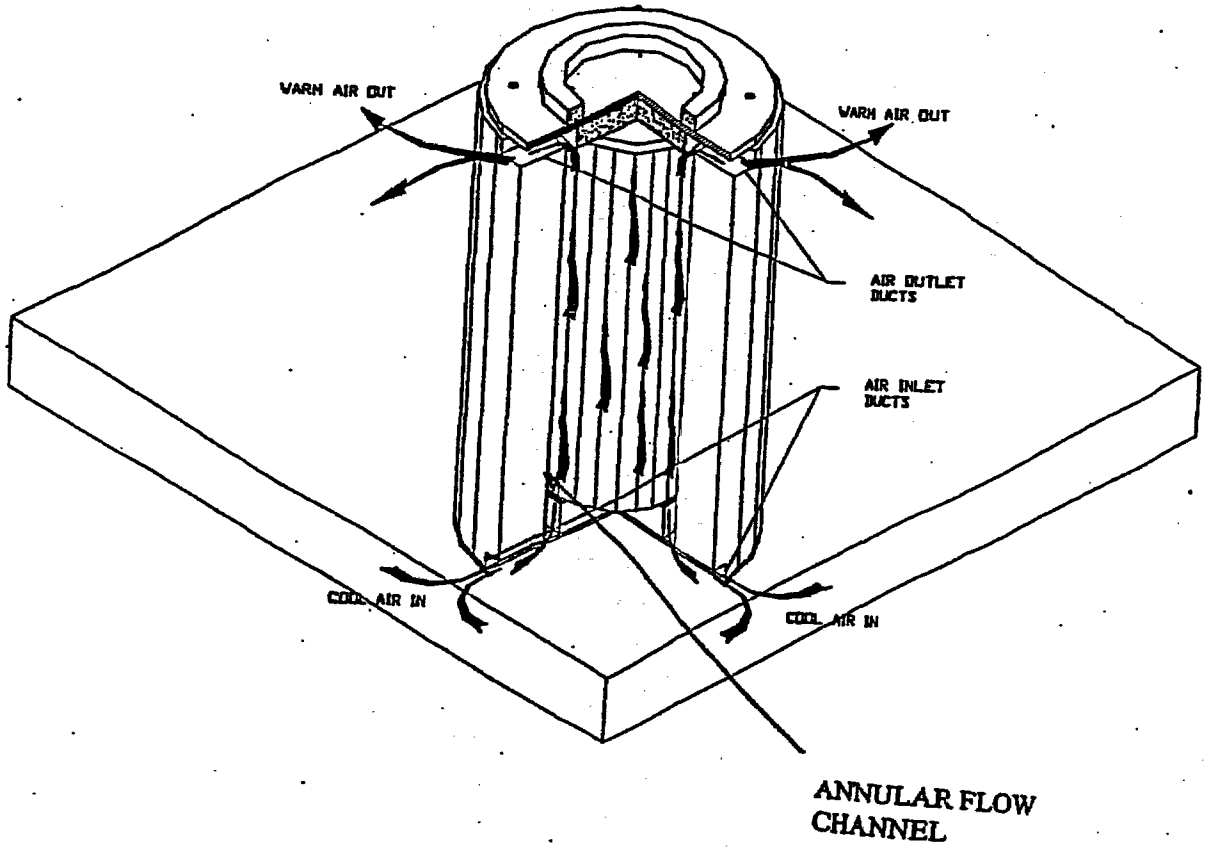


FIGURE 4.0.2: VENTILATION COOLING OF A HI-STORM SYSTEM

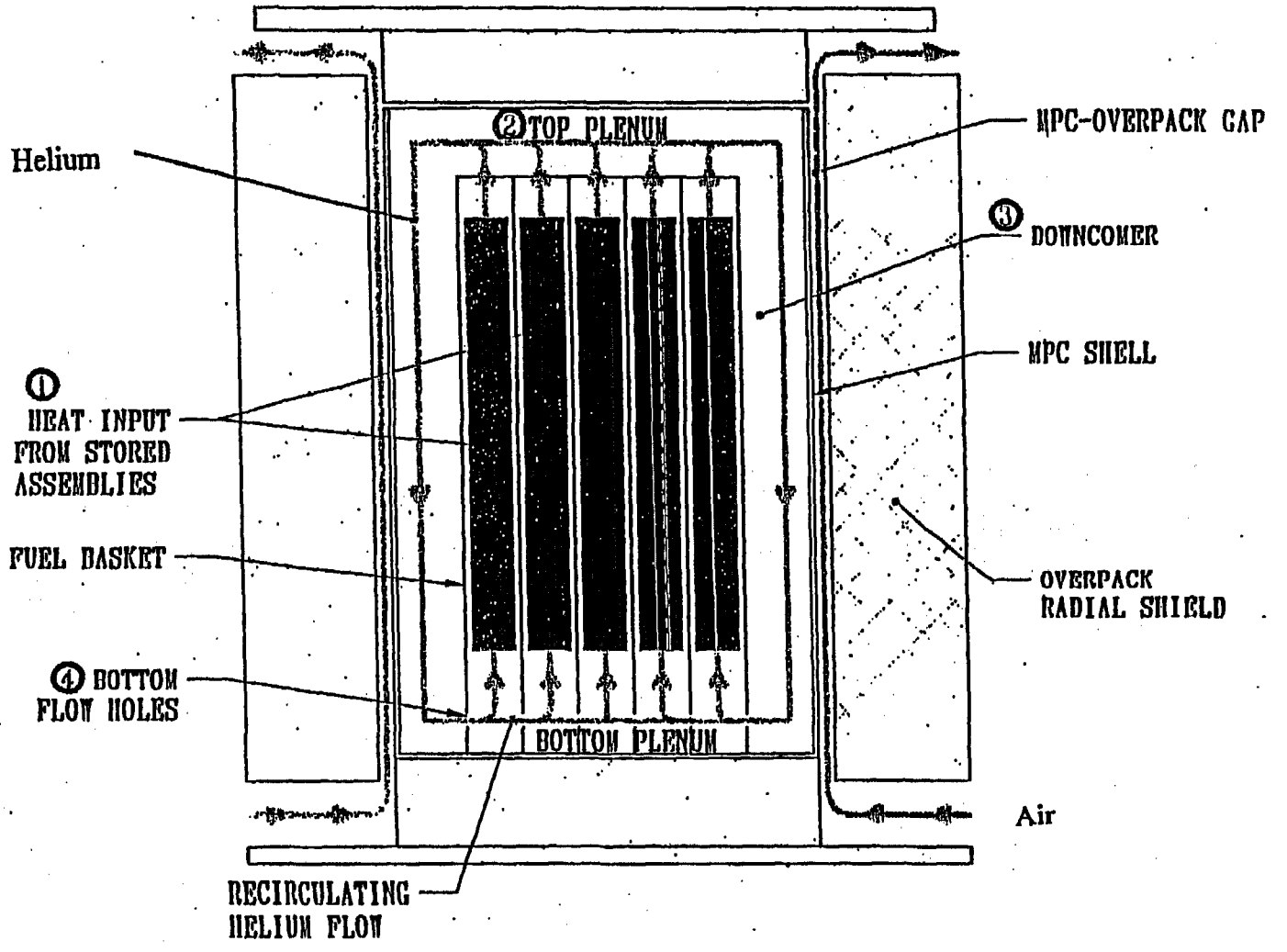


FIGURE 4.0.3: COUPLED FLUID MODEL OF THE HI-STORM SYSTEM

4.1 DISCUSSION

As discussed in Chapter 2, this revision of the HI-STORM FSAR seeks to establish complete compliance with the provisions of Reference [4.1.4]. To ensure explicit compliance, the new condition "short term operations," corresponding to fuel loading activities, is defined in Chapter 2.

In Revision 1 of this FSAR, fuel loading, which includes MPC cavity drying, MPC lid welding, helium pressurization, and MPC transfer operations, was treated as part of the "off-normal" condition. It is now treated as a distinct fuel thermal state. Specifically, the maximum fuel cladding temperature for the fuel loading condition now formally referred to as "short term operations" is set equal to the PCT limit for normal storage conditions for all CSF. Potential thermally challenging states for the spent fuel arise if the fuel drying process utilizes pressure reduction (i.e., vacuum drying), or when the loaded MPC is inside the HI-TRAC transfer cask. In the latter state, the rate of heat rejection from the MPC is somewhat less compared to the normal storage condition when the MPC is inside the ventilated overpack. Because the HI-TRAC transfer cask handling subsequent to helium pressurization of the MPC typically involves keeping the equipment vertical, the thermosiphon action inside the MPC is fully operational during these activities. As a result, the increase in the fuel cladding temperature in the HI-TRAC compared to the HI-STORM storage condition is fairly modest. The increase is more significant in the case where the HI-TRAC transfer cask, for reasons such as vertical height restrictions or seismic constraints at a plant, must be handled in the horizontal orientation. When the HI-TRAC is horizontal, the cessation of the thermosiphon action results in an additional rise in the fuel cladding temperature. Therefore, the short term evolutions that may be thermally limiting are analyzed as listed below:

i. Vacuum Drying
ii. Loaded MPC in HI-TRAC in the vertical orientation
iii. Loaded MPC in HI-TRAC in the horizontal orientation

The threshold MPC heat generation rate at which the HI-STORM peak cladding temperature reaches a steady state equilibrium value approaching the normal storage peak clad temperature limit is computed in this chapter. Likewise, the MPC heat generation rates that produce the steady state equilibrium temperature approaching the normal storage peak clad temperature limit for the MPC-in-HI-TRAC condition in both vertical and horizontal configurations are computed in this chapter. These computed heat generation rates directly bear upon the compliance of the system with Reference [4.1.4], and are accordingly adopted in the system Technical Specifications for high burnup fuel (HBF).

For moderate burnup fuel (MBF), it is reasonable to provide additional latitude during short term operations in light of the relative paucity of hydrides in the fuel cladding at low and moderate burnup levels reported in the literature. Accordingly, it is proposed that the permissible MPC heat generation rate be allowed to be governed by the more restrictive of the following two criteria (for

MBF only):

a. *The maximum estimated cladding hoop stress (σ_{max}) during short term operations does not exceed 90 MPa.*

b. *The maximum computed cladding temperature during short term operations does not exceed 570°C.*

For MBF that does not muster compliance with the aforementioned σ_{max} criterion (a), the Reference [4.1.4] temperature limit shall apply. Because the cladding stress is a function of the cladding thickness (and hence a function of the extent of cladding corrosion), the estimation of the cladding stress, of necessity, must be fuel-specific. The user of the HI-STORM system can exceed the threshold heat generation rates computed in this chapter (for MBF only) if fuel specific analysis using the methodology presented in this FSAR is performed to establish compliance with the two foregoing criteria. Section 4.5 contains a detailed treatment of all short term operations for both HBF and MBF.

A ~~sectional~~ *cutaway* view of the HI-STORM dry storage system has been presented ~~earlier~~ *in Figure 4.0.2.* ~~(see Figure 1.2.1).~~ The system consists of a sealed MPC situated inside a vertical ventilated storage overpack. ~~Air inlet and outlet ducts that allow for air cooling of the stored MPC are located at the bottom and top, respectively, of the cylindrical overpack. The SNF assemblies reside inside the MPC, which is sealed with a welded lid to form the confinement boundary.~~ The MPC contains an all-alloy honeycomb basket structure with square-shaped compartments of appropriate dimensions to allow insertion of the fuel assemblies prior to welding of the MPC lid and closure ring. Each box panel, with the exception of exterior panels on the MPC-68 and MPC-32, is equipped with a ~~Boral (~~thermal neutron absorber~~)~~ panel sandwiched between an ~~alloy~~ *Alloy X* steel sheathing plate and the box panel, along the entire length of the active fuel region. The MPC is backfilled with helium up to the design-basis initial fill level (Table *4.4.38*~~1.2.2~~). This provides a stable, inert environment for long-term storage of the SNF. Heat is rejected from the SNF in the HI-STORM System to the environment by passive heat transport mechanisms only.

The helium backfill gas is an integral part of the MPC thermal design. The helium fills all the spaces between solid components and provides an improved conduction medium (compared to air) for dissipating decay heat in the MPC. ~~Additionally, helium in the spaces between the fuel basket and the MPC shell is heated differentially and, therefore, subject to the "Rayleigh" effect which is discussed in detail later. For added conservatism, the increase in the heat transfer rate due to the Rayleigh effect contribution is neglected in this revision of the FSAR.~~ To ensure that the helium gas is retained and is not diluted by lower conductivity air, the MPC confinement boundary is designed and fabricated to comply with the provisions of the ASME B&PV Code Section III, Subsection NB (to the maximum extent practical), as an all-seal-welded pressure vessel with redundant closures. It is demonstrated in Section 11.1.3 that the failure of one field-welded pressure boundary seal will not result in a breach of the pressure boundary. The helium gas is therefore retained and undiluted, and

may be credited in the thermal analyses.

An important thermal design criterion imposed on the HI-STORM System is to limit the maximum fuel cladding temperature to within design basis limits (Table *4.3.1* ~~4.3.7~~) for long-term storage of design basis SNF assemblies. An equally important design criterion is to minimize temperature gradients in the MPC so as to minimize thermal stresses. In order to meet these design objectives, the MPC baskets are designed to possess certain distinctive characteristics, which are summarized in the following.

The MPC design minimizes resistance to heat transfer within the basket and basket periphery regions. This is ensured by an uninterrupted panel-to-panel connectivity realized in the all-welded honeycomb basket structure. The MPC design incorporates top and bottom plenums with interconnected downcomer paths. The top plenum is formed by the gap between the bottom of the MPC lid and the top of the honeycomb fuel basket, and by elongated semicircular holes in each basket cell wall. The bottom plenum is formed by large elongated semicircular holes at the base of all cell walls. The MPC basket is designed to eliminate structural discontinuities (i.e., gaps) which introduce large thermal resistances to heat flow. Consequently, temperature gradients are minimized in the design, which results in lower thermal stresses within the basket. Low thermal stresses are also ensured by an MPC design that permits unrestrained axial and radial growth of the basket. The possibility of stresses due to restraint on basket periphery thermal growth is eliminated by providing adequate basket-to-canister shell gaps to allow for basket thermal growth during heat-up to design basis temperatures.

~~It is heuristically apparent from the geometry of the MPC that the basket metal, the fuel assemblies, and the contained helium mass will be at their peak temperatures at or near the longitudinal axis of the MPC. The temperatures will attenuate with increasing radial distance from this axis, reaching their lowest values at the outer surface of the MPC shell. Conduction along the metal walls and radiant heat exchange from the fuel assemblies to the MPC metal mass would therefore result in substantial differences in the bulk temperatures of helium columns in different fuel storage cells. Since two fluid columns at different temperatures in communicative contact cannot remain in static equilibrium, the non-isotropic temperature field in the MPC internal space due to conduction and radiation heat transfer mechanisms guarantee the incipience of the third mode of heat transfer: natural convection.~~

~~The preceding paragraph~~ *Section 4.0.1 describes* ~~introduced~~ the internal helium thermosiphon feature engineered into the MPC design. It is recognized that the backfill helium pressure, in combination with low pressure drop circulation passages in the MPC design, induces a thermosiphon upflow through the multi-cellular basket structure to aid in removing the decay heat from the stored fuel assemblies. The decay heat absorbed by the helium during upflow through the basket is rejected to the MPC shell during the subsequent downflow of helium in the peripheral downcomers. This helium thermosiphon heat extraction process significantly reduces the burden on the MPC metal basket structure for heat transport by conduction, thereby minimizing internal basket temperature

gradients and resulting thermal stresses.

The helium columns traverse the vertical storage cavity spaces, redistributing heat within the MPC. Elongated holes in the bottom of the cell walls, liberal flow space and elongated holes at the top, and wide-open downcomers along the outer periphery of the basket ensure a smooth helium flow regime. The most conspicuous beneficial effect of the helium thermosiphon circulation, as discussed above, is the mitigation of internal thermal stresses in the MPC. Another beneficial effect is reduction of the peak fuel cladding temperatures of the fuel assemblies located in the interior of the basket. ~~In the original HI-STORM licensing analyses, no credit for the thermosiphon action was taken. To partially compensate for the reduction in the computed heat rejection capability due to the complete neglect of the global thermosiphon action within the MPC, heat conduction elements made of aluminum were interposed in the large peripheral spaces between the MPC shell and the fuel basket. These heat conduction elements, shown in the MPC Drawings in Section 1.5, are engineered such that they can be installed in the peripheral spaces to create a nonstructural thermal connection between the basket and the MPC shell. In their installed condition, the heat conduction elements will contact the MPC shell and the basket walls. MPC manufacturing procedures have been established to ensure that the thermal design objectives for the conduction elements set forth in this document are realized in the actual hardware. The presence of heat conduction elements in the canister design has been conservatively neglected in the thermal models of the HI-STORM 100 System in this revision of the Safety Analysis Report.~~

~~Four~~ *Three* distinct MPC basket geometries are evaluated for thermal performance in the HI-STORM System. For intact PWR fuel storage, the *24-cell flux trap (*MPC-24*/24E/24EF)*, ~~MPC-24E,~~ and *32-cell non-flux trap (*MPC-32*)* designs are available. Four locations are designated for storing damaged PWR fuel in the MPC-24E design. A 68-cell MPC design (MPC-68, MPC-68F, and MPC-68FF) is available for storing BWR fuel (intact or damaged, including fuel debris). All of the *three* ~~four~~ basic MPC geometries (MPC-32, MPC-24*/24E/24EF* ~~, MPC-24E~~ and MPC-68) are described in Chapter 1 wherein their ~~design~~ *licensing* drawings can also be found.

The design maximum decay heat loads for storage of intact zircaloy clad fuel in the four MPCs are ~~listed in Tables 4.4.20, 4.4.21, 4.4.28, and 4.4.29~~ *listed in Table 4.4.39 for single region storage (also referred to as uniform storage)*. Storage of intact stainless steel is *permitted for a low decay heat limit set forth in Chapter 2 (Tables 2.1.17 through 2.1.21)* ~~evaluated in Subsection 4.3.2.~~. Storage of zircaloy clad fuel with stainless steel clad fuel in an MPC is permitted. In this scenario, the zircaloy clad fuel ~~is conservatively stipulated to~~ *must* meet the lower decay heat limits for stainless steel clad fuel. Storage of damaged, zircaloy clad fuel is evaluated in Subsection 4.4.1.1.4. The axial heat distribution in each fuel assembly is assumed to follow the burnup profiles set forth by Table 2.1.11.

Thermal analysis of the HI-STORM System is based on including all three ~~fundamental~~ modes of heat transfer, namely conduction, natural convection and radiation. ~~Different combinations of these modes are active in different parts of the system.~~ These modes are properly identified and

conservatively analyzed within each part of the MPC, the HI-STORM storage overpack and the HI-TRAC transfer cask, to enable bounding calculations of the temperature distribution within the HI-STORM System to be performed. In addition to storage within the HI-STORM overpack, loaded MPCs will also be located for short durations inside the transfer cask (HI-TRAC) designed for moving MPCs into and out of HI-STORM storage modules.

Heat is dissipated from the outer surfaces of the *HI-STORM* storage overpack and HI-TRAC *transfer cask* to the environment by buoyancy induced airflow (natural convection) and thermal radiation. Heat transport through the cylindrical wall of the storage overpack and HI-TRAC is solely by conduction. While stored in a HI-STORM overpack, heat is rejected from the surface of the MPC via the parallel action of thermal radiation to the inner shell of the overpack and convection to a buoyancy driven airflow in the annular space between the outer surface of the MPC and the inner shell of the overpack. ~~This situation is similar to the familiar case of natural draft flow in furnace stacks.~~ When placed into a HI-TRAC cask for transfer operations, heat is rejected from the surface of the MPC to the inner shell of the HI-TRAC by conduction and thermal radiation.

Within the MPC, heat is transferred between metal surfaces (e.g., between neighboring fuel rod surfaces) via a combination of conduction through a gaseous medium (helium) and thermal radiation. Heat is transferred between the fuel basket and the MPC shell by thermal radiation and conduction. ~~The heat transfer between the fuel basket external surface and the MPC shell inner surface is further influenced by the "Rayleigh" effect. The heat transfer augmentation effect of this mechanism, as discussed earlier, is conservatively neglected.~~

As discussed *in Subsection 4.4.6 and Appendix 4.B,* ~~later in this chapter,~~ an array of conservative assumptions bias the results of the thermal analysis towards ~~much~~ reduced computed margins than would be obtained by a *more* rigorous analysis of the problem. ~~In particular, the thermal model employed in determining the MPC temperatures is consistent with the model presented in Rev. 9 of the HI-STAR FSAR submittal (Docket No. 72-1008).~~

~~As discussed in Chapter 2, the HI-STORM MPCs are identical to those utilized in the NRC-accepted HI-STAR System (Docket 71-1008 for storage). As such, many of the analysis methods utilized herein for performing thermal evaluations of the HI-STORM MPCs are identical to those already accepted for the HI-STAR System. Specifically, the analysis methods for evaluation of the following items are identical to those for the HI-STAR System:~~

~~i. fuel assembly effective thermal conductivity~~
~~ii. MPC fuel basket effective thermal conductivity~~
~~iii. MPC fuel basket peripheral region effective thermal conductivity~~
~~iv. aluminum heat conduction elements effective thermal conductivity~~
~~v. MPC internal cavity free volume~~
~~vi. MPC contents effective heat capacity and density~~
~~vii. bounding fuel rod internal pressures and hoop stresses~~

~~In addition, thermal properties for all materials common to both the HI-STORM and HI-STAR systems are identical, including stainless and carbon steels, zircaloy, UO_2, aluminum alloy 1100, Boral, Holtite-A, helium, air and paint.~~

The complete thermal analysis is performed using the industry standard ANSYS finite element modeling package [4.1.1] and the finite volume Computational Fluid Dynamics (CFD) code FLUENT [4.1.2]. ANSYS has been previously used and accepted by the NRC on numerous dockets [4.4.10 ,4.V.5.a]. The FLUENT CFD program is independently benchmarked and validated with a wide class of theoretical and experimental studies reported in the technical journals. Additionally, Holtec has confirmed the code's capability to reliably predict temperature fields in dry storage applications *in a benchmarking report [4.1.5]* using independent full-scale test data from a loaded cask [4.1.3]. A series of Holtec topical reports, culminating in *the Holtec Report:* "Topical report on the HI-STAR/HI-STORM ~~t~~ *T*hermal ~~m~~ *M*odel and its ~~b~~ *B*enchmarking with ~~f~~ *F*ull-*Size* ~~size~~ ~~c~~ *C*ask ~~t~~ *T*est ~~d~~ *D*ata" *[4.1.5]*, ~~Holtec Report HI-992252, Rev. 1,~~ documents the comparison of the Holtec thermal model against the ~~full-size~~ *test data obtained from a* cask *loaded with irradiated fuel from Surry nuclear reactor* ~~test data~~ [4.1.3]. In *this benchmarking report,* ~~rReference [4.1.3],~~ the Holtec thermal model is shown to overpredict the measured fuel cladding temperature by a modest amount for every test set. In early 2000, PNL evaluated the thermal performance of HI-STORM 100 at discrete ambient temperatures using the COBRA-SFS Code. (Summary report communicated by T.E. Michener to J. Guttman (NRC staff) dated May 31, 2000 titled "TEMPEST Analysis of the Utah ISFSI Private Fuel Storage Facility and COBRA-SFS Analysis of the Holtec HI-STORM 100 Storage System"). The above-mentioned ~~topical~~ *benchmarking* report ~~has been updated to~~ includes a comparison of the Holtec thermal model results with the PNL solution. ~~Once again,~~ *which shows that* the Holtec thermal model ~~is~~ *continues to be* uniformly conservative. ~~, albeit by small margins~~. The benchmarking of the Holtec thermal model *[4.1.5]* against the EPRI test data [4.1.3] and PNL COBRA-SFS study validate the suitability of the thermal model employed to evaluate the thermal performance of the HI-STORM 100 System in this document.

4.2 SUMMARY OF THERMAL PROPERTIES OF MATERIALS

Materials† present in the MPCs include stainless steels (Alloy X), ~~Boral~~ neutron absorber *(Boral or METAMIC)*, ~~aluminum Alloy 1100 heat conduction elements,~~ and helium. Materials present in the HI-STORM storage overpack include carbon steels and concrete. Materials present in the HI-TRAC transfer cask include carbon steels, lead, Holtite-A neutron shield, and demineralized water‡. In Table 4.2.1, a summary of references used to obtain cask material properties for performing all thermal analyses is presented.

Individual thermal conductivities of the alloys that comprise the Alloy X materials and the bounding Alloy X thermal conductivity are reported in Appendix 1.A of this report. Tables 4.2.2~~,~~ *and* 4.2.3 ~~and 4.2.9~~ provide numerical thermal conductivity data of materials at several representative temperatures. Thermal conductivity data for *constituents of* Boral ~~components~~ (i.e., B_4C core and aluminum cladding) is provided in Table 4.2.8. *Boral is a compressed neutron absorbing core clad with a thin layer of aluminum on both sides. Because of its sandwich construction, its conduction properties are directionally dependent (i.e. non-isotropic). In contrast to Boral, METAMIC is a homogeneous neutron absorbing material with a thermal conductivity that is higher than the Boral neutron absorbing B_4C core (Figure 4.2.3) but lower than Boral's aluminum cladding. The equivalent conductivity of a Boral panel, defined as the Square Root of the Mean Sum of Squares (SRMSS) conductivity in two principal directions (through thickness and width) is closely matched by METAMIC§. Therefore, the two materials are considered equivalent in their heat transfer performance.* ~~The temperature dependence of the thermal conductivities of helium and air is shown in Figure 4.2.1~~.

For the HI-STORM overpack, the thermal conductivity of concrete and the emissivity/absorptivity of painted surfaces are particularly important. Recognizing the considerable variations in reported values for these properties, we have selected values that are conservative with respect to both authoritative references and values used in analyses on previously licensed cask dockets. Specific discussions of the conservatism of the selected values are included in the following paragraphs.

As specified in Table 4.2.1, the concrete thermal conductivity is taken from Marks' Standard Handbook for Mechanical Engineers, which is conservative compared to a variety of recognized concrete codes and references. Neville, in his book "Properties of Concrete" (4th Edition, 1996), gives concrete conductivity values as high as 2.1 Btu/(hr×ft×°F). For concrete with siliceous aggregates, the type to be used in HI-STORM overpacks, Neville reports conductivities of at least 1.2 Btu/(hr×ft×°F). Data from Loudon and Stacey, extracted from Neville, reports conductivities of

---

† Aluminum Alloy 1100 heat conduction elements installed in some early serial number MPCs (MPC-68 & MPC-68F) are removed from the MPC design in Rev. 2 of the HI-STORM FSAR. Accordingly, all information and discussion pertaining to Alloy 1100 material is deleted from this section.

‡ Water from a primary source (e.g. lake or river) from which ionic impurities and precipitates have been removed.

§ For example, at 482°F, the through-thickness and width direction conductivities of Boral (B_4C thickness fraction = 0.82) is computed as 52.9 and 58.2 Btu/ft-hr-°F respectively. The SRMSS conductivity = $[(52.9^2 + 58.2^2)/2]^{0.5}$ is 55.61 BTU/ft/hr-°F compared to lowerbound METAMIC conductivity (Figure 4.2.3) of 55.68 Btu/ft-hr-°F (at 482°F).

0.980 to 1.310 Btu/(hr×ft×°F) for normal weight concrete protected from the weather. ACI-207.1R provides thermal conductivity values for seventeen structures (mostly dams) at temperatures from 50-150°F. Every thermal conductivity value reported in ACI-207.1R is greater than the 1.05 Btu/(hr×ft×°F) value used in the HI-STORM thermal analyses.

Additionally, the NRC has previously approved analyses that use higher conductivity values than those applied in the HI-STORM thermal analysis. For example, thermal calculations for the NRC approved Vectra NUHOMS cask system (June 1996, Rev. 4A) used thermal conductivities as high as 1.17 Btu/(hr×ft×°F) at 100°F. Based on these considerations, the concrete thermal conductivity value stipulated for HI-STORM thermal analyses is considered to be conservative.

Holtite-A is a composite material consisting of approximately 37 wt% epoxy polymer, 1 *wt%* B_4C and 62 *wt%* Aluminum trihydrate. Thermal conductivity of the polymeric component is low because polymers are generally characterized by a low conductivity (0.05 to 0.2 Btu/ft-hr-°F). Addition of fillers in substantial amounts raises the mixture conductivity up to a factor of ten. Thermal conductivity of epoxy filled resins with Alumina is reported in the technical literature† as approximately 0.5 Btu/ft-hr-°F and higher. In the HI-STORM FSAR, a conservatively postulated conductivity of 0.3 Btu/ft-hr-°F is used in the thermal models for the neutron shield region (in the HI-TRAC transfer cask). As the thermal inertia of the neutron shield is not credited in the analyses, the density and heat capacity properties are not reported herein.

Surface emissivity data for key materials of construction are provided in Table 4.2.4. The emissivity properties of painted external surfaces are generally excellent. Kern [4.2.5] reports an emissivity range of 0.8 to 0.98 for a wide variety of paints. In the HI-STORM thermal analysis, an emissivity of 0.85†† is applied to painted surfaces. A conservative solar absorptivity coefficient of 1.0 is applied to all exposed overpack surfaces.

In Table 4.2.5, the heat capacity and density of the ~~different~~ *MPC,* overpack *and CSF* materials are presented. These properties are used in performing transient (i.e., hypothetical fire accident condition) analyses. ~~The temperature dependence of the viscosities of helium and air are provided in Table 4.2.6 and plotted in Figure 4.2.2.~~

The heat transfer coefficient for exposed surfaces is calculated by accounting for both natural convection and thermal radiation heat transfer. The natural convection coefficient depends upon the product of Grashof (Gr) and Prandtl (Pr) numbers. Following the approach developed by Jakob and

---

† "~~Prinicples~~*Principles* of Polymer Systems", F. Rodriguez, Hemisphere Publishing Company (Chapter 10).

†† This is conservative with respect to prior cask industry practice, which has historically utilized higher emissivities. For example, a higher emissivity for painted surfaces ($\varepsilon = 0.95$) is used in the previously licensed TN-32 cask TSAR (Docket 72-1021).

Hawkins [4.2.9], the product Gr×Pr is expressed as $L^3\Delta TZ$, where L is height of the overpack, ΔT is overpack surface temperature differential and Z is a parameter based on air properties, which are known functions of temperature, evaluated at the average film temperature. The temperature dependence of Z is provided in Table 4.2.7.

Table 4.2.1

SUMMARY OF HI-STORM SYSTEM MATERIALS
THERMAL PROPERTY REFERENCES

Material	Emissivity	Conductivity	Density	Heat Capacity
Helium	N/A	Handbook [4.2.2]	Ideal Gas Law	Handbook [4.2.2]
Air	N/A	Handbook [4.2.2]	Ideal Gas Law	Handbook [4.2.2]
Zircaloy	EPRI [4.2.3]	NUREG [4.2.6], [4.2.7]	Rust [4.2.4]	Rust [4.2.4]
UO_2	Not Used	NUREG [4.2.6], [4.2.7]	Rust [4.2.4]	Rust [4.2.4]
Stainless Steel	Kern [4.2.5]	ASME [4.2.8]	Marks' [4.2.1]	Marks' [4.2.1]
Carbon Steel	Kern [4.2.5]	ASME [4.2.8]	Marks' [4.2.1]	Marks' [4.2.1]
Boral†	Not Used	Test Data	Test Data	Test Data
Holtite-A††	Not Used	Lower Bound Value Used	Not Used	Not Used
Concrete	Not Used	Marks' [4.2.1]	Marks' [4.2.1]	Handbook [4.2.2]
Lead	Not Used	Handbook [4.2.2]	Handbook [4.2.2]	Handbook [4.2.2]
Water	Not Used	ASME [4.2.10]	ASME [4.2.10]	ASME [4.2.10]
~~Aluminum Alloy 1100 (Optional Heat Conduction Elements)~~	~~Handbook [4.2.2]~~	~~ASME [4.2.8]~~	~~ASME [4.2.8]~~	~~ASME [4.2.8]~~
*METAMIC***	*Not Used*	*Test Data*	*Test Data*	*Test Data*

† AAR Structures Boral thermophysical test data.

~~†† From neutron shield manufacturer's data [1.2.11].~~

** *Test data provided by METAMIC Inc.*

Table 4.2.2

SUMMARY OF HI-STORM SYSTEM MATERIALS
THERMAL CONDUCTIVITY DATA

Material	@ 200°F (Btu/ft-hr-°F)	@ 450°F (Btu/ft-hr-°F)	@ 700°F (Btu/ft-hr-°F)
Helium	0.0976	0.1289	0.1575
Air[††]	0.0173	0.0225	0.0272
Alloy X	8.4	9.8	11.0
Carbon Steel	24.4	23.9	22.4
Concrete[‡‡]	1.05	1.05	1.05
Lead	19.4	17.9	16.9
Water	0.392	0.368	N/A

†† At lower temperatures, Air conductivity is between 0.0139 Btu/ft-hr-°F (at 32°F) and 0.0176 Btu/ft-hr-°F, at 212°F.

‡‡ Assumed constant for the entire range of temperatures.

Table 4.2.3

SUMMARY OF FUEL ELEMENT COMPONENTS
THERMAL CONDUCTIVITY DATA

Zircaloy Cladding		**Fuel (UO_2)**	
Temperature (°F)	Conductivity (Btu/ft-hr-°F)	Temperature (°F)	Conductivity (Btu/ft-hr-°F)
392	8.28†	100	3.48
572	8.76	448	3.48
752	9.60	570	3.24
932	10.44	793	2.28†

† Lowest values of conductivity used in the thermal analyses for conservatism.

Table 4.2.4

SUMMARY OF MATERIALS SURFACE EMISSIVITY DATA

Material	Emissivity
Zircaloy	0.80
Painted surfaces	0.85
Stainless steel	0.36
Carbon Steel	0.66
~~Sandblasted Aluminum~~	~~0.40~~

Note: The emissivity of a metal surface is a function of the surface finish. In general, oxidation of a metal surface increases the emissivity. As stated in Marks' Standard Handbook for Mechanical Engineers: "Unless extraordinary pains are taken to prevent oxidation, however, a metallic surface may exhibit several times the emittance or absorptance of a polished specimen." This general statement is substantiated with a review of tabulated emissivity data from several standard references. These comparisons show that oxidized metal surfaces do indeed have higher emissivities than clean surfaces.

Table 4.2.5

DENSITY AND HEAT CAPACITY PROPERTIES SUMMARY

Material	Density (lbm/ft^3)	Heat Capacity (Btu/lbm-°F)
Helium	(Ideal Gas Law)	1.24
Zircaloy	409	0.0728
Fuel (UO_2)	684	0.056
Carbon steel	489	0.1
Stainless steel	501	0.12
Boral	154.7	0.13
Concrete	142†	0.156
Lead	710	0.031
Water	62.4	0.999
~~Aluminum Alloy 1100 (Optional Heat Conduction Elements)~~	~~169.9~~	~~0.23~~
METAMIC	*163.4 – 166.6*	*0.22 – 0.29*

† ~~A minimum allowable density for concrete is specified as 146 lb/ft^3 (HI-STORM Overpack Serial Numbers 1 through 7) and 155 lb/ft^3 (HI-STORM Overpack Serial Number 8 onward) in Appendix 1.D~~. For conservatism in transient heatup calculations, *the density is understated.* ~~a lower value is specified here~~.

Table 4.2.6

GASES VISCOSITY† VARIATION WITH TEMPERATURE

Temperature (°F)	Helium Viscosity (Micropoise)††	Temperature (°F)	Air Viscosity (Micropoise)
167.4	220.5	32.0	172.0
200.3	228.2	70.5	182.4
297.4	250.6	260.3	229.4
346.9	261.8	-	-
463.0	288.7	-	-
537.8	299.8	-	-
737.6	338.8	-	-

† Obtained from Rohsenow and Hartnett [4.2.2].

†† This data is also provided in graphical form in Figure 4.2.2.

Table 4.2.7

VARIATION OF NATURAL CONVECTION PROPERTIES PARAMETER "Z" FOR AIR WITH TEMPERATURE†

Temperature (°F)	Z (ft^{-3}°F^{-1})
40	2.1×10^{6}
140	9.0×10^{5}
240	4.6×10^{5}
340	2.6×10^{5}
440	1.5×10^{5}

† Obtained from Jakob and Hawkins [4.2.9].

Table 4.2.8

BORAL COMPONENT MATERIALS†
THERMAL CONDUCTIVITY DATA

Temperature (°F)	B_4C Core Conductivity (Btu/ft-hr-°F)	Aluminum Cladding Conductivity (Btu/ft-hr-°F)
212	48.09	100.00
392	48.03	104.51
572	47.28	108.04
752	46.35	109.43

† Both B_4C and aluminum cladding thermal conductivity values are obtained from AAR Structures Boral thermophysical test data.

Table 4.2.9

[INTENTIONALLY DELETED]

~~HEAT CONDUCTION ELEMENTS (ALUMINUM ALLOY 1100)~~
~~THERMAL CONDUCTIVITY DATA~~

~~Temperature (°F)~~	~~Conductivity (Btu/ft-hr-°F)~~
~~100~~	~~131.8~~
~~200~~	~~128.5~~
~~300~~	~~126.2~~
~~400~~	~~124.5~~

FIGURES 4.2.1 and 4.2.2

[INTENTIONALLY DELETED]

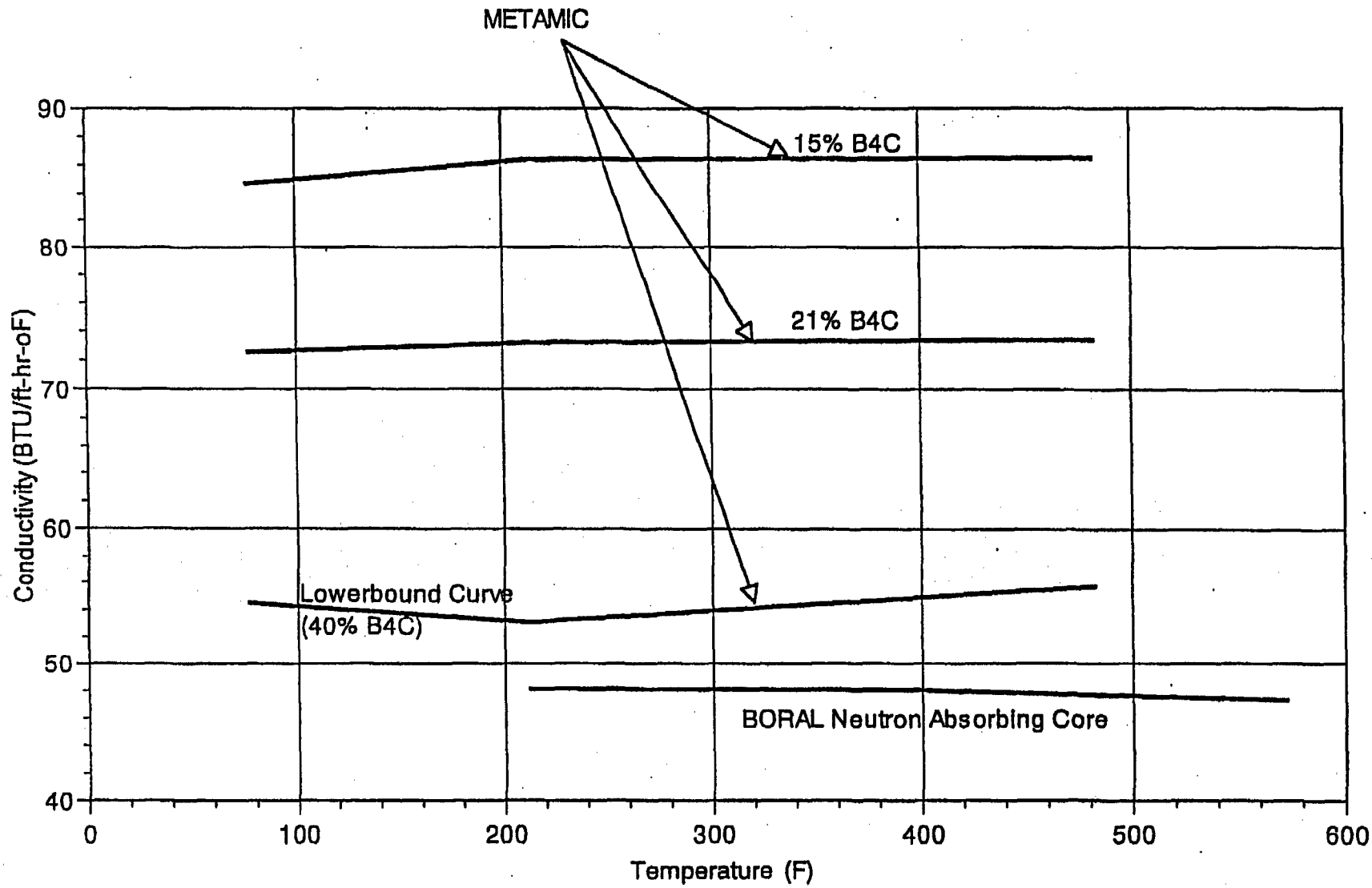


FIGURE 4.2.3: COMPARISON OF THERMAL CONDUCTIVITY OF METAMIC AND THE CERMET CORE OF A BORAL NEUTRON ABSORBER

4.3 SPECIFICATIONS FOR COMPONENTS

HI-STORM 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. The neutron shielding ability of Holtite-A neutron shield material used in the HI-TRAC ~~onsite~~ transfer *cask* ~~overpack~~ is ensured by demonstrating that the material exposure temperatures are maintained below the maximum allowable limit. Long-term integrity of SNF is ensured by the HI-STORM System thermal *evaluation which demonstrates that* ~~performance that demonstrates that~~ fuel cladding temperatures are maintained below design basis limits. *Neutron absorber materials* ~~Boral~~ used in MPC baskets for criticality control (~~a composite material composed of~~ *made from* B_4C and aluminum) ~~is~~ *are* stable up to 1000°F[†] ~~for short-term and 850°F for long-term dry storage~~[‡]. However, for conservatism, a significantly lower ~~maximum~~ temperature limit is *specified for thermal evaluation.* ~~imposed~~. The overpack concrete, the primary function of which is shielding, will maintain its structural, thermal and shielding properties provided that *the regulatory* ~~American Concrete Institute (ACI)~~ temperature limits *[4.4.10]* are not exceeded.

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-STORM System thermal performance *material temperature limits for* ~~under~~ *long term normal, short term operations,* off-normal ~~or~~ *and* ~~hypothetical~~ accident conditions ~~, material temperature limits for short-duration events~~ are provided in Table 4.3.1. *In Table 4.3.1, cladding temperature limits stipulated by ISG-11, Rev. 2 [4.1.4] are adopted for Commercial Spent Fuel (CSF). These limits are applicable to all fuel types, burnup levels and cladding materials that are approved by the NRC for power generation. Subsections 4.3.1 through 4.3.3 and their associated figures and tables are therefore no longer needed and are deleted in their entirety.*

4.3.1 Subsection Intentionally Deleted

4.3.2 Subsection Intentionally Deleted

4.3.3 Subsection Intentionally Deleted

---

† B_4C is a refractory material that is unaffected by high temperature (on the order of 1000°F) and aluminum is solid at temperatures in excess of 1000°F.

~~‡ AAR Advanced Structures Boral thermophysical test data.~~

Table 4.3.1

HI-STORM SYSTEM MATERIAL TEMPERATURE LIMITS

Material	**Normal Long-Term Temperature Limits [°F]**	**Short Term *Operations, Off-Normal and Accident* Temperature Limits [°F]**
~~Zircaloy fuel cladding~~ *CSF Cladding (Zirconium alloys and Stainless Steel)*	~~(Moderate‡ Burnup)~~ ~~See Table 4.3.7~~ *752*	*752 (Short Term Operations)* 1058 *(Off-Normal and Accident)*
~~Stainless steel fuel cladding~~	~~806~~	~~1058~~
Boral§ *Neutron Absorber*	800	950
Holtite-A*†††*	~~300~~ *N/A*	~~300~~ *350 (Short Term Operations)*
Concrete	~~200~~ *300*	350
Water	~~307~~ *N/A*	*307** (Short Term Operations)* N/A *(Off-Normal and Accident)*

~~‡ High burnup fuel storage limits are established in Appendix 4.A.~~

~~§ Based on AAR Structures Boral thermophysical test data.~~

††† See Section 1.2.1.3.2.

** *Saturation temperature at HI-TRAC water jacket design pressure.*

Tables 4.3.2 through 4.3.9 are intentionally deleted.

FIGURES 4.3.1 THROUGH 4.3.4

[INTENTIONALLY DELETED]

4.4 THERMAL EVALUATION FOR NORMAL CONDITIONS OF STORAGE

Under long-term storage conditions, the HI-STORM System (i.e., HI-STORM overpack and MPC) thermal evaluation is performed with the MPC cavity backfilled with helium. The details of the thermal models, analyses and results for long term storage scenarios are presented in this section. An overview of the methodology for characterizing the thermal-hydraulic properties of the MPC and a description of the global HI-STORM model used for obtaining the HI-STORM temperature field are provided. The modeling details are discussed in Subsections 4.4.1.1.2 through 4.4.1.1.9. As discussed in Subsection 4.0.1, the HI-STORM model incorporates the "coupled-fluid" approach needed to solve the thermo-fluid interaction of two fluids in the HI-STORM system, viz. air in the annulus and helium in the MPC.

4.4.1 Thermal Model

The MPC basket design consists of three distinct geometries to hold 24 or 32 PWR, or 68 BWR fuel assemblies. The basket is a matrix of square compartments designed to hold the fuel assemblies in a vertical position. The basket is a honeycomb structure of alloy steel (Alloy X) plates with full-length edge-welded intersections to form an integral basket configuration. All individual cell walls, except outer periphery cell walls in the MPC-68 and MPC-32, are provided with neutron absorber sandwiched between the box wall and a stainless steel sheathing plate over the full length of the active fuel region.

To ensure a robust margin in the HI-STORM system, the MPC property characterizations employ a composite of limiting thermal-hydraulic characteristics among each of two fuel classes (PWR and BWR fuel). These characteristics are: (a) An upperbound planar thermal resistance, (b) a lowerbound axial thermal conductivity and (c) an upperbound axial flow resistance. The fuel types that obtain the limiting result for each of the three characteristics (a), (b) and (c) above are listed below for ready reference.

I) Fuel Class: PWR Fuel
   (a) Planar thermal resistance: W 17x17 OFA
   (b) Axial thermal conductivity: W 14x14 OFA
   (c) Axial flow resistance: B&W 15x15

II) Fuel Class: BWR Fuel
   (a) Planar thermal resistance: GE-11 9x9
   (b) Axial thermal conductivity: GE 7x7
   (c) Axial flow resistance: GE-12/14 10x10

The design basis decay heat generation (Q_d) is defined in Table 4.4.39. The decay heat is conservatively considered to be non-uniformly distributed over the active fuel length based on the design basis axial burnup distributions provided in Chapter 2 (Table 2.1.11).

4.4.1.1 Analytical Model - General Remarks

Transport of heat from the heat generation region (fuel assemblies) to the outside environment (ambient air or ground) is analyzed broadly in using three thermal models.

1. The first model considers transport of heat from the fuel assembly to the basket cell walls. This model recognizes the combined effects of conduction (through helium) and radiation, and is essentially a finite element technology based update of the classical Wooton & Epstein [4.4.1] formulation (which considered radiative heat exchange between fuel rod surfaces).

2. The second model considers heat transport within an MPC fuel basket cross section by conduction and radiation. The effective cross sectional thermal conductivity of the basket region, obtained from a combined fuel assembly/basket heat conduction-radiation model developed on ANSYS, is applied to an axisymmetric thermal model of the HI-STORM System on the FLUENT [4.1.2] code. To model MPC internal convection heat transfer, the fuel basket is rendered as a porous media having effective flow resistance characteristics.

3. The third model deals with the transmission of heat from the MPC exterior surface to the external environment (heat sink). The upflowing air stream in the MPC/cask annulus extracts most of the heat from the external surface of the MPC, and a small amount of heat is radially deposited on the HI-STORM inner surface by conduction and radiation. Heat rejection from the outside cask surfaces to ambient air is considered by accounting for natural convection and radiative heat transfer mechanisms from the vertical (cylindrical shell) and top cover (flat) surfaces. The reduction in radiative heat exchange between cask outside vertical surfaces and ambient air, because of blockage from the neighboring casks arranged for normal storage at an ISFSI pad as described in Section 1.4, is recognized in the analysis. The overpack top plate is modeled as a heated surface in convective and radiative heat exchange with air and as a recipient of heat input through insolation. Insolation on the cask surfaces is based on 12-hour levels prescribed in 10CFR71, averaged over a 24-hour period, after accounting for partial blockage conditions on the sides of the overpack.

Subsections 4.4.1.1.1 through 4.4.1.1.9 contain a description of the mathematical models devised to articulate the temperature field in the HI-STORM System. The description begins with the method to characterize the heat transfer behavior of the prismatic (square) opening referred to as the "fuel space" with a heat emitting fuel assembly situated in it. The methodology utilizes a finite element procedure to replace the heterogeneous SNF/fuel space region with an equivalent solid body having a well-defined temperature-dependent conductivity. In the following subsection, the method to replace the "composite" walls of the fuel basket cells with an equivalent "solid" wall is presented. Having created the mathematical equivalents for the SNF/fuel spaces and the fuel basket walls, the method to represent the MPC cylinder containing the fuel basket by an equivalent cylinder whose thermal conductivity is a function of the spatial location and coincident temperature is presented.

Following the approach of presenting descriptions starting from the inside and moving to the outer region of a cask, the next subsections present the mathematical model to simulate the overpack.

Subsection 4.4.1.1.9 concludes the presentation with a description of how the different models for the specific regions within the HI-STORM System are assembled into the final FLUENT model.

4.4.1.1.1 Overview of the Thermal Model

Thermal analysis of the HI-STORM System is performed by assuming that the system is subject to its maximum heat duty with each storage location occupied and with the heat generation rate in each stored fuel assembly equal to the design-basis maximum value. While the assumption of equal heat generation imputes a certain symmetry to the cask thermal problem, the thermal model must incorporate three attributes of the physical problem to perform a rigorous analysis of a fully loaded cask, namely:

i. While the rate of heat conduction through metals is a relatively weak function of temperature, radiation heat exchange is a nonlinear function of surface temperatures.

ii. Heat generation in the MPC is axially non-uniform due to non-uniform axial burnup profiles in the fuel assemblies.

iii. Inasmuch as the transfer of heat occurs from inside the basket region to the outside, the temperature field in the MPC is spatially distributed with the maximum values reached in the central core region.

The cross section bounded by the inside of the storage cell, which surrounds the assemblage of fuel rods and the interstitial helium gas, is replaced with an "equivalent" square (solid) section characterized by an effective thermal conductivity. Figure 4.4.1 pictorially illustrates the homogenization concept. Further details of this procedure for determining the effective conductivity are presented in Subsection 4.4.1.1.2; it suffices to state here that the effective conductivity of the cell space will be a function of temperature because the radiation heat transfer (a major component of the heat transport between the fuel rods and the surrounding basket cell metal) is a strong function of the temperatures of the participating bodies. Therefore, in effect, every storage cell location will have a different value of effective conductivity (depending on the coincident temperature) in the homogenized model. The temperature-dependent fuel assembly region effective conductivity is determined by a finite volume procedure, as described in Subsection 4.4.1.1.2.

In the next step of homogenization, a planar section of an MPC fuel basket is considered. With each storage cell inside space replaced with an equivalent solid square, the MPC cross section consists of a metallic gridwork (basket cell walls with each square cell space containing a solid fuel cell square of effective thermal conductivity, which is a function of temperature) . There are four distinct materials in this section, namely the homogenized fuel cell , the Alloy X structural materials (including neutron absorber sheathing), neutron absorber, and helium gas. Each of the four constituent materials in this section has a different conductivity. It is emphasized that the conductivity of the homogenized fuel cells is a strong function of temperature.

In order to replace this thermally heterogeneous MPC fuel basket section with an equivalent conduction-only region, resort to the finite element procedure is necessary. Because the rate of

transport of heat within the MPC fuel basket is influenced by radiation, which is a temperature-dependent effect, the equivalent conductivity of the MPC fuel basket region must also be computed as a function of temperature. Finally, it is recognized that the MPC section consists of two discrete regions, namely, the fuel basket region and the peripheral region. The peripheral region is the space between the peripheral storage cells and the MPC shell. This space is essentially full of helium surrounded by Alloy X plates. Accordingly, as illustrated in Figure 4.4.2 for MPC-68, the MPC cross section is replaced with two homogenized regions with temperature-dependent conductivities. In particular, the effective conductivity of the fuel cells is subsumed into the equivalent conductivity of the basket cross section. The finite element procedure used to accomplish this is described in Subsection 4.4.1.1.4. The ANSYS finite element code is the vehicle for all modeling efforts described in the foregoing.

In summary, appropriate finite-element models are used to replace the MPC cross section with an equivalent two-region homogeneous conduction lamina whose local conductivity is a known function of coincident absolute temperature. Thus, the MPC cylinder containing discrete fuel assemblies, helium, neutron absorber and Alloy X, is replaced with a right circular cylinder whose material conductivity will vary with radial and axial position as a function of the coincident temperature. The thermal modeling duly recognizes the MPC as a non-isotropic 3-D conduction media. In particular, the MPC fuel basket in-plane conductivity and the axial conductivity differ considerably. This is because while in-plane heat transfer is interrupted by gaps, in the axial direction heat flows in an uninterrupted manner in the fuel rods, the fuel cell walls and neutron absorber plates. Accordingly, the thermal properties of the MPC fuel basket continuum are characterized by two conductivities:

(i) Planar conductivity (K_r) as a function of temperature
(ii) Axial conductivity (K_{ax}) as a function of temperature

The procedure for computing K_r and K_{ax} is described in Subsection 4.4.1.1.4. As stated earlier in this section, the MPC thermal property evaluations conservatively employ limiting fuel types for computing K_r and K_{ax}. These limiting fuel types are listed earlier in Subsection 4.4.1. Finally, to model internal convection in the MPC, the fuel storage cells are rendered as porous media having effective pressure drop characteristics. The methodology for modeling flow resistance is discussed next.

For simulating the flow resistance of fuel cells, the FLUENT porous media pressure drop model is employed. In this model, pressure drop is computed (in MKS units) as follows:

$$\frac{\Delta P}{L} = \frac{\mu}{\alpha} V + C\left(\frac{1}{2}\rho V^2\right)$$

where:

$\Delta P/L$ is pressure drop per unit length (Pa/m)
V is superficial fluid velocity (m/s)
μ is fluid viscosity (kg/m-s)
ρ is fluid density (kg/m^3)

α is permeability parameter (m^2)
C is inertial resistance factor (m^{-1})

The PWR and BWR fuel assemblies occupying the fuel cells in an MPC are essentially an array of regularly spaced fuel rods supported by an open grid structure at discrete locations. From basic hydraulics, pressure drop in a fuel cell is obtained as the sum of two parts: (i) frictional loss from the array of fuel rods and (ii) pressure losses from fluid contractions and expansions at the grid locations. For maximizing frictional losses, the thermal analysis employs the limiting fuel type for which the smallest hydraulic diameter is obtained. For maximizing grid spacer flow resistances, two assumptions are employed:

a) Spacers are a gridwork of thick (0.05 inch) plates
b) Bounding expansion and contraction loss factors are used

The first assumption above understates flow area at the grid spacer location and the second assumption overstates hydraulic losses. Together the assumptions ensure a substantial conservatism in the fuel assembly pressure drop characteristics.

Employing basic principles of fluid mechanics, the frictional and grid hydraulic losses are computed and appropriate values of the parameters (α and C) in the porous media pressure drop model obtained. The flow resistances are computed for the limiting fuel type‡‡ in each class (PWR or BWR fuel). The limiting fuel types were listed earlier in Section 4.1.1. The principal steps for computing "α" and "C" are provided next.

Step 1: Compute Hydraulic Diameter (D_h)

$$D_h = \frac{4A_f}{WP}$$

where:
A_f is cell flow area (cell opening area minus area of fuel rods)
WP is wetted perimeter (sum of rods circumference and cell walls perimeter)

Step 2: Compute Porosity (ε)

$$\varepsilon = \frac{A_f}{A_o}$$

where A_o is cell area defined by the square of cell pitch

Step 3: Compute "α"

$$\alpha = \frac{\varepsilon D_h^2}{32}$$

‡‡ The limiting fuel type has the smallest hydraulic diameter.

Step 4: Compute Cumulative Grids Loss Factor (K)

$$K = N\,(Kc + Ke)$$

where:

N is number of flow grids
Kc is flow contraction coefficient
Ke is expansion coefficient

Step 5: Compute "C"

$$C = \frac{K}{\varepsilon_g^2 L}$$

where:

L is fuel cell length
ε_g is ratio of grids flow area to cell area A_o

The porous media pressure model described in the foregoing is benchmarked with proprietary pressure drop data to confirm that the model predictions are conservative. As stated previously, the flow resistance parameters (α and C) are obtained for the limiting fuel type in each class (PWR or BWR fuel) for use in the HI-STORM thermal evaluations.

Internal circulation of helium in the sealed MPC is modeled as flow in a porous media in the fueled region containing the SNF (including top and bottom plenums). The basket-to-MPC shell clearance space is modeled as a helium filled radial gap to include the downcomer flow in the thermal model. The downcomer region, as illustrated in Figure 4.4.2, consists of an azimuthally varying gap formed by the square-celled basket outline and the cylindrical MPC shell. At the locations of closest approach a differential expansion gap (a small clearance on the order of 1/10 of an inch) is engineered to allow free thermal expansion of the basket. At the widest locations, the gaps are on the order of the fuel cell opening (~6" (BWR) and ~9" (PWR) MPCs). It is heuristically evident that heat dissipation by conduction is maximum at the closest approach locations (low thermal resistance path) and that convective heat transfer is highest at the widest gap locations (large downcomer flow). In the FLUENT thermal model, a radial gap that is large compared to the basket-to-shell clearance and small compared to the cell opening is used. As a relatively large gap penalizes heat dissipation by conduction and a small gap throttles convective flow, the use of a single gap in the FLUENT model understates both conduction and convection heat transfer in the downcomer region.
In this manner, a loaded MPC standing upright on the ISFSI pad in a HI-STORM overpack is replaced in the model as a right circular cylinder with spatially varying temperature-dependent conductivity. Heat is generated within the basket space in this cylinder in the manner of the prescribed axial burnup distribution. In addition, heat is deposited from insolation on the external surface of the overpack. Under steady state conditions the total heat due to internal generation and insolation is dissipated from the outer cask surfaces by natural convection and thermal radiation to the ambient environment and from heating of upward flowing air in the annulus. Details of the elements of mathematical modeling are provided in the following.

4.4.1.1.2 Fuel Region Effective In-Plane Thermal Conductivity Calculation

Thermal properties of a large number of PWR and BWR fuel assembly configurations manufactured by the major fuel suppliers (i.e., Westinghouse, CE, B&W, and GE) have been evaluated for inclusion in the HI-STORM System thermal analysis. Bounding PWR and BWR fuel assembly configurations are determined through a screening process using the simplified procedure described below. This is followed by the determination of temperature-dependent properties for the bounding PWR and BWR fuel assembly configurations to be used for cask thermal analysis using a finite volume (FLUENT) approach.

To determine which of the PWR assembly types listed in Table 4.4.1 should be used in the thermal model for the PWR fuel baskets (MPC-24, MPC-24E, MPC-32), we first establish which assembly type has the maximum in-plane thermal resistance. The same determination is made for the MPC-68, out of the SNF types listed in Table 4.4.2. For this purpose, we utilize a simplified procedure that we describe below.

Each fuel assembly consists of a large array of fuel rods typically arranged on a square layout. Every fuel rod in this array is generating heat due to radioactive decay in the enclosed fuel pellets. There is a finite temperature difference required to transport heat from the innermost fuel rods to the storage cell walls. To identify the CSF with the highest in-plane thermal resistance the model proposed by Wooton and Epstein [4.4.1] is used.

According to [4.4.1], transport of heat energy within any cross section of a fuel assembly occurs through a combination of radiative energy exchange and conduction through the helium gas that fills the interstices between the fuel rods in the array. With the assumption of uniform heat generation within any given horizontal cross section of a fuel assembly, the combined radiation and conduction heat transport effects result in the following heat flow equation:

$$Q = \sigma C_o F_\varepsilon A [T_C^4 - T_B^4] + 13.5740 L K_{es} [T_C - T_B]$$

where:

F_ε = Emissivity Factor

$$= \frac{1}{\left(\frac{1}{\varepsilon_C} + \frac{1}{\varepsilon_B} - 1\right)}$$

ε_C, ε_B = emissivities of fuel cladding, fuel basket (see Table 4.2.4)

C_o = Assembly Geometry Factor

$$= \frac{4N}{(N+1)^2} \text{ (when N is odd)}$$

$$= \frac{4}{N+2} \text{ (when N is even)}$$

N = Number of rows or columns of rods arranged in a square array
A = fuel assembly "box" heat transfer area = 4 × width × length
L = fuel assembly length
K_{cs} = fuel assembly constituent materials volume fraction weighted mixture conductivity
T_C = hottest fuel cladding temperature (°R)
T_B = box temperature (°R)
Q = net radial heat transport from the assembly interior
σ = Stefan-Boltzmann Constant (0.1714×10^{-8} Btu/ft^2-hr-°R^4)

In the above heat flow equation, the first term is the Wooten-Epstein radiative heat flow contribution while the second term is the conduction heat transport contribution based on the classical solution to the temperature distribution problem inside a square shaped block with uniform heat generation [4.4.5]. The 13.574 factor in the conduction term of the equation is the shape factor for two-dimensional heat transfer in a square section. Planar fuel assembly heat transport by conduction occurs through a series of resistances formed by the interstitial helium fill gas, fuel cladding and enclosed fuel. An effective planar mixture conductivity is determined by a volume fraction weighted sum of the individual constituent material resistances. For BWR assemblies, this formulation is applied to the region inside the fuel channel. A second conduction and radiation model is applied between the channel and the fuel basket gap. These two models are combined, in series, to yield a total effective conductivity.

The effective conductivities of the fuel types for several representative PWR and BWR assemblies are presented in Tables 4.4.1 and 4.4.2. At higher temperatures (approximately 450°F and above), the zircaloy clad fuel assemblies with the lowest effective thermal conductivities are the W-17×17 OFA (PWR) and the GE11-9×9 (BWR). Based on this simplified analysis, the W-17×17 OFA PWR and GE11-9×9 BWR fuel assemblies are determined to be the bounding fuel types for planar conductivity of zircaloy clad fuel.

Having established the governing (most resistive) PWR and BWR SNF types, we use a finite-volume code to determine the effective conductivities in a rigorous manner. Detailed conduction-radiation finite-volume models of the bounding PWR and BWR fuel assemblies developed on the FLUENT code are shown in Figures 4.4.3 and 4.4.4, respectively. As discussed later in this subsection, the models depicted in these figures have been benchmarked with results from independent technical sources to confirm that they yield conservative results.

The combined fuel rods-helium matrix is replaced by an equivalent homogeneous material that fills the basket opening by the following two-step procedure. In the first step, the FLUENT-based fuel assembly model is solved by applying equal heat generation per unit length to the individual fuel

rods and a uniform boundary temperature along the basket cell opening inside periphery. The temperature difference between the peak cladding and boundary temperatures is used to determine an effective conductivity as described in the next step. For this purpose, we consider a two-dimensional cross section of a square shaped block with an edge length of 2L and a uniform volumetric heat source (q_g), cooled at the periphery with a uniform boundary temperature. Under the assumption of constant material thermal conductivity (K), the temperature difference (ΔT) from the center of the cross section to the periphery is analytically given by [4.4.5]:

$$\Delta T = 0.29468 \frac{q_g L^2}{K}$$

This analytical formula is applied to determine the effective material conductivity from a known quantity of heat generation applied in the FLUENT model (smeared as a uniform heat source, q_g) basket opening size and ΔT calculated in the first step.

As discussed earlier, the effective fuel space conductivity must be a function of the temperature coordinate. The above two-step analysis is carried out for a number of reference temperatures. In this manner, the effective conductivity as a function of temperature is established.

Temperature-dependent effective conductivities of PWR and BWR design basis fuel assemblies (most resistive SNF types) are shown in Figure 4.4.5. The finite volume results are also compared to results reported from independent technical sources. From this comparison, it is readily apparent that FLUENT-based fuel assembly conductivities are conservative. The FLUENT computed values (not the published literature data) are used in the MPC thermal analysis presented in this document.

4.4.1.1.3 Effective Thermal Conductivity of Neutron Absorber/Sheathing/Box Wall Sandwich

Each MPC basket cell wall (except the MPC-68 and MPC-32 outer periphery cell walls) is manufactured with a neutron absorbing plate for criticality control. Each neutron absorber plate is sandwiched in a sheathing-to-basket wall pocket. A schematic of the "Box Wall- Neutron absorber-Sheathing" sandwich geometry of an MPC basket is illustrated in Figures 4.4.6 and 4.4.7. During fabrication, a uniform normal pressure is applied to each "Box Wall-Neutron Absorber-Sheathing" sandwich in the assembly fixture during welding of the sheathing periphery on the box wall. This ensures adequate surface-to-surface contact for elimination of any macroscopic gaps. The mean coefficient of linear expansion of the neutron absorber is higher than the thermal expansion coefficients of the basket and sheathing materials. Consequently, basket heat-up from the stored SNF will further ensure a tight fit of the neutron absorber plate in the sheathing-to-box pocket. The presence of small microscopic gaps due to less than perfect surface finish characteristics requires consideration of an interfacial contact resistance between the neutron absorber and box-sheathing surfaces. A conservative contact resistance resulting from a 2 mil neutron absorber to pocket gap is applied in the analysis. In other words, no credit is taken for the interfacial pressure between neutron absorber and stainless plate/sheet stock produced by the fixturing and welding process.

Heat conduction properties of a composite "Box Wall- Neutron absorber-Sheathing" sandwich in the two principal basket cross sectional directions as illustrated in Figure 4.4.6 (i.e., lateral "out-of-plane" and longitudinal "in-plane") are unequal. In the lateral direction, heat is transported across layers of sheathing, - helium gap, neutron absorber and box wall resistances that are essentially in series (except for the small helium filled end regions shown in Figure 4.4.7). Heat conduction in the longitudinal direction, in contrast, is through an array of essentially parallel resistances comprised of these several layers listed above. For the ANSYS based MPC basket thermal model, corresponding non-isotropic effective thermal conductivities in the two orthogonal sandwich directions are determined and applied in the analysis.

These non-isotropic conductivities are determined by constructing two-dimensional finite-element models of the composite "Box Wall- Neutron Absorber-Sheathing" sandwich in ANSYS. A fixed temperature (T_c) is applied to one edge of the model and a fixed heat flux is applied to the other edge, and the model is solved to obtain the average temperature (T_h) of the fixed-flux edge. The equivalent thermal conductivity is the obtained using the resulting temperature difference across the sandwich as input to a one-dimensional Fourier equation as follows:

$$K_{eff} = \frac{q \times L}{T_h - T_c}$$

where:

K_{eff} = effective thermal conductivity
q = heat flux applied in the ANSYS model
L = ANSYS model heat transfer path length
T_h = ANSYS calculated average edge temperature
T_c = specified edge temperature

The heat transfer path length will vary, depending on the direction of transfer (i.e., in-plane or out-of-plane).

4.4.1.1.4 Modeling of Fuel Basket Conductive Heat Transport

The total conduction heat rejection capability of a fuel basket is a combination of planar and axial contributions. These component contributions are calculated independently for each MPC basket design and reported herein .

The planar heat rejection capability of each MPC basket design (i.e., MPC-24, MPC-68, MPC-32 and MPC-24E) is evaluated by developing a thermal model of the combined fuel assemblies and composite basket walls geometry on the ANSYS finite element code. The ANSYS model includes a geometric layout of the basket structure in which the basket "Box Wall- Neutron Absorber-Sheathing" sandwich is replaced by a "homogeneous wall" with an equivalent thermal conductivity. Since the thermal conductivity of the Alloy X material is a weakly varying function of temperature, the equivalent "homogeneous wall" will have a temperature-dependent effective conductivity. Similarly, as illustrated in Figure 4.4.7, the conductivities in the "in-plane" and "out-of-plane" directions of the equivalent "homogeneous wall" are different. Finally, as discussed earlier, the fuel assemblies and the surrounding basket cell openings are modeled as homogeneous heat generating

regions with an effective temperature dependent in-plane conductivity. The methodology used to reduce the heterogeneous MPC basket - fuel assemblage to equivalent thermal properties in the in-plane and axial directions is discussed in this subsection.

(i) <u>In-Plane Conductivity of the Fuel Basket</u>

Consider a cylinder of height, L, and radius, r_o, with a uniform volumetric heat source term, q_g, insulated top and bottom faces, and its cylindrical boundary maintained at a uniform temperature, T_c. The maximum centerline temperature (T_h) to boundary temperature difference is readily obtained from classical one-dimensional conduction relationships (for the case of a conducting region with uniform heat generation and a constant thermal conductivity K_e):

$$(T_h - T_c) = q_g r_o^2/(4 K_e)$$

Noting that the total heat generated in the cylinder (Q_t) is $\pi r_o^2 L q_g$, the above temperature rise formula can be reduced to the following simplified form in terms of total heat generation per unit length (Q_t/L):

$$(T_h - T_c) = (Q_t / L)/ (4 \pi K_e)$$

This simple analytical approach is employed to determine an effective basket cross-sectional conductivity by applying an equivalence between the ANSYS finite element model of the basket and the analytical case. The equivalence principle employed in the thermal analysis is depicted in Figure 4.4.2. The 2-dimensional ANSYS finite element model of the MPC basket is solved by applying a uniform heat generation per unit length in each basket cell region (depicted as Zone 1 in Figure 4.4.2) and a constant basket periphery boundary temperature, T_c'. Noting that the basket region with uniformly distributed heat sources and a constant boundary temperature is equivalent to the analytical case of a cylinder with uniform volumetric heat source discussed earlier, an effective MPC basket conductivity (K_{eff}) is readily derived from the analytical formula and ANSYS solution leading to the following relationship:

$$K_{eff} = N (Q_f'/L) / (4 \pi [T_h' - T_c'])$$

where:

N = number of fuel assemblies
(Q_f'/L) = per fuel assembly heat generation per unit length applied in ANSYS model
T_h' = peak basket cross-section temperature from ANSYS model

Cross sectional views of MPC basket ANSYS models are depicted in Figures 4.4.9 and 4.4.10. Temperature-dependent equivalent thermal conductivities of the fuel regions and composite basket walls, as determined from analysis procedures described earlier, are applied to the ANSYS model. The planar ANSYS conduction model is solved by applying a constant basket periphery temperature with uniform heat generation in the fuel region. The equivalent planar thermal conductivity values are lower bound values because, among other elements of conservatism, the effective conductivity of

the most resistive SNF types as stated earlier in Subsection 4.4.1 ( ) is used in the MPC finite element simulations.

The basket in-plane conductivities are computed for intact fuel storage and containerized fuel stored in Damaged Fuel Containers (DFCs). The MPC-24E is provided with four enlarged cells designated for storing damaged fuel. The MPC-68 has sixteen peripheral locations for damaged fuel storage in generic DFC designs. As a substantial fraction of the basket cells are occupied by intact fuel, the overall effect of DFC fuel storage on the basket heat dissipation rate is quite small. Including the effect of reduced conductivity of the DFC cells in MPC-24E, the basket conductivity is computed to drop slightly (~0.6%). In a bounding evaluation in which the sixteen outer cells are occupied with damaged fuel, the effect of reduced conductivity on the PCT is negligible (~ 1°F). Therefore, DFCs do not pose a limitation on safe storage of fuel. The fuel basket planar conductivities are provided in Table 4.4.3.

(ii) Axial Conductivity of the Fuel Basket

The axial heat rejection capability of each MPC basket design is determined by calculating the area occupied by each material in a fuel basket cross-section, multiplying by the corresponding material thermal conductivity, summing the products and dividing by the total fuel basket cross-sectional area. In accordance with NUREG-1536 guidelines, the only portion of the fuel assemblies credited in these calculations is the fuel rod cladding (i.e. the contribution of fuel pellets to axial heat conduction is ignored). In Table 4.4.3 the MPC fuel basket planar and axial conductivities are provided.

4.4.1.1.5 Subsection Intentionally Deleted

4.4.1.1.6 Subsection Intentionally Deleted

4.4.1.1.7 Annulus Air Flow and Heat Exchange

The HI-STORM storage overpack is provided with four inlet ducts at the bottom and four outlet ducts at the top. The ducts are provided to enable relatively cooler ambient air to flow through the annular gap between the MPC and storage overpack in the manner of a classical "chimney". Hot air is vented from the top outlet ducts to the ambient environment. Buoyancy forces induced by density differences between the ambient air and the heated air column in the MPC-to-overpack annulus sustain airflow through the annulus.

In contrast to a classical chimney, however, the heat input to the HI-STORM annulus air does not occur at the bottom of the stack. Rather, the annulus air picks up heat from the lateral surface of the MPC shell as it flows upwards. The height dependent heat absorption by the annulus air must be properly accounted for to ensure that the buoyant term in the Bernoulli equation is not overstated making the solution unconservative. For this purpose, a "coupled-fluid" model of the HI-STORM System is constructed for all HI-STORM analyses. The model, depicted in Figure 4.0.3 and discussed in Subsection 4.0.1 includes both fluids, viz. air (in the annulus) and helium (in the MPC),

in one model. The model incorporates the interaction between the annulus air and the hot helium circulating inside the MPC.

4.4.1.1.8 Determination of Solar Heat Input

The intensity of solar radiation incident on an exposed surface depends on a number of time varying terms. The solar heat flux strongly depends upon the time of the day as well as on latitude and day of the year. Also, the presence of clouds and other atmospheric conditions (dust, haze, etc.) can significantly attenuate solar intensity levels. Rapp [4.4.2] has discussed the influence of such factors in considerable detail.

Consistent with the guidelines in NUREG-1536 [4.4.10], solar input to the exposed surfaces of the HI-STORM overpack is determined based on 12-hour insolation levels recommended in 10CFR71 (averaged over a 24-hour period) and applied to the most adversely located cask after accounting for partial blockage of incident solar radiation on the lateral surface of the cask by surrounding casks. In reality, the lateral surfaces of the cask receive solar heat depending on the azimuthal orientation of the sun during the course of the day. In order to bound this heat input, the lateral surface of the cask is assumed to receive insolation input with the solar insolation applied horizontally into the cask array. The only reduction in the heat input to the lateral surface of the cask is due to partial blockage offered by the surrounding casks. In contrast to its lateral surface, the top surface of HI-STORM is fully exposed to insolation without any mitigation effects of blockage from other bodies. In order to calculate the view factor between the most adversely located HI-STORM system in the array and the environment, a conservative geometric simplification is used. The system is reduced to a concentric cylinder model, with the inner cylinder representing the HI-STORM unit being analyzed and the outer shell representing a reflecting boundary (no energy absorption).

Thus, the radius of the inner cylinder (R_i) is the same as the outer radius of a HI-STORM overpack. The radius of the outer cylinder (R_o) is set such that the rectangular space ascribed to a cask is preserved. This is further explained in the next subsection. It can be shown that the view factor from the outer cylinder to the inner cylinder (F_{o-i}) is given by [4.4.3]:

$$F_{o-i} = \frac{1}{R} - \frac{1}{\pi R} \times [\cos^{-1}(\frac{B}{A}) - \frac{1}{2L}\{\sqrt{(A+2)^2 - (2R)^2} \times \cos^{-1}(\frac{B}{RA}) + B\sin^{-1}(\frac{1}{R}) - \frac{\pi A}{2}\}]$$

where:

F_{o-i} = View Factor from the outer cylinder to the inner cylinder
R = Outer Cylinder Radius to Inner Cylinder Radius Ratio (R_o/R_i)
L = Overpack Height to Radius Ratio
$A = L^2 + R^2 - 1$
$B = L^2 - R^2 + 1$

Applying the theorem of reciprocity, the view factor (F_{i-a}) from outer overpack surface, represented by the inner cylinder, to the ambient can be determined as:

$$F_{i-a} = 1 - F_{o-i}\frac{R_o}{R_i}$$

Finally, to bound the quantity of heat deposited onto the HI-STORM surface by insolation, the absorptivity of the cask surfaces is assumed to be unity.

4.4.1.1.9 FLUENT Model for HI-STORM

In the preceding subsections, a series of analytical and numerical models to define the thermal characteristics of the various elements of the HI-STORM System are presented. The thermal modeling begins with the replacement of the Spent Nuclear Fuel (SNF) cross section and surrounding fuel cell space with a porous region with an equivalent conductivity. Since radiation is an important constituent of the heat transfer process in the SNF/storage cell space, and the rate of radiation heat transfer is a strong function of the surface temperatures, it is necessary to treat the equivalent region conductivity as a function of temperature. Because of the relatively large range of temperatures in a loaded HI-STORM System under the design basis heat loads, the effects of variation in the thermal conductivity of the Alloy X basket wall with temperature are included in the numerical analysis model. The presence of significant radiation effects in the storage cell spaces adds to the imperative to treat the equivalent storage cell lamina conductivity as temperature-dependent.

Numerical calculations and FLUENT finite-volume simulations described in Subsection 4.4.1.1.2 provide the equivalent thermal conductivity as a function of temperature for the limiting (thermally most resistive) BWR and PWR spent fuel types. Utilizing the most limiting SNF (established through a simplified analytical process for comparing conductivities) ensures that the numerical idealization for the fuel space effective conductivity is conservative for all fuel types.

Having replaced the fuel spaces by porous blocks with a temperature-dependent conductivity essentially renders the basket into a non-homogeneous three-dimensional solid where the non-homogeneity is introduced by the honeycomb basket structure composed of interlocking basket panels. The basket panels themselves are a composite of Alloy X cell wall, neutron absorber, and Alloy X sheathing metal. A conservative approach to replace this composite section with an equivalent "solid wall" was described earlier.

In the next step, a planar section of the MPC is considered. The MPC contains a non-symmetric basket lamina wherein the equivalent fuel spaces are separated by the "equivalent" solid metal walls. The space between the basket and the MPC, called the peripheral gap, is filled with helium gas. At this stage in the thermal analysis, the SNF/basket/MPC assemblage has been replaced with a two-zone (Figure 4.4.2) cylindrical region whose thermal conductivity is a strong function of temperature.

The thermal model for the HI-STORM overpack is prepared as a three-dimensional axisymmetric body. For this purpose, the hydraulic resistances of the inlet ducts and outlet ducts, respectively, are represented by equivalent axisymmetric porous media. Two overpack configurations are evaluated – a classic design (HI-STORM 100) and an enhanced version (HI-STORM 100S) overpack The HI-STORM 100 and HI-STORM 100S overpacks are thermally similar in as much as they yield thermal

solutions that are in reasonable agreement (~5°F). .The axial resistance to airflow in the MPC/overpack annulus (which includes longitudinal channels to "cushion" the stresses in the MPC structure during a postulated non-mechanistic tip-over event) is replaced by a hydraulically equivalent annulus. The surfaces of the ducts and annulus are assumed to have a relative roughness (ε) of 0.001. This value is appropriate for rough cast iron, wood stave and concrete pipes, and is bounding for smooth painted surfaces (all readily accessible internal and external HI-STORM overpack carbon steel surfaces are protected from corrosion by painting or galvanization). Finally, it is necessary to describe the external boundary conditions to the overpack situated on an ISFSI pad. An isolated HI-STORM will take suction of cool air from and reject heated air to, a semi-infinite half-space. In a rectilinear HI-STORM array, however, the unit situated in the center of the grid is evidently hydraulically most disadvantaged, because of potential interference to air intake from surrounding casks. To simulate this condition in a conservative manner, we erect a hypothetical cylindrical barrier around the centrally local HI-STORM. The radius of this hypothetical cylinder, R_o, is computed from the equivalent cask array downflow hydraulic diameter (D_h) which is obtained as follows:

$$D_h = \frac{4 \times \text{Flow Area}}{\text{Wetted Perimeter}}$$

$$= \frac{4\left(A_o - \frac{\pi}{4} d_o^{\,2}\right)}{\pi d_o}$$

where:

A_o = Minimum tributary area ascribable to one HI-STORM (see Figure 4.4.24).
d_o = HI-STORM overpack outside diameter

The hypothetical cylinder radius, R_o, is obtained by adding half D_h to the radius of the HI-STORM overpack. In this manner, the hydraulic equivalence between the cask array and the HI-STORM overpack to hypothetical cylindrical annulus is established.

For purposes of the design basis analyses reported in this chapter, the tributary area A_o is assumed to be equal to 346 sq. ft. Sensitivity studies on the effect of the value of A_o on the thermal performance of the HI-STORM System shows that the system response is essentially insensitive to the assumed value of the tributary area. For example, a thermal calculation using A_o=225 sq. ft. (corresponding to 15 ft. square pitch) and design basis heat load showed that the peak cladding temperature is less than 1°C greater than that computed using A_o = 346 sq. ft. Therefore, the distance between the vertically arrayed HI-STORMs in an ISFSI should be guided by the practical (rather than thermal) considerations, such as personnel access to maintain air ducts or painting the cask external surfaces.

The internal surface of the hypothetical cylinder of radius R_o surrounding the HI-STORM module is conservatively assumed to be insulated. Any thermal radiation heat transfer from the HI-STORM

overpack to this insulated surface will be perfectly reflected, thereby bounding radiative blocking from neighboring casks. Then, in essence, the HI-STORM module is assumed to be confined in a large cylindrical "tank" whose wall surface boundaries are modeled as zero heat flux boundaries. The air in the "tank" is the source of "feed air" to the overpack. The air in the tank is replenished by ambient air from above the top of the HI-STORM overpacks. There are two sources of heat input to the exposed surface of the HI-STORM overpack. The most important source of heat input is the internal heat generation within the MPC. The second source of heat input is insolation, which is conservatively quantified by assuming a bounding absorbtivity of unity. .

The FLUENT model consisting of the axisymmetric 3-D MPC space, the overpack, and the enveloping tank is schematically illustrated in Figure 4.4.13. A summary of the essential features of this model is presented in the following:

- A lower bound canister pressure of 105 psia under normal operating condition is postulated.
- Heat input due to insolation is applied to the top surface and the cylindrical surface of the overpack with a bounding maximum solar absorbtivity equal to 1.0.
- The heat generation in the MPC is assumed to be uniform in a horizontal plane for uniform loading and in each planar region (inner and outer) in regionalized storage. and varies in the axial direction in accordance with the axial power distribution listed in Chapter 2.
- The most disadvantageously placed cask (i.e., the one subjected to maximum radiative blockage), is modeled.
- The bottom surface of the overpack, in contact with the ISFSI pad , rejects heat through the pad to the constant temperature (77°F) earth below.

The finite-volume model constructed in this manner will produce an axisymmetric temperature distribution. The peak temperature will occur at the centerline and is expected to be above the axial location of peak heat generation. As will be shown in Subsection 4.4.2, the results of the finite-volume solution bear out these observations.

As discussed in Chapter 2 of this FSAR, the commercial nuclear fuel (CSF) can be stored in any of the MPCs in a uniform or regionalized configuration. The uniform storage arrangement implies that the heat emission rate of every fuel assembly (intact or canisterized) must be less than or equal to the MPC heat duty Q_d (Table 4.4.39) divided by the number of storage locations. In regionalized storage, the storage locations in the fuel basket are divided into two regions, denoted as Region 1 and Region 2, respectively. The cells located in the core region of the basket constitute Region 1. The peripheral cells define Region 2. The regionalization of the fuel basket must be configured such that a fuel assembly located in Region 1 is completely surrounded by Region 2 cells. Thus the Region 2 fuel serves as an effective blocker of gamma radiation emitted by the fuel located in Region 1. Regionalizing the storage of SNF such that the low heat emitting (therefore, lower dose emitting fuel) is located in Region 2 and the high heat emitting fuel is confined to the core of the basket (Region 1) provides an effective method to minimize the radiation emitted from canister at a storage

pad or in transport. In particular, the dose limits prescribed for HLW packages in 10CFR71 mandate that careful attention be paid to minimize dose emitted from an MPC. By regionalizing fuel storage in each basket, the aggregate dose at the site boundary can also be minimized which has a direct relevance to public health and safety. Figures 4.4.27, 4.4.28 and 4.4.29 depict Region 1 and 2 arrangements for MPC-32, MPC-24/24E, and MPC-68 respectively. In Table 4.4.30, the cell number allocations for regionalized storage are provided. In order to minimize the dose from the MPC the design basis heat emission rate of fuel in Region 2 must be as low as practical. Denoting N_1 and N_2 respectively as the number of storage cells in regions 1 and 2, it follows that the MPC heat load (Q) in regionalized storage is:

$$Q = N_1 q_1 + N_2 q_2$$

where q_1 and q_2 are the per assembly heat generation rates in Regions 1 and 2 respectively. For reasons that we explain below, when $q_1 = q_2$ (uniform storage), the value of Q is the maximum, say Q_d. In fact Q decreases monotonically as the ratio $X = q_1/q_2$ increases. Stated differently, the greater the disparity in the specific heat generation rates in Regions 1 and 2, the greater the penalty on the heat duty of the MPC. The reason that $Q < Q_d$ for regionalized storage§§ becomes clear by considering the temperature profile in a typical MPC. As can be seen from Figures 4.4.19 and 4.4.20, the maximum cladding temperature is at its peak value at the centerline (core) of the basket and decreases in the manner of a parabola with the radial co-ordinate. In other words, the design basis heat duty of the basket, Q_d, is governed by the permitted value of the Peak Cladding Temperature (PCT) which occurs at the centerline of the MPC. If the uniform storage (equal heat generation rate in each cell) is replaced by two regions and the MPC decay heat heat non-uniformly distributed in a manner that the core region is populated with more heat emissive fuel and outer region with relatively less emissive fuel, then clearly the center line temperature will be further elevated. To meet the PCT limit, the heat generation rate in Region 2 must reduced resulting in an overall reduction in the heat duty of the MPC. To be sure, increasing the rate of heat generation in the core region of the MPC has a salutary effect of increasing the thermosiphon driven upflow of helium through the core region. However, calculations show that the enhanced heat transfer through a more vigorous helium circulation is not enough to cancel out the elevation in PCT due to locally increased heat generation.

To summarize, the HI-STORM 100 System is evaluated for two fuel storage scenarios. In one scenario, designated as uniform loading, every basket cell is assumed to be occupied with fuel producing heat at the maximum rate. In another scenario, denoted as regionalized loading, a two-region fuel loading configuration is stipulated. The two regions are defined as an inner region (for storing hot fuel) and an outer region with low decay heat fuel physically enveloping the inner region. This scenario is depicted in Figure 4.4.25. The inner region is shown populated with fuel having a heat load of q_1 and the outer region with fuel of heat load q_2, where $q_1 > q_2$. . As depicted in Figures 4.4.27, 4.4.28 and 4.4.29 four central locations in the MPC-24 and MPC-24E, twelve inner cells in MPC-32 and 32 in MPC-68 are designated as inner region locations in the regionalized storage scenario.

---

§§ The reduction in Q due to regionalization is the direct outcome of placing comparatively hotter fuel in the core region. The opposite arrangement, wherein the hotter fuel is located in Region 2, would permit the heat duty of the MPC to be increased beyond Q_d. Placing more heat emissive fuel in the outer region, however would be anti-ALARA and therefore precluded as a storage option in the HI-STORM system.

As discussed previously, under regionalized storage the MPC heat load (Q), defined previously, is monotonically reduced as the disparity between the inner and outer region heat loads defined by the parameter X (equal to q_1/q_2) is increased. Following this logic path, the design heat loads under regionalized storage are defined by a mathematical function Q(X) that satisfies two attributes:

1) Q(X) reduces to uniform storage design heat load (Q_d) as X approaches unity
2) Q(X) monotonically reduces with increasing X

A function Q(X) satisfying (1) and (2) above is adopted for regionalized storage. The mathematical expression for Q(X) is:

$$Q(X) = \frac{2Q_d}{(1+X^a)}$$

where a is a constant (= 0.15) and X is permitted to vary from a minimum value of 1 to a maximum value of 6. Graphical plots of Q(X) are provided in Figures 4.4.31 and 4.4.32 for PWR and BWR MPCs, respectively. To confirm that the ISG-11, Rev. 2 cladding temperature limits are met for the design heat loads stated in the manner above, an array of HI-STORM thermal analyses are performed over a broad range of X (from 1 to 6) for each MPC type. From these analyses the variation in peak cladding temperature with X is obtained, results reported and evaluated in Subsection 4.4.2.

For the physical problem of regionalized storage, an infinite number of combinations of q_1 and q_2 exist that would satisfy the PCT limits of the stored fuel. To provide maximum flexibility to the ISFSI owner, an array of thermal analyses for each MPC type are performed to establish a continuum of regionalized storage options defined by the decay heats ratio X and the design heat load function Q(X). The permissible regionalized heat loads are computed by a four step process:

Step 1: Select a numerical value of X between 1 and 6.

Step 2: Compute maximum permissible MPC heat load:
$Q = 2Q_d/(1+X^a)$

Step 3: Compute permissible Region 2 SNF heat generation:
$q_2 = Q/(N_1X + N_2)$

Step 4: Compute permissible Region 1 SNF heat generation:
$q_1 = q_2X$

4.4.1.1.10 Subsection Intentionally Deleted

4.4.1.1.11 Subsection Intentionally Deleted

4.4.1.1.12 Subsection Intentionally Deleted

4.4.1.1.13 Subsection Intentionally Deleted

4.4.1.1.14 MPC Helium Backfill Pressure

The quantity of helium emplaced in the MPC cavity shall be sufficient to produce an operating pressure of 105 psia during normal storage at the design basis heat load. Thermal analyses performed on the different MPC designs indicate that this operating pressure requires a certain minimum helium backfill pressure (P_b) specified at a reference temperature (70°F). The minimum backfill pressure for each MPC type is provided in Table 4.4.37. A theoretical upper limit on the helium backfill pressure also exists and is defined by the design pressure of the MPC vessel (Table 2.2.1 in Chapter 2). The upper limit of P_b is reported in the last column of Table 4.4.37. To bound the minimum and maximum backfill pressures listed in Table 4.4.37 with a margin, a helium backfill specification for all MPCs is set forth in Table 4.4.38.

There are two methods available for ensuring that the appropriate quantity of helium has been placed in an MPC:

i. By pressure measurement
ii. By measuring the quantity of MPC helium backfill (in standard cubic feet)

The direct pressure measurement approach is more convenient if the FHD method of MPC drying is used. In this case, a certain quantity of helium is already in the MPC. Because the helium is mixed inside the MPC during the FHD operation, the temperature of the helium gas at the MPC's exit, along with the pressure provides a reliable means to compute the inventory of helium in the MPC cavity using pressure and temperature gages. The pressure in the FHD system, after adjustment from the measured helium temperature to the reference temperature (70°F), is adjusted by addition or withdrawal of helium such that it lies in the mid-range of the P_b specifications.

When vacuum drying is used as the method for MPC drying, then it is more convenient to fill the MPC by introducing a known quantity of helium (in standard cubic feet) by measuring the quantity of helium introduced using a calibrated mass flow meter. The required quantity of helium (F) is computed by the product of net free nominal volume and helium specific volume at a target pressure that lies in the mid-range of the P_b specifications.

The net free nominal volume of the MPC is obtained by subtracting A from B, where

A = MPC cavity volume in the absence of contents (fuel and non-fuel hardware) computed from nominal design dimensions

B = Total volume of the contents (fuel including DFCs, if used) based on nominal design dimensions

Using commercially available mass flow totalizers, an MPC cavity is filled with the computed quantity of helium (F).

4.4.1.2 Test Model

A detailed analytical model for thermal design of the HI-STORM System was developed using the FLUENT CFD code and the industry standard ANSYS modeling package, as discussed in Subsection 4.4.1.1. As discussed throughout this chapter and specifically in Section 4.4.6, the analysis incorporates significant conservatisms so as to compute bounding fuel cladding temperatures. Furthermore, compliance with specified limits of operation is demonstrated with adequate margins. In view of these considerations, the HI-STORM System thermal design complies with the thermal criteria set forth in the design basis (Sections 2.1 and 2.2) for long-term storage under normal conditions. Additional experimental verification of the thermal design is therefore not required.

4.4.2 Maximum Temperatures

All four principal MPC-basket designs developed for the HI-STORM System have been analyzed to determine temperature distributions under long-term normal storage conditions, and the results summarized in this subsection. A cross-reference of HI-STORM thermal analyses at other conditions with associated subsection of the FSAR summarizing obtained results is provided in Table 4.4.22. The MPC baskets are considered to be fully loaded with design basis PWR or BWR fuel assemblies, as appropriate. The systems are arranged in an ISFSI array and subjected to design basis normal ambient conditions with insolation. Both uniform loading and regionalized loading scenarios are analyzed. For uniform loading, the MPC thermal payload is assumed to be at the design maximum (Q_d defined in Section 4.0). For regionalized loading, the storage scenarios are defined by the ratio $X = q_1/q_2$ and an associated maximum heat load Q(X) defined in Subsection 4.4.1.1.9. The HI-STORM System is analyzed under regionalized storage for X ranging from 1 to 6.

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 then 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 in-plane thermal conductivity for an equivalent homogenous region. It is important to note that the result of this analysis is a function of thermal conductivity versus temperature. This function for fuel thermal conductivity is then input to the fuel basket effective in-plane thermal conductivity calculation described in Subsection 4.4.1.1.4. This calculation uses a finite-element methodology, wherein each fuel cell region containing multiple finite-elements has temperature-varying thermal conductivity properties. The resultant temperature-varying fuel basket thermal conductivity computed by this basket-fuel composite model is then input to the fuel basket region of the FLUENT cask model.

Because the FLUENT cask model incorporates the results of the fuel basket submodel, which in turn incorporates the fuel assembly submodel, the peak temperature reported from the FLUENT model is the peak temperature in any component. In a dry storage cask, the hottest components are the fuel

assemblies. It should be noted that, because the fuel assembly models described in Subsection 4.4.1.1.2 include the fuel pellets, the FLUENT calculated peak temperatures reported in Tables 4.4.9, 4.4.10 and 4.4.26 are actually peak pellet centerline temperatures which bound the peak cladding temperatures, and are therefore conservatively reported as the cladding temperatures.

Applying the radiative blocking factor applicable for the worst case cask location, representative axial temperature plots of the fuel cladding are shown in Figures 4.4.16 and 4.4.17 for a MPC-24 and a MPC-68 to depict the thermosiphon effect in PWR and BWR SNF. From these plots, the upward movement of the hot spot is quite evident. As discussed in this chapter, these calculated temperature distributions incorporate many conservatisms. The maximum fuel clad temperatures for zircaloy clad fuel assemblies are listed in Tables 4.4.9, 4.4.10 and 4.4.26 which also summarize maximum calculated temperatures in different parts of the MPCs and HI-STORM overpack (Table 4.4.36).

In Figures 4.4.19 and 4.4.20, respectively, representative radial temperature distributions in an MPC-24 and an MPC-68 in the horizontal plane where the maximum fuel cladding temperature occurs are graphed. Finally, axial variations of the ventilation air temperatures and that of the overpack inner shell surface are graphed in Figure 4.4.26 for a bounding MPC. .

As stated in Subsection 4.4.1.1.9 an array of HI-STORM analyses are performed for the regionalized storage option to characterize the variation of peak cladding temperature (PCT) with the decay heats ratio X. The PCT variations are graphed in Figures 4.4.33 through 4.4.35 over the permitted range of X. From these graphs it is evident that the PCT is below the ISG-11, Rev. 2 limits for all MPCs.

The following additional observations can be derived by inspecting the temperature field obtained from the finite volume analysis:

- The fuel cladding temperatures are below the regulatory limit (ISG-11, Rev. 2) under all storage scenarios (uniform and regionalized) and all MPCs.
- The maximum temperature of the basket structural material is within the stipulated design temperature.
- The maximum temperature of the neutron absorber is below the design temperature limit.
- The maximum temperatures of the MPC pressure boundary materials are well below their respective ASME Code limits.
- The maximum temperatures of concrete are within the NRC's recommended limits [4.4.10] (See Table 4.3.1.)

The calculated temperatures are based on a series of analyses, described previously in this chapter, that incorporate many conservatisms. A list of the significant conservatisms is provided in Subsection 4.4.6 with a follow up discussion in Appendix 4.B. As such, the calculated temperatures are upper bound values that would exceed actual temperatures.

The above observations lead us to conclude that the temperature field in the HI-STORM System with a fully loaded MPC containing design-basis heat emitting SNF complies with all regulatory and industry temperature limits. In other words, the thermal environment in the HI-STORM System will be conducive to long-term safe storage of spent nuclear fuel.

4.4.3 Minimum Temperatures

In Table 2.2.2 of this report, the minimum ambient temperature condition for the HI-STORM storage overpack and MPC is specified to be -40°F. If, conservatively, a zero decay heat load with no solar input is applied to the stored fuel assemblies, then every component of the system at steady state would be at a temperature of -40°F. All HI-STORM storage overpack and MPC materials of construction will satisfactorily perform their intended function in the storage mode at this minimum temperature condition. Structural evaluations in Chapter 3 show the acceptable performance of the overpack and MPC steel and concrete materials at low service temperatures. Criticality and shielding evaluations (Chapters 5 and 6) are unaffected by temperature.

4.4.4 Maximum Internal Pressure

The MPC is initially filled with dry helium after fuel loading and drying prior to installing the MPC closure ring. During normal storage, the gas temperature within the MPC rises to its maximum operating basis temperature as determined based on the thermal analysis methodology described earlier. The gas pressure inside the MPC will also increase with rising temperature. The pressure rise is determined based on the ideal gas law, which states that the absolute pressure of a fixed volume of gas is proportional to its absolute temperature. Tables 4.4.12, 4.4.13, 4.4.24, and 4.4.25 present summaries of the calculations performed to determine the net free volume in the MPC-24, MPC-68, MPC-32, and MPC-24E, respectively.

The MPC maximum gas pressure is considered for a postulated accidental release of fission product gases caused by fuel rod rupture. For these fuel rod rupture conditions, the amounts of each of the release gas constituents in the MPC cavity are summed and the resulting total pressures determined from the Ideal Gas Law. Based on fission gases release fractions (per NUREG 1536 criteria [4.4.10]), net free volume and initial fill gas pressure, the bounding maximum gas pressures with 1% (normal), 10% (off-normal) and 100% (accident condition) rod rupture are given in Table 4.4.14. The maximum gas pressures listed in Table 4.4.14 are all below the MPC internal design pressure listed in Table 2.2.1.

The inclusion of PWR non-fuel hardware (BPRA control elements and thimble plugs) to the PWR baskets influences the MPC internal pressure through two distinct effects. The presence of non-fuel hardware increases the effective basket conductivity, thus enhancing heat dissipation and lowering fuel temperatures as well as the temperature of the gas filling the space between fuel rods. The gas volume displaced by the mass of non-fuel hardware lowers the cavity free volume. These two effects, namely, temperature lowering and free volume reduction, have opposing influence on the MPC cavity pressure. The first effect lowers gas pressure while the second effect raises it. In the HI-STORM thermal analysis, the computed temperature field (with non-fuel hardware excluded) has

been determined to provide a conservatively bounding temperature field for the PWR baskets (MPC-24, MPC-24E, and MPC-32). The MPC cavity free space is computed based on volume displacement by the heaviest fuel (bounding weight) with non-fuel hardware included. This approach ensures conservative bounding pressures.

During in-core irradiation of BPRAs, neutron capture by the B-10 isotope in the neutron absorbing material produces helium. Two different forms of the neutron absorbing material are used in BPRAs: Borosilicate glass and B_4C in a refractory solid matrix (Al_2O_3). Borosilicate glass (primarily a constituent of Westinghouse BPRAs) is used in the shape of hollow pyrex glass tubes sealed within steel rods and supported on the inside by a thin-walled steel liner. To accommodate helium diffusion from the glass rod into the rod internal space, a relatively high void volume (~40%) is engineered in this type of rod design. The rod internal pressure is thus designed to remain below reactor operation conditions (2,300 psia and approximately 600°F coolant temperature). The B_4C- Al_2O_3 neutron absorber material is principally used in B&W and CE fuel BPRA designs. The relatively low temperature of the poison material in BPRA rods (relative to fuel pellets) favors the entrapment of helium atoms in the solid matrix.

Several BPRA designs are used in PWR fuel that differ in the number, diameter, and length of poison rods. The older Westinghouse fuel (W-14x14 and W-15x15) has used 6, 12, 16, and 20 rods per assembly BPRAs and the later (W-17x17) fuel uses up to 24 rods per BPRA. The BPRA rods in the older fuel are much larger than the later fuel and, therefore, the B-10 isotope inventory in the 20-rod BPRAs bounds the newer W-17x17 fuel. Based on bounding BPRA rods internal pressure, a large hypothetical quantity of helium (7.2 g-moles/BPRA) is assumed to be available for release into the MPC cavity from each fuel assembly in the PWR baskets. The MPC cavity pressures (including helium from BPRAs) are summarized in Table 4.4.14.

4.4.5 Maximum Thermal Stresses

Thermal stress in a structural component is the resultant sum of two factors, namely: (i) restraint of free end expansion and (ii) non-uniform temperature distribution. To minimize thermal stresses in load bearing members, the HI-STORM System is engineered with adequate gaps to permit free thermal expansion of the fuel basket and MPC in axial and radial directions. In this sub-section, differential thermal expansion calculations are performed to demonstrate that engineered gaps in the HI-STORM System are adequate to accommodate thermal expansion. . To facilitate structural integrity evaluations, temperature distributions are provided herein (Tables 4.4.9, 4.4.10 and 4.4.26).

As stated above, the HI-STORM System is engineered with gaps for the fuel basket and MPC to thermally expand without restraint of free end expansion. Differential thermal expansion of the following gaps are evaluated:

a) Fuel Basket-to-MPC Radial Gap
b) Fuel Basket-to-MPC Axial Gap
c) MPC-to-Overpack Radial Gap
d) MPC-to-Overpack Axial Gap

To demonstrate that the fuel basket and MPC are free to expand without restraint, it is required to show that differential thermal expansion from fuel heat up is less than the as-built gaps that exist in the HI-STORM System. For this purpose a suitably bounding temperature profile (T(r)) for the fuel basket is established in Figure 4.4.30 wherein the center temperature (TC) is set at the limit (752°F) for fuel cladding (conservatively bounding assumption) and the basket periphery (TP) conservatively postulated at an upperbound of 600°F (See Tables 4.4.9, 4.4.10 and 4.4.26 for the maximum computed basket periphery temperatures). To maximize the fuel basket differential expansion, the basket periphery-to-MPC shell temperature difference is conservatively maximized ($\Delta T = 175°F$). From the bounding temperature profile T(r) and ΔT, the mean fuel basket temperature (T1) and MPC shell temperature (T2) are computed as follows:

$$T1 = \frac{\int_0^1 rT(r)dr}{\int_0^1 rdr}$$

$$= 676°F$$

$$T2 = TP - \Delta T$$
$$= 425°F$$

The differential radial growth of the fuel basket (Y1) from an initial reference temperature (To = 70°F) is computed as:

$$Y1 = R*\{A1*(T1 - To) - A2*(T2 - To)\}$$

where:

R = Basket radius (conservatively assumed to be the MPC radius)

A1, A2 = Coefficients of thermal expansion for fuel basket and MPC shell at T1 and T2 respectively for Alloy-X (Chapter 1 and Table 3.3.1)

For computing the relative axial growth of the fuel basket in the MPC, bounding temperatures for the fuel basket (TC) and MPC shell temperature T2 computed above (assuming a maximum basket periphery-to-MPC shell temperature differential) are adopted. The differential expansion is computed by a formula similar to the one for radial growth after replacing R with basket height (H) which is conservatively assumed to be that of the MPC cavity.

For computing the radial and axial MPC-to-overpack differential expansions, the MPC shell is postulated at its design temperature (Chapter 2, Table 2.2.3) and thermal expansion of the overpack is ignored. Even with the conservative computation of the differential expansions in the manner of the foregoing, it is evident from the data compiled in Table 4.4.40 that the differential expansions are a fraction of their respective gaps.

4.4.6 Evaluation of System Performance for Normal Conditions of Storage

The HI-STORM System thermal analysis is based on a detailed and complete heat transfer model that conservatively accounts for all modes of heat transfer in various portions of the MPC and overpack. A comprehensive discussion of HI-STORM conservatisms is provided in Appendix 4.B. A numbered list of the many thermal modeling conservatisms for long-term storage is provided hereunder:

1. The most severe levels of environmental factors for long-term normal storage, which are an ambient temperature of 80°F and 10CFR71 insolation levels, were coincidentally imposed on the system.

2. The most adversely located*** HI-STORM System in an ISFSI array was considered for analysis.

3. The thermosiphon effect in the MPC, which is intrinsic to the HI-STORM fuel basket design is included in the thermal analyses, using a conservative model, by assuming the top and bottom plenum spaces for helium flow to be less than actual values.

4. No credit was considered for contact between fuel assemblies and the MPC basket wall or between the MPC basket and the basket supports. The fuel assemblies and MPC basket were conservatively considered to be in concentric alignment.

5. The MPC is assumed to be loaded with an SNF which has the maximum resistance to heat dissipation (planar and axial) and helium flow of all fuel types in its category (BWR or PWR), as applicable.

6. The design basis maximum decay heat loads are used for all thermal-hydraulic analyses. For casks loaded with fuel assemblies having decay heat generation rates less than design basis, additional thermal margins of safety will exist.

7. Conservative bounding flow resistance factors are employed to simulate flow through MPC 3-D continuum.

8. Axial heat transfer through fuel pellets is ignored.

9. Heat dissipation by grid spacers, top & bottom fittings is ignored.

10. Insolation heating assumed with a bounding absorbtivity (=1.0).

11. A margin between the computed peak cladding temperature and 400°C limit is provided for all MPCs.

*** In an ISFSI array, HI-STORM Overpacks at interior locations are relatively more disadvantaged in their lateral access to ambient air and in their effectiveness to radiate heat to environment. To bound these effects, a reference cask is enclosed in a hypothetical reflecting cylinder as described in Section 4.4.1.1.9 and Appendix 4B.

12. The computed values of MPC axial conductivities are understated in the thermal models.

13. The flux trap flow areas (MPC-24/24E designs) are ignored in the MPC thermosiphon cooling models.

Temperature distribution results obtained from this highly conservative thermal model show that the maximum fuel cladding temperature limits are met with adequate margins. Expected margins during normal storage will be much greater due to the many conservative assumptions incorporated in the analysis. The long-term impact of decay heat induced temperature levels on the HI-STORM System structural and neutron shielding materials is considered to be negligible. The maximum local MPC basket temperature level is below the recommended limits for structural materials in terms of susceptibility to stress, corrosion and creep-induced degradation. Furthermore, stresses induced due to imposed temperature gradients are within Code limits. Therefore, it is concluded that the HI-STORM System thermal design is in compliance with 10CFR72 requirements.

Table 4.4.1

SUMMARY OF PWR FUEL ASSEMBLY EFFECTIVE THERMAL CONDUCTIVITIES

Fuel	@ 200°F (Btu/ft-hr-°F)	@ 450°F (Btu/ft-hr-°F)	@ 700°F (Btu/ft-hr-°F)
W - 17×17 OFA	0.182	0.277	0.402
W - 17×17 Standard	0.189	0.286	0.413
W - 17×17 Vantage	0.182	0.277	0.402
W - 15×15 Standard	0.191	0.294	0.430
W - 14×14 Standard	0.182	0.284	0.424
W - 14×14 OFA	0.175	0.275	0.413
B&W - 17×17	0.191	0.289	0.416
B&W - 15×15	0.195	0.298	0.436
CE - 16×16	0.183	0.281	0.411
CE - 14×14	0.189	0.293	0.435
HN† - 15×15 SS	0.180	0.265	0.370
W - 14×14 SS	0.170	0.254	0.361
B&W-15x15 Mark B-11	0.187	0.289	0.424
CE-14x14 (MP2)	0.188	0.293	0.434
IP-1 (14x14) SS	0.125	0.197	0.293

† Haddam Neck Plant B&W or Westinghouse stainless steel clad fuel assemblies.

Table 4.4.2

SUMMARY OF BWR FUEL ASSEMBLY EFFECTIVE THERMAL CONDUCTIVITIES

Fuel	@ 200°F (Btu/ft-hr-°F)	@ 450°F (Btu/ft-hr-°F)	@ 700°F (Btu/ft-hr-°F)
Dresden 1 - 8×8†	0.119	0.201	0.319
Dresden 1 - 6×6†	0.126	0.215	0.345
GE - 7×7	0.171	0.286	0.449
GE - 7×7R	0.171	0.286	0.449
GE - 8×8	0.168	0.278	0.433
GE - 8×8R	0.166	0.275	0.430
GE10 - 8×8	0.168	0.280	0.437
GE11 - 9×9	0.167	0.273	0.422
AC††-10×10 SS	0.152	0.222	0.309
Exxon-10×10 SS	0.151	0.221	0.308
Damaged Dresden-1 8×8† (in a Holtec damaged fuel container)	0.107	0.169	0.254
Humboldt Bay-7x7†	0.127	0.215	0.343
Dresden-1 Thin Clad 6x6†	0.124	0.212	0.343
Damaged Dresden-1 8x8 (in TN D-1 canister)†	0.107	0.168	0.252
8x8 Quad+ Westinghouse†	0.164	0.276	0.435

† Cladding temperatures of low heat emitting Dresden (intact and damaged) SNF in the HI-STORM System will be bounded by design basis fuel cladding temperatures. Therefore, these fuel assembly types are excluded from the list of fuel assemblies (zircaloy clad) evaluated to determine the most resistive SNF type.

†† Allis-Chalmers stainless steel clad fuel assemblies.

Table 4.4.3

MPC BASKET PLANAR AND AXIAL[†††] THERMAL CONDUCTIVITY VALUES

Basket	@200°F (Btu/ft-hr-°F)	@450°F (Btu/ft-hr-°F)	@700°F (Btu/ft-hr-°F)
MPC-24/24E	1.127 (planar) 2.427 (axial)	1.535 (planar) 2.666 (axial)	2.026 (planar) 2.870 (axial)
MPC-68	1.025 (planar) 2.186 (axial)	1.257 (planar) 2.379 (axial)	1.500 (planar) 2.548 (axial)
MPC-32	0.964 (planar) 1.773 (axial)	1.214 (planar) 1.933 (axial)	1.486 (planar) 2.079 (axial)

††† The axial thermal conductivity values reported herein are understated 5%.

Table 4.4.4 through 4.4.8

[INTENTIONALLY DELETED]

Table 4.4.9

HI-STORM† SYSTEM LONG-TERM NORMAL
UNIFORM STORAGE MAXIMUM TEMPERATURES
(MPC-24/24E‡‡‡ BASKET)

Component	Normal Condition Temp. (°F)	Long-Term Temperature Limit (°F)
Fuel Cladding	688	752§§§
MPC Basket****	633	725†††
Basket Periphery	569	725†††
MPC Shell	466	500

† Bounding overpack temperatures are provided in Table 4.4.36.

‡‡‡ Limiting values reported.
§§§ The temperature limit is in accordance with ISG-11, Rev. 2.
**** The maximum neutron absorber temperature is essentially the same as the maximum MPC basket temperature reported in this table.
††† The ASME Code allowable temperature of the fuel basket Alloy X materials is 800°F. This lower temperature limit is imposed to add additional conservatism to the analysis of the HI-STORM System.

Table 4.4.10

HI-STORM† SYSTEM LONG-TERM NORMAL
UNIFORM STORAGE MAXIMUM TEMPERATURES
(MPC-68 BASKET)

Component	Normal Condition Temp. (°F)	Long-Term Temperature Limit (°F)
Fuel Cladding	731	752††
MPC Basket††††	706	725†††
Basket Periphery	579	725†††
MPC Shell	470	500

† Bounding overpack temperatures are provided in Table 4.4.36.

†† The temperature limit is in accordance with ISG-11, Rev. 2.

†††† The maximum neutron absorber temperature is essentially the same as the maximum MPC basket temperature reported in this table.

††† The ASME Code allowable temperature of the fuel basket Alloy X materials is 800°F. This lower temperature limit is imposed to add additional conservatism to the analysis of the HI-STORM System.

Table 4.4.11

INTENTIONALLY DELETED

Table 4.4.12

SUMMARY OF MPC-24 FREE VOLUME CALCULATIONS

Item	Volume (ft^3)
Cavity Volume	367
Basket Metal Volume	45
Bounding Fuel Assemblies Volume	79
Basket Supports and Fuel Spacers Volume	7
Net Free Volume	236 (6683 liters)

Table 4.4.13

SUMMARY OF MPC-68 FREE VOLUME CALCULATIONS

Item	Volume (ft^3)
Cavity Volume	367
Basket Metal Volume	35
Bounding Fuel Assemblies Volume	93
Basket Supports and Fuel Spacers Volume	12
Net Free Volume	227 (6428 liters)

Table 4.4.14
SUMMARY OF MPC CONFINEMENT BOUNDARY PRESSURES†
FOR LONG-TERM STORAGE

Condition	Pressure (psig)‡‡‡‡
MPC-24:	
Initial backfill (at 70°F)	48.8
Normal condition	97.9
With 1% rods rupture	98.6
With 10% rods rupture	104.8
With 100% rods rupture	167.1
MPC-68:	
Initial backfill (at 70°F)	48.8
Normal condition	97.3
With 1% rods rupture	97.7
With 10% rods rupture	101.7
With 100% rods rupture	141.5
MPC-32:	
Initial backfill (at 70°F)	48.8
Normal Condition	97.0
With 1% rods rupture	98.0
With 10% rods rupture	106.6
With 100% rods rupture	192.3
MPC-24E:	
Initial backfill (at 70°F)	48.8
Normal Condition	97.9
With 1% rods rupture	98.6
With 10% rods rupture	105.0
With 100% rods rupture	169.3

† Per NUREG-1536, pressure analyses with postulated rods rupture (including BPRA rods for PWR fuel) is performed with release of 100% of the ruptured fuel rod fill gas and 30% of the significant radioactive gaseous fission products.

‡‡‡‡ The pressures reported in this table are computed assuming the helium backfill pressure is at it's upperbound limit (Table 4.4.38).

Table 4.4.15 through 4.4.21

[INTENTIONALLY DELETED]

Table 4.4.22
MATRIX OF HI-STORM SYSTEM THERMAL EVALUATIONS

Scenario	Description	Ultimate Heat Sink	Analysis Type	Principal Input Parameters	Results in FSAR Subsection
1	Long Term Normal	Ambient	SS	N_T, Q_D, ST, SC, I_O	4.4.2
2	Off-Normal Environment	Ambient	SS(B)	O_T, Q_D, ST, SC, I_O	11.1.2
3	Extreme Environment	Ambient	SS(B)	E_T, Q_D, ST, SC, I_O	11.2.15
4	Partial Ducts Blockage	Ambient	SS(B)	N_T, Q_D, ST, SC, $I_{1/2}$	11.1.4
5	Ducts Blockage Accident	Overpack	TA	N_T, Q_D, ST, SC, I_C	11.2.13
6	Fire Accident	Overpack	TA	Q_D, F	11.2.4
7	Tip Over Accident	Overpack	AH	Q_D	11.2.3
8	Debris Burial Accident	Overpack	AH	Q_D	11.2.14

Legend:

N_T - Maximum Annual Average (Normal) Temperature (80°F)
O_T - Off-Normal Temperature (100°F)
E_T - Extreme Hot Temperature (125°F)
Q_D - Design Basis Maximum Heat Load
SS - Steady State
SS(B) - Bounding Steady State
TA - Transient Analysis
AH - Adiabatic Heating

I_O - All Inlet Ducts Open
$I_{1/2}$ - Half of Inlet Ducts Open
I_C - All Inlet Ducts Closed
ST - Insolation Heating (Top)
SC - Insolation Heating (Curved)
F - Fire Heating (1475°F)

Table 4.4.23

[INTENTIONALLY DELETED]

Table 4.4.24

SUMMARY OF MPC-32 FREE VOLUME CALCULATIONS

Item	Volume (ft^3)
Cavity Volume	367
Basket Metal Volume	25
Bounding Free Assemblies Volume	106
Basket Supports and Fuel Spacers Volume	9
Net Free Volume	227 (6428 liters)

Table 4.4.25

SUMMARYOF MPC-24E FREE VOLUME CALCULATIONS

Item	Volume (ft^3)
Cavity Volume	367
Basket Metal Volume	52
Bounding Fuel Assemblies Volume	79
Basket Supports and Fuel Spacers Volume	7
Net Free Volume	229 (6484 liters)

Table 4.4.26

HI-STORM† SYSTEM LONG-TERM NORMAL UNIFORM STORAGE MAXIMUM TEMPERATURES (MPC-32 BASKET)

Component	Normal Condition Temp. (°F)	Long-Term Temperature Limit (°F)
Fuel Cladding	662	752††
MPC Basket§§§§	621	725†††
Basket Periphery	565	725†††
MPC Shell	463	500

† Bounding overpack temperatures are provided in Table 4.4.36.

†† The temperature limit is in accordance with ISG-11, Rev. 2.

§§§§ The maximum neutron absorber temperature is essentially the same as the maximum MPC basket temperature reported in this table.

††† The ASME Code allowable temperature of the fuel basket Alloy X materials is 800°F. This lower temperature limit is imposed to add additional conservatism in the analysis of the HI-STORM Systems.

Table 4.4.27 through 4.4.29

[INTENTIONALLY DELETED]

Table 4.4.30

MPC REGIONALIZED STORAGE CONFIGURATIONS

MPC Type	Inner Region Assemblies (n_1)	Outer Region Assemblies (n_2)	Depicted in Figure
MPC-24/24E	4	20	Figure 4.4.28
MPC-32	12	20	Figure 4.4.27
MPC-68	32	36	Figure 4.4.29

Table 4.4.31 through 4.4.35

[INTENTIONALLY DELETED]

Table 4.4.36

BOUNDING LONG-TERM NORMAL STORAGE
HI-STORM OVERPACK TEMPERATURES

Component†	Local Section Temperature†† (°F)	Long-Term Temperature Limit (°F)
Inner shell	243	350
Outer shell	183	350
Lid bottom plate	412	450
Lid top plate	218	450
MPC pedestal plate	180	350
Baseplate	108	350
Radial shield	193	300

† See Figure 1.2.8 for a description of HI-STORM components.
†† Section temperature is defined as the through-thickness average temperature.

Table 4.4.37

ACCEPTABLE RANGE OF MPC HELIUM BACKFILL PRESSURE*****

MPC	Minimum Backfill Pressure [psig]	Maximum Backfill Pressure [psig]
MPC-32	45.0	50.5
MPC-24/24E	44.5	50.0
MPC-68	44.9	50.4

***** The pressures tabulated herein are at a reference gas temperature of 70°F.

Table 4.4.38

MPC HELIUM BACKFILL PRESSURE SPECIFICATIONS

Item	Specification
Minimum Pressure	45.2 psig @ 70°F Reference Temperature
Maximum Pressure	48.8 psig @ 70°F Reference Temperature

Table 4.4.39

DESIGN MAXIMUM HEAT LOADS FOR THE HI-STORM SYSTEM

MPC Type	Heat Load (Q_d) [kW]
PWR MPCs	38
BWR MPCs	35.5

Table 4.4.40

SUMMARY OF HI-STORM DIFFERENTIAL EXPANSIONS

Gap Description	Cold Gap (U), inch	Differential Expansion (V), inch	Is Free Expansion Criteria Satisfied (i.e. U > V)
Fuel Basket-to-MPC Radial Gap	0.1875	0.096	Yes
Fuel Basket-to-MPC Axial Gap	1.25	0.499	Yes
MPC-to-Overpack Radial Gap	0.5	0.139	Yes
MPC-to-Overpack Axial Gap	1.0	0.771	Yes

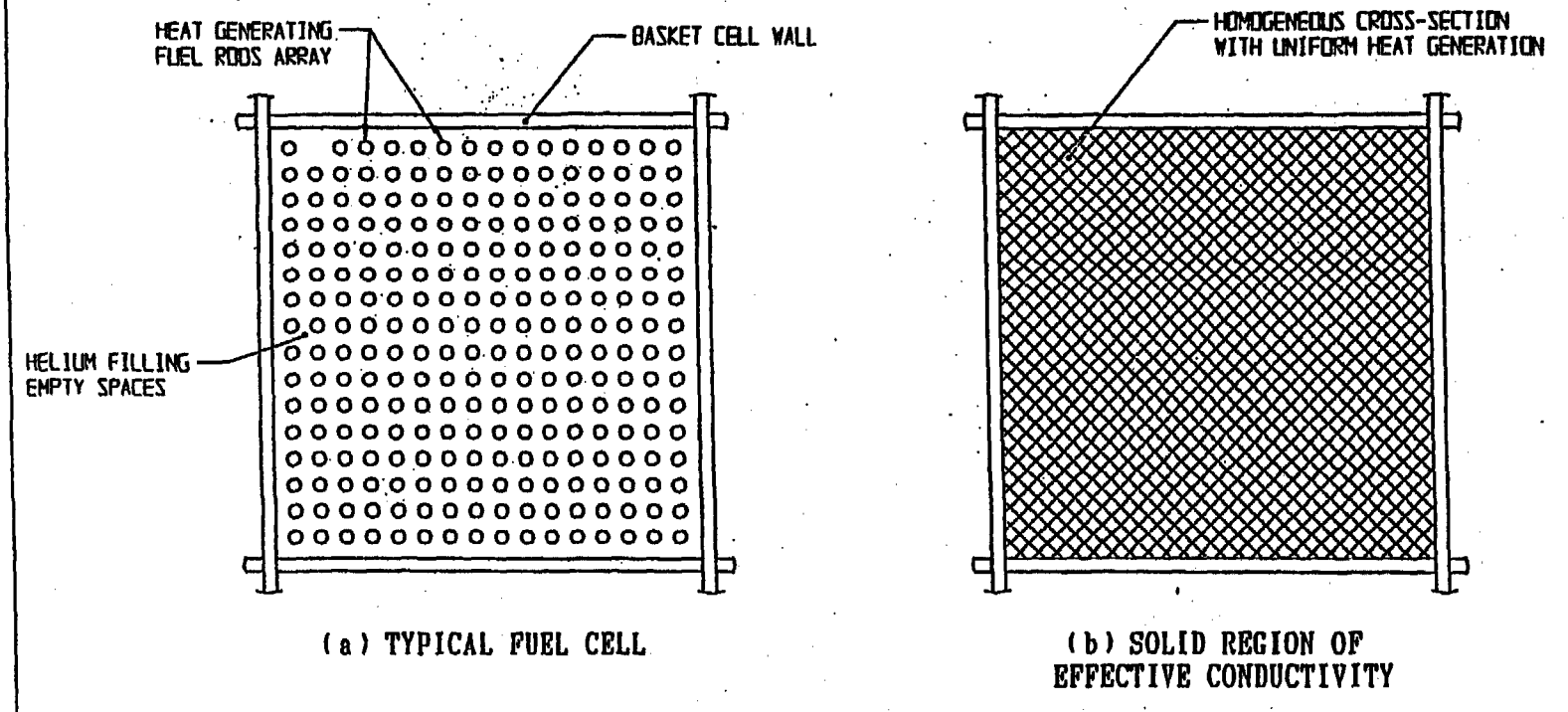
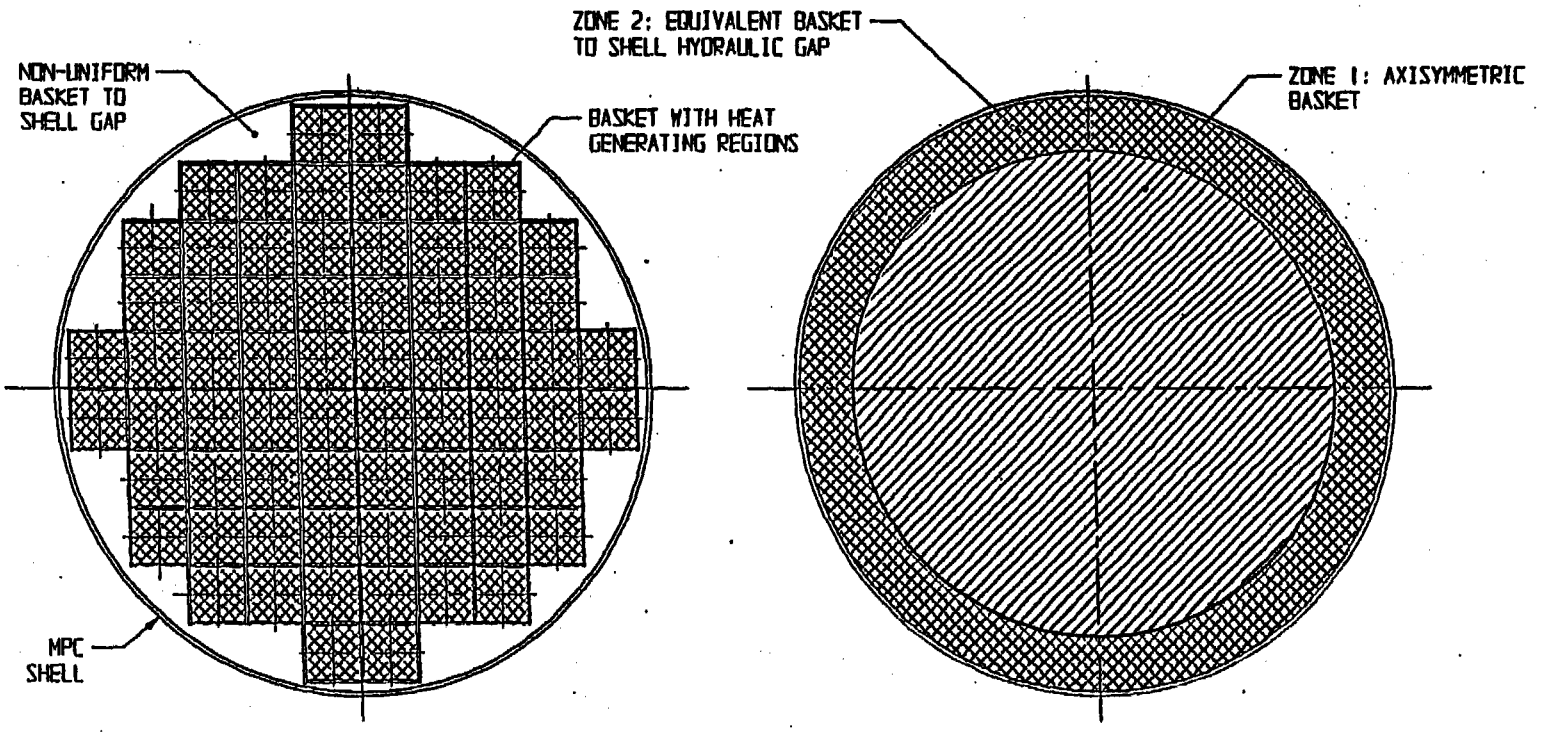


FIGURE 4.4.1; HOMOGENIZATION OF THE STORAGE CELL CROSS-SECTION

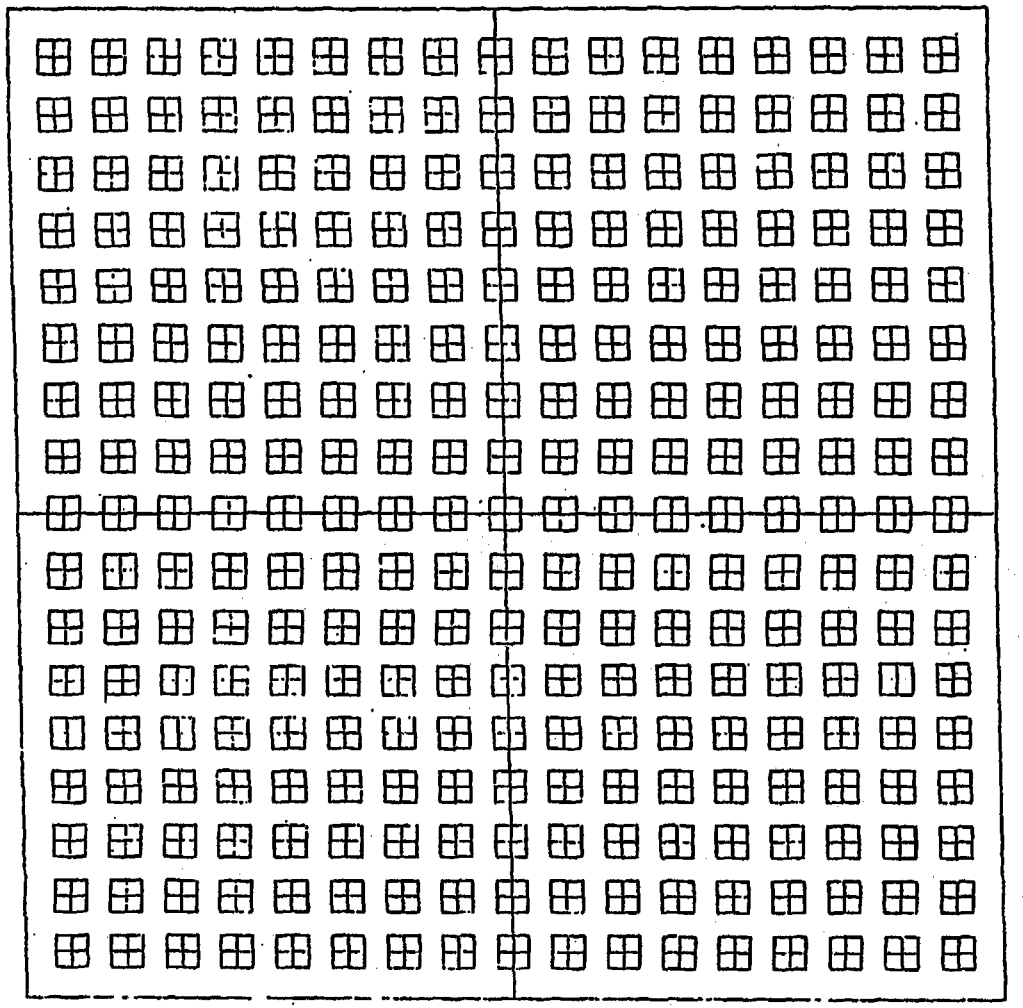


(a) TYPICAL MPC CROSS-SECTION

(b) EQUIVALENT TWO ZONE MODEL

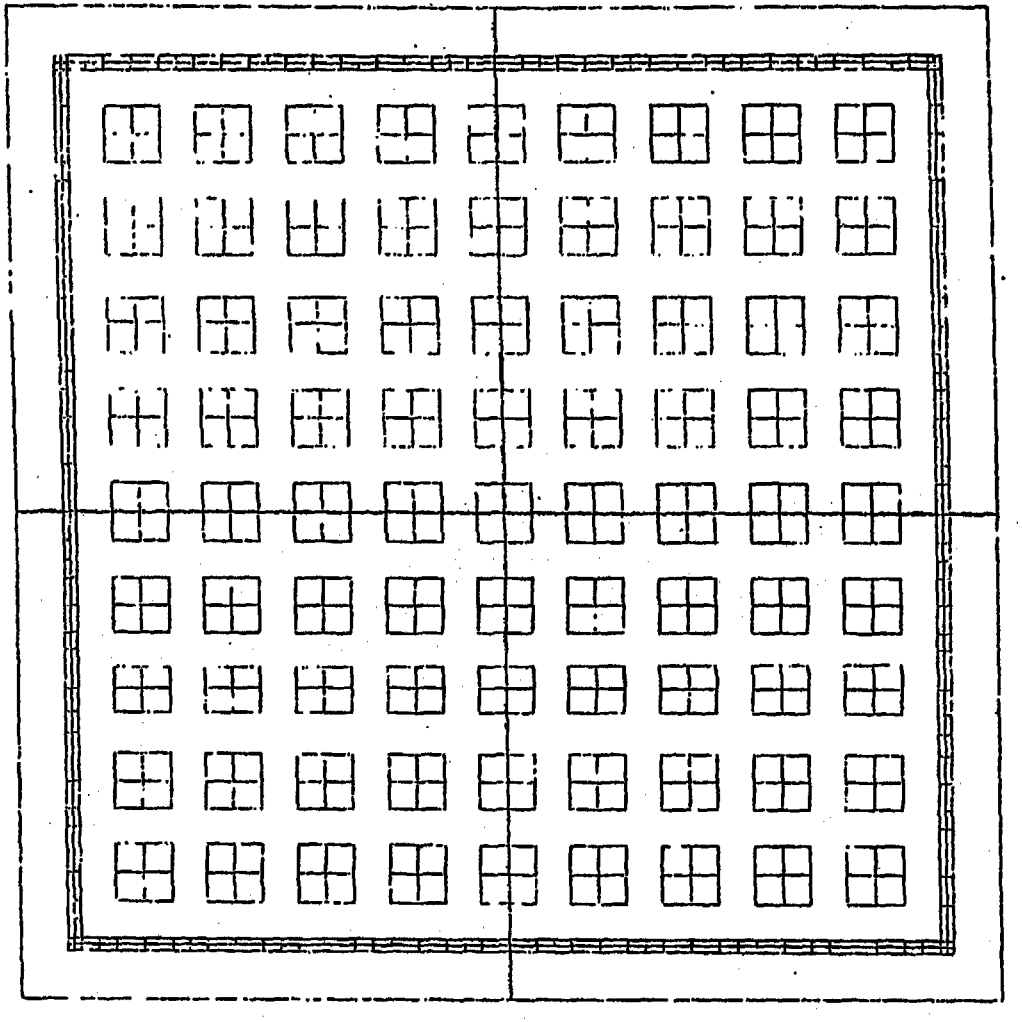
FIGURE 4.4.2; MPC CROSS-SECTION REPLACED WITH AN EQUIVALENT TWO ZONE AXISYMMETRIC BODY

\PROJECTS\5014\HI951312\CH_4\4_4_2



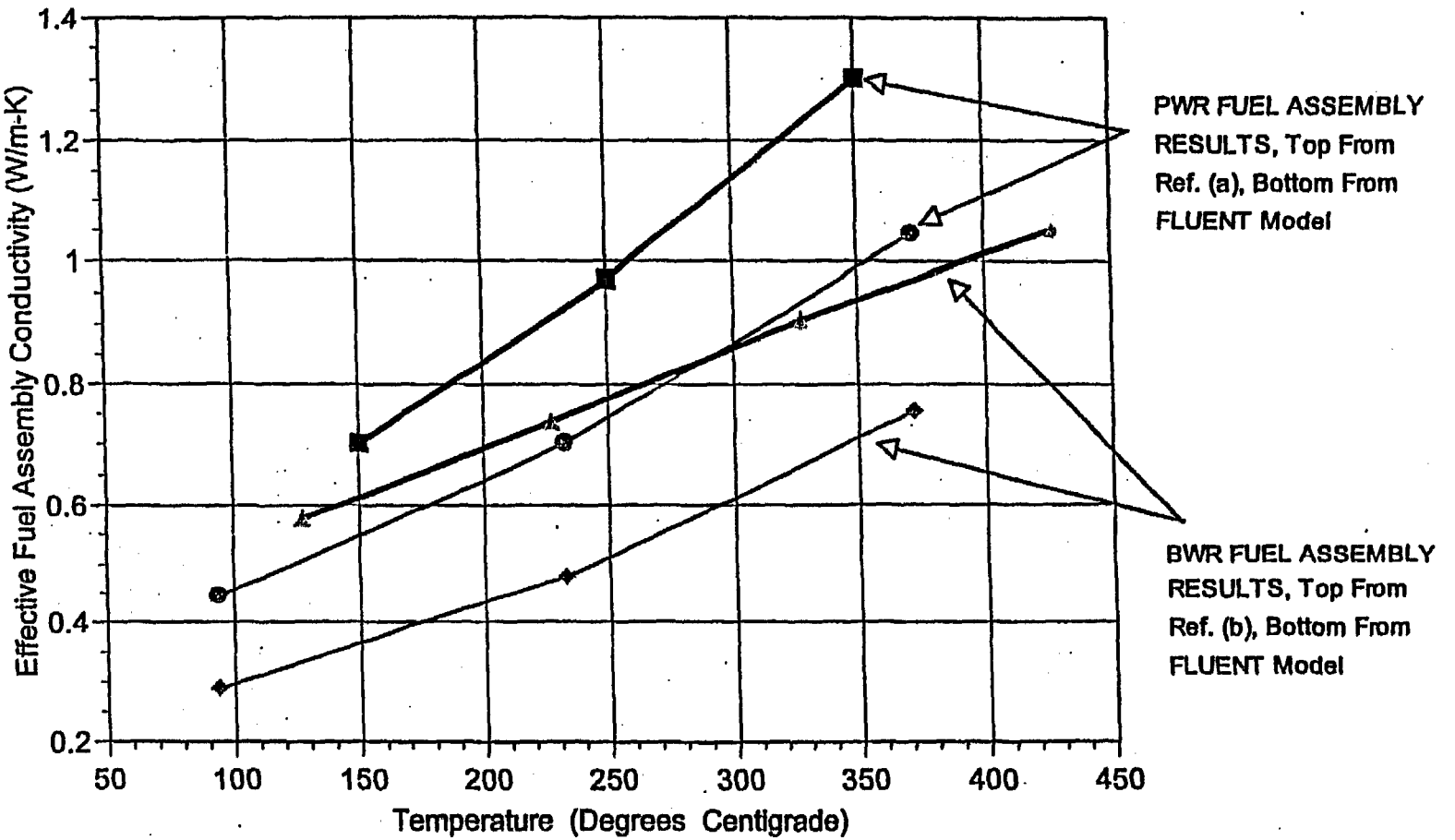
Y Z X	W17X17 OFA FUEL ASSEMBLY MODEL	Nov 11 1997 Fluent 4.32 Fluent Inc.

FIGURE 4.4.3: WESTINGHOUSE 17x17 OFA PWR FUEL ASSEMBLY MODEL



Y X Z	GE11-9X9 CHANNELED FUEL ASSEMBLY MODEL	Nov 11 1997 Fluent 4.32 Fluent Inc.

FIGURE 4.4.4: GENERAL ELECTRIC 9x9 BWR FUEL ASSEMBLY MODEL



(a) "Determination of SNF Peak Temperatures in the Waste Package", Bahney & Doering, *HLRWM Sixth Annual Conf.*, Pages 671-673, (April 30 - May 5, 1995).

(b) "A Method for Determining the Spent-Fuel Contribution to Transport Cask Containment Requirements", *Sandia Report SAND90-2406*, page II-132, (1992).

FIGURE 4.4.5; COMPARISON OF FLUENT CALCULATED FUEL ASSEMBLY CONDUCTIVITY RESULTS WITH PUBLISHED TECHNICAL DATA

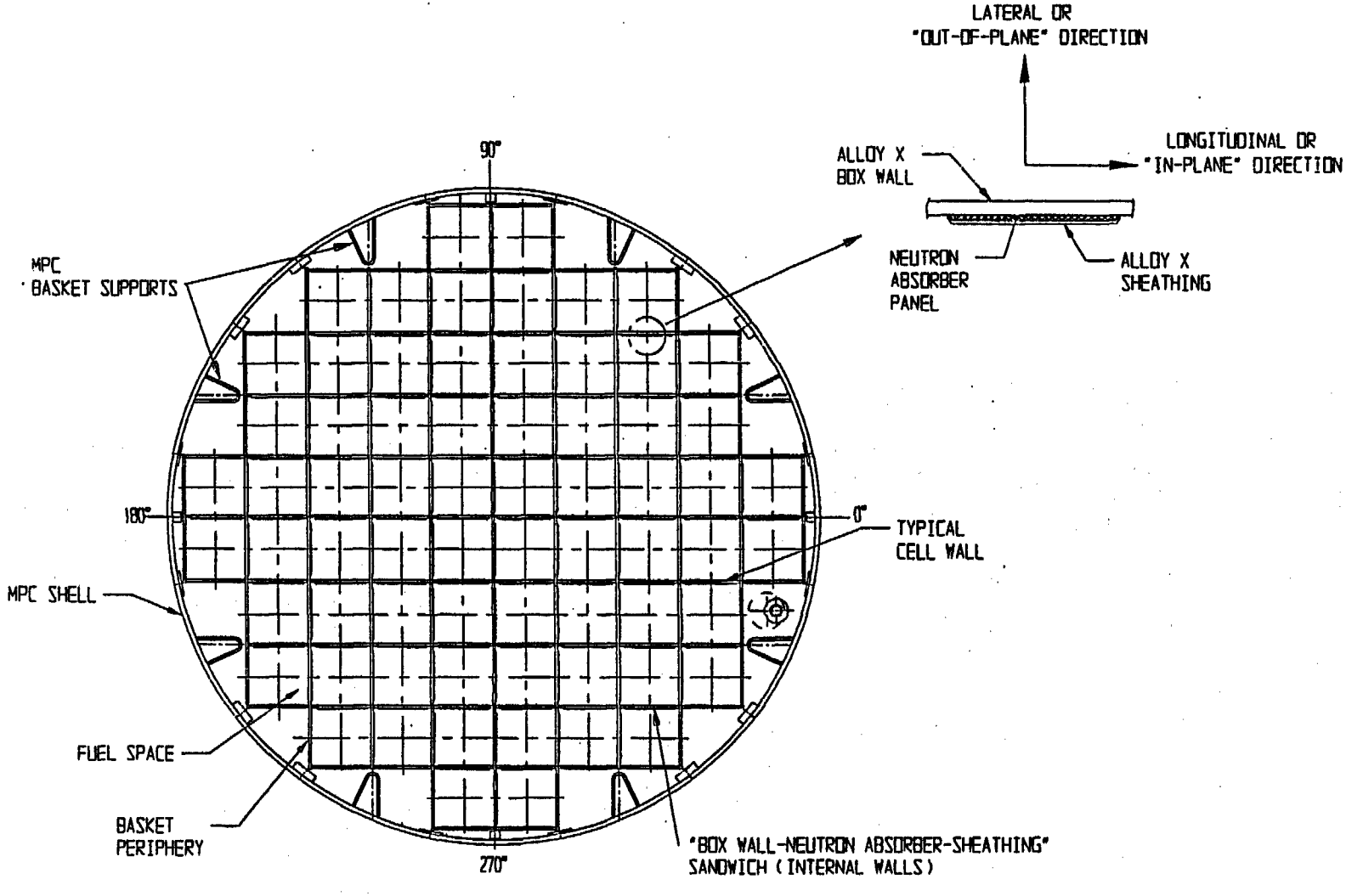


FIGURE 4.4.6; TYPICAL MPC BASKET PARTS IN A CROSS-SECTIONAL VIEW

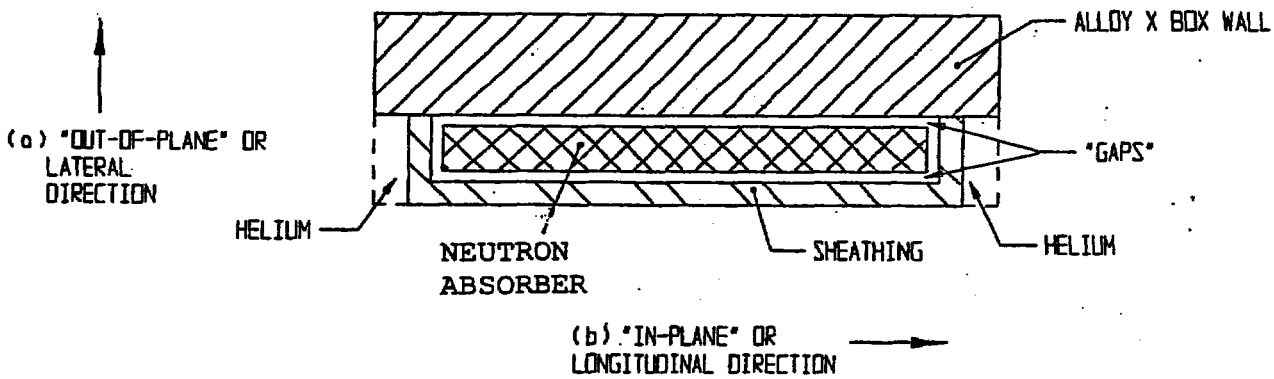


FIGURE 4.4.7: "BOX WALL-NEUTRON ABSORBER-SHEATHING SANDWICH"

FIGURE 4.4.8

THIS FIGURE INTENTIONALLY DELETED.

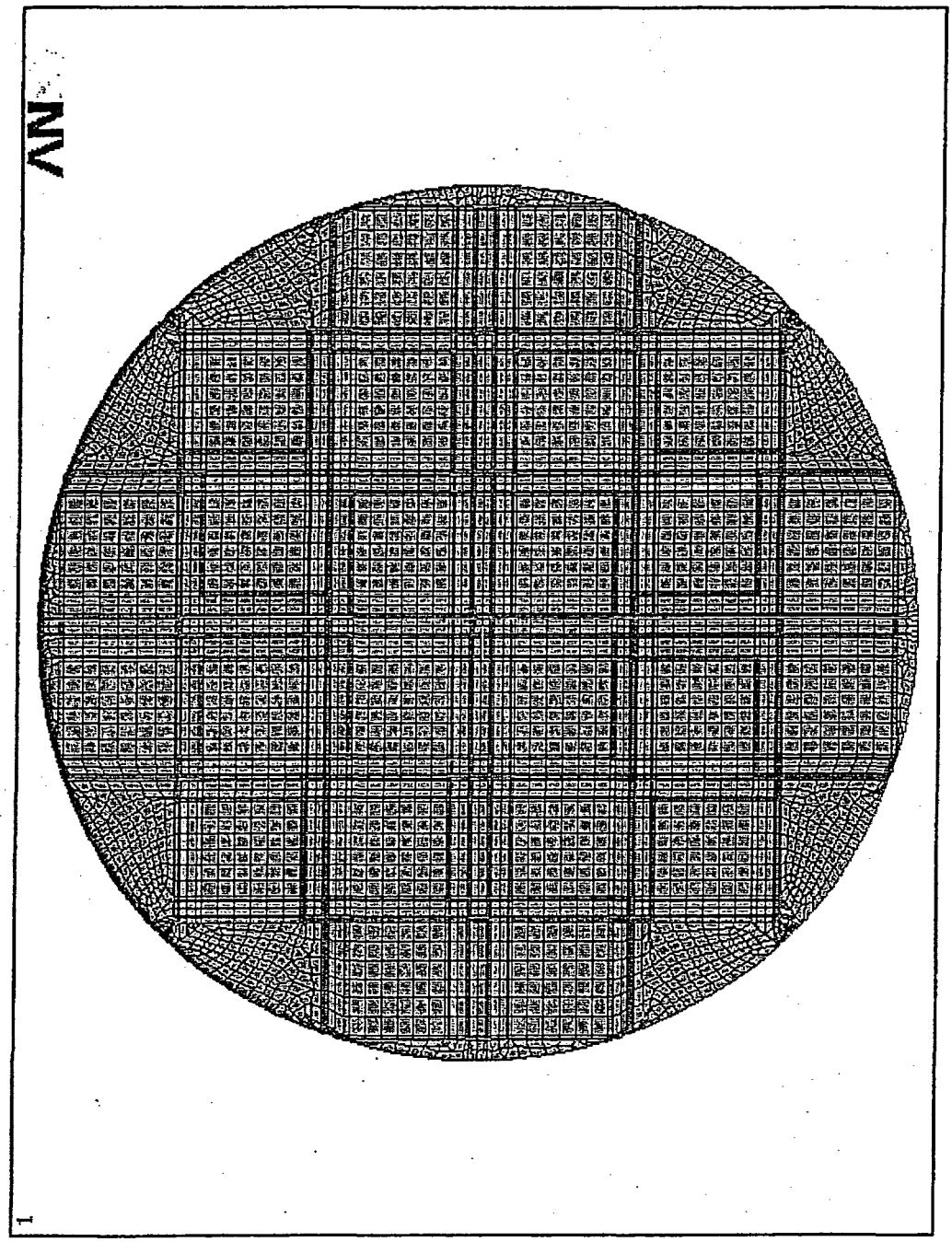


FIGURE 4.4.9; MPC-24 BASKET CROSS-SECTION ANSYS FINITE-ELEMENT MODEL

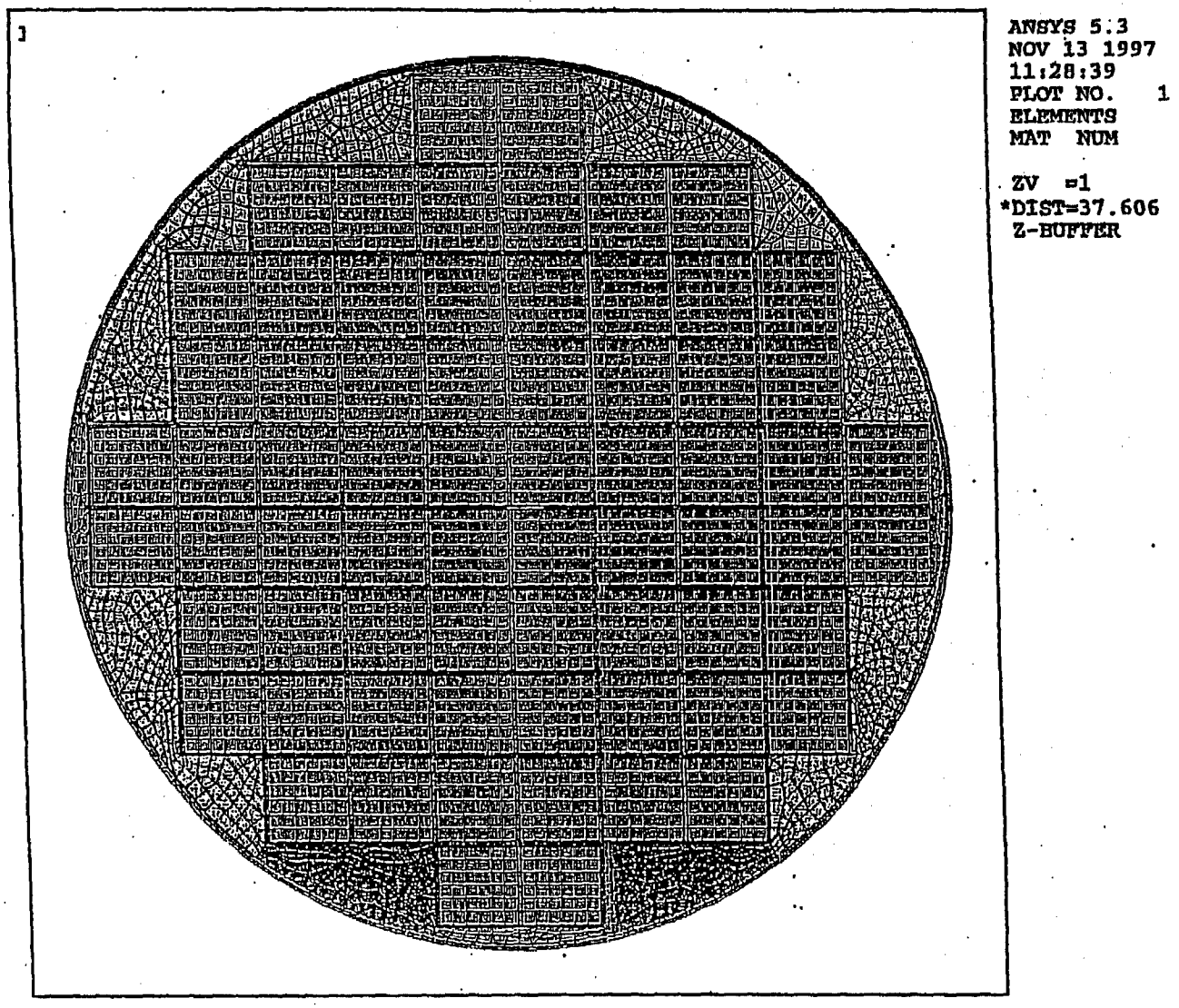


FIGURE 4.4.10; MPC-68 BASKET CROSS-SECTION ANSYS FINITE ELEMENT MODEL

FIGURE 4.4.11

[INTENTIONALLY DELETED]

FIGURE 4.4.12

[INTENTIONALLY DELETED]

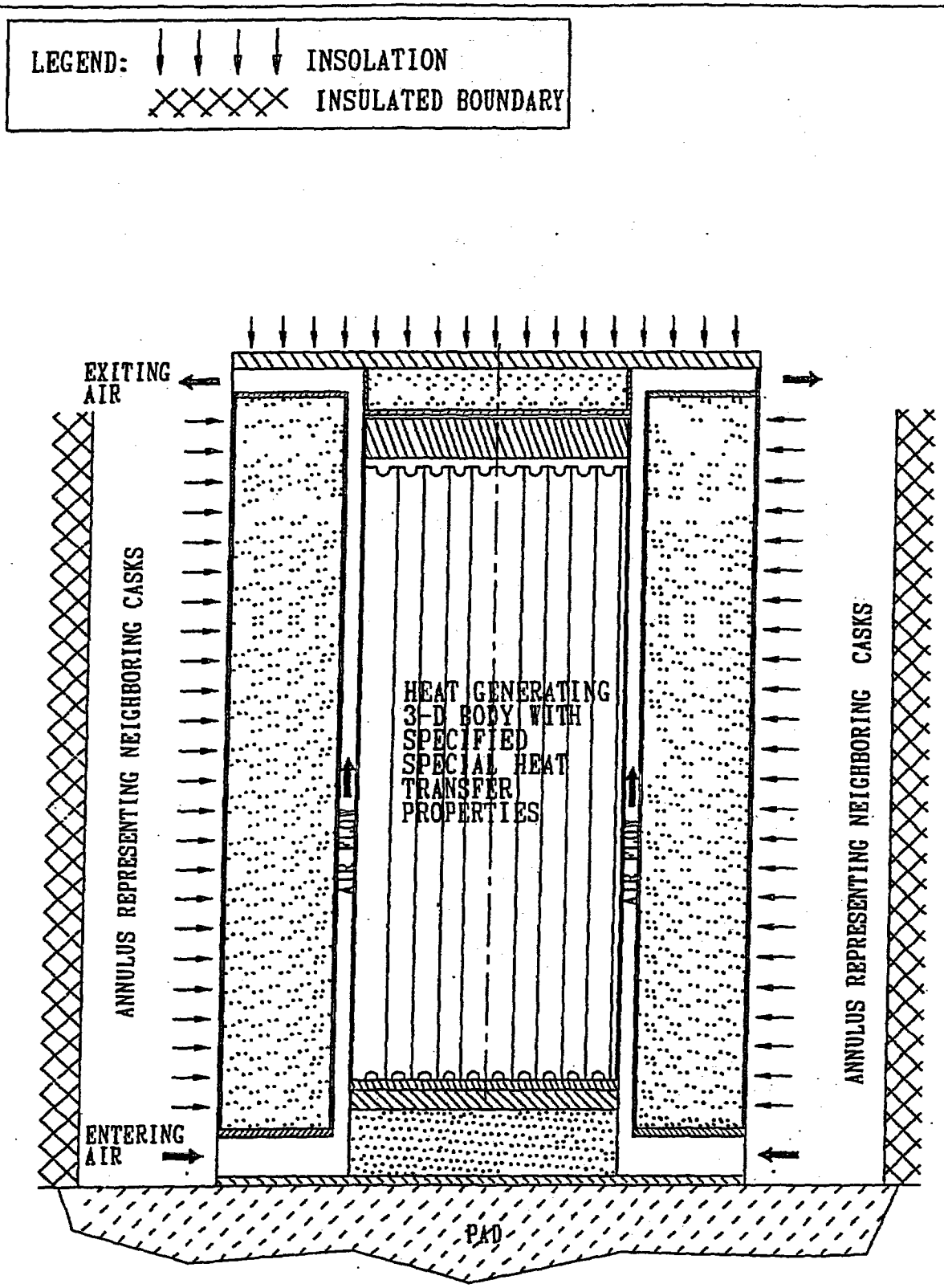


FIGURE 4.4.13; SCHEMATIC DEPICTION OF THE HI-STORM THERMAL ANALYSIS

FIGURE 4.4.14

INTENTIONALLY DELETED

FIGURE 4.4.15

THIS FIGURE INTENTIONALLY DELETED.

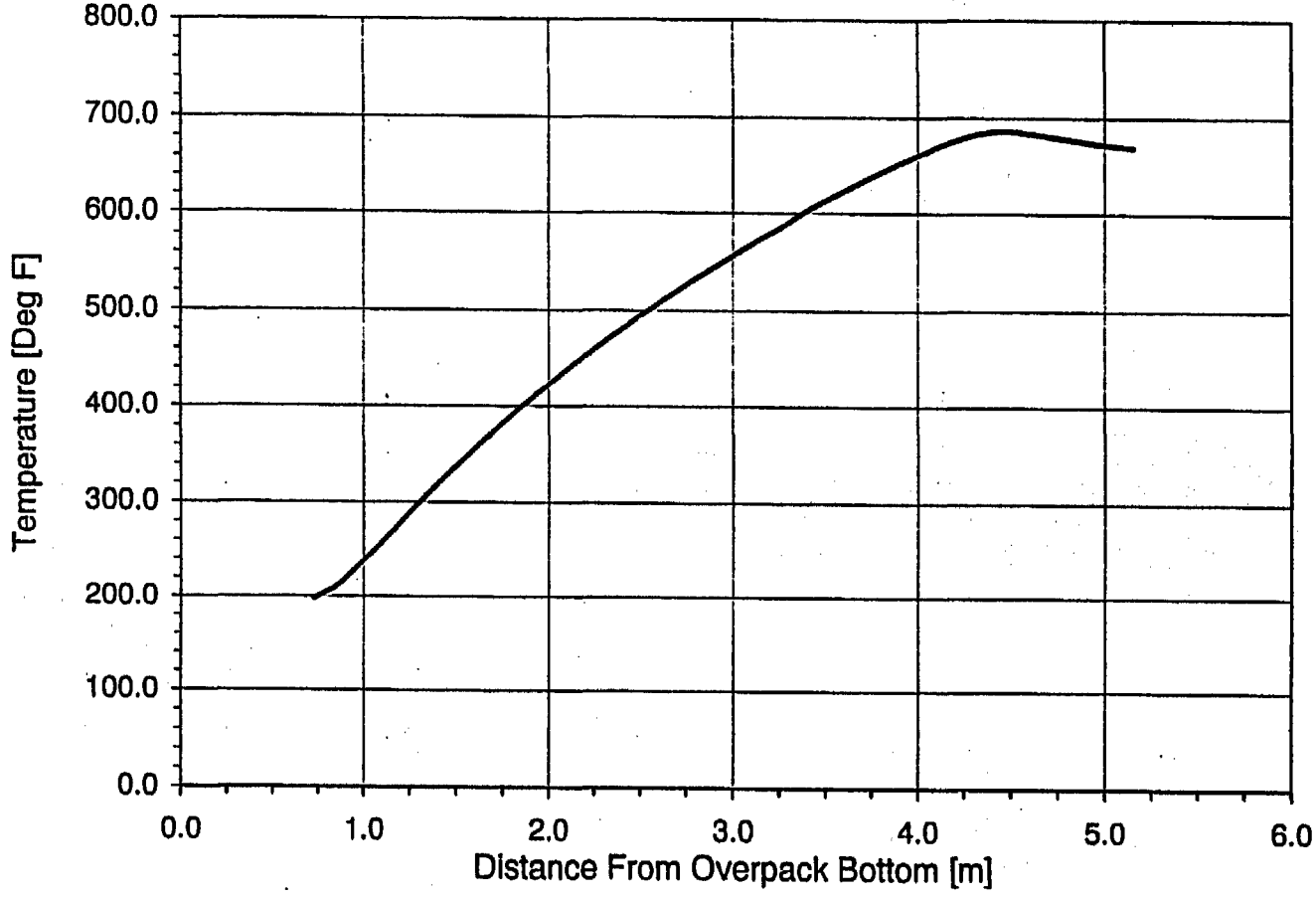


FIGURE 4.4.16: MPC-24 PEAK ROD AXIAL TEMPERATURE PROFILE

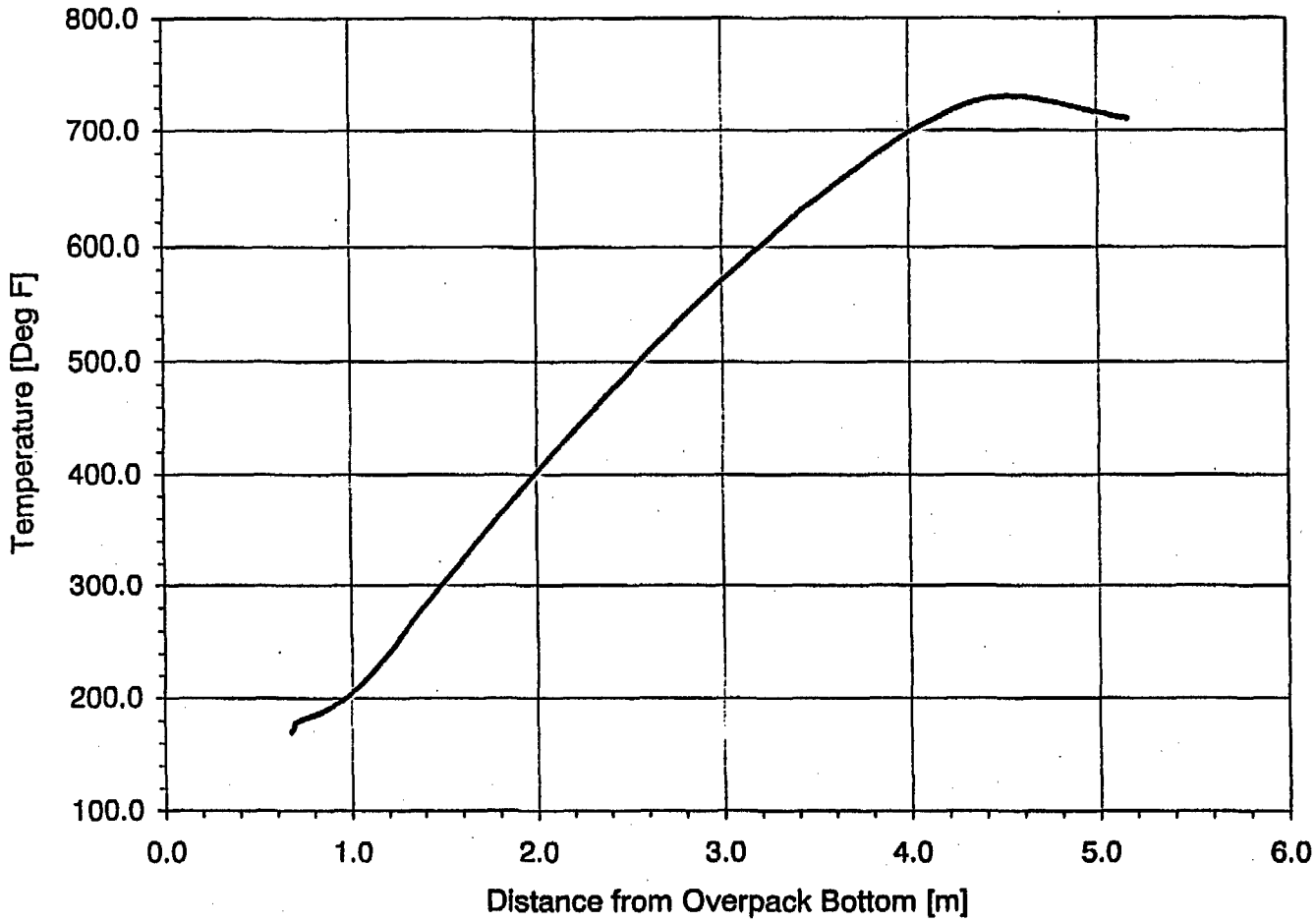


FIGURE 4.4.17: MPC-68 PEAK ROD AXIAL TEMPERATURE PROFILE

FIGURE 4.4.18

THIS FIGURE INTENTIONALLY DELETED.

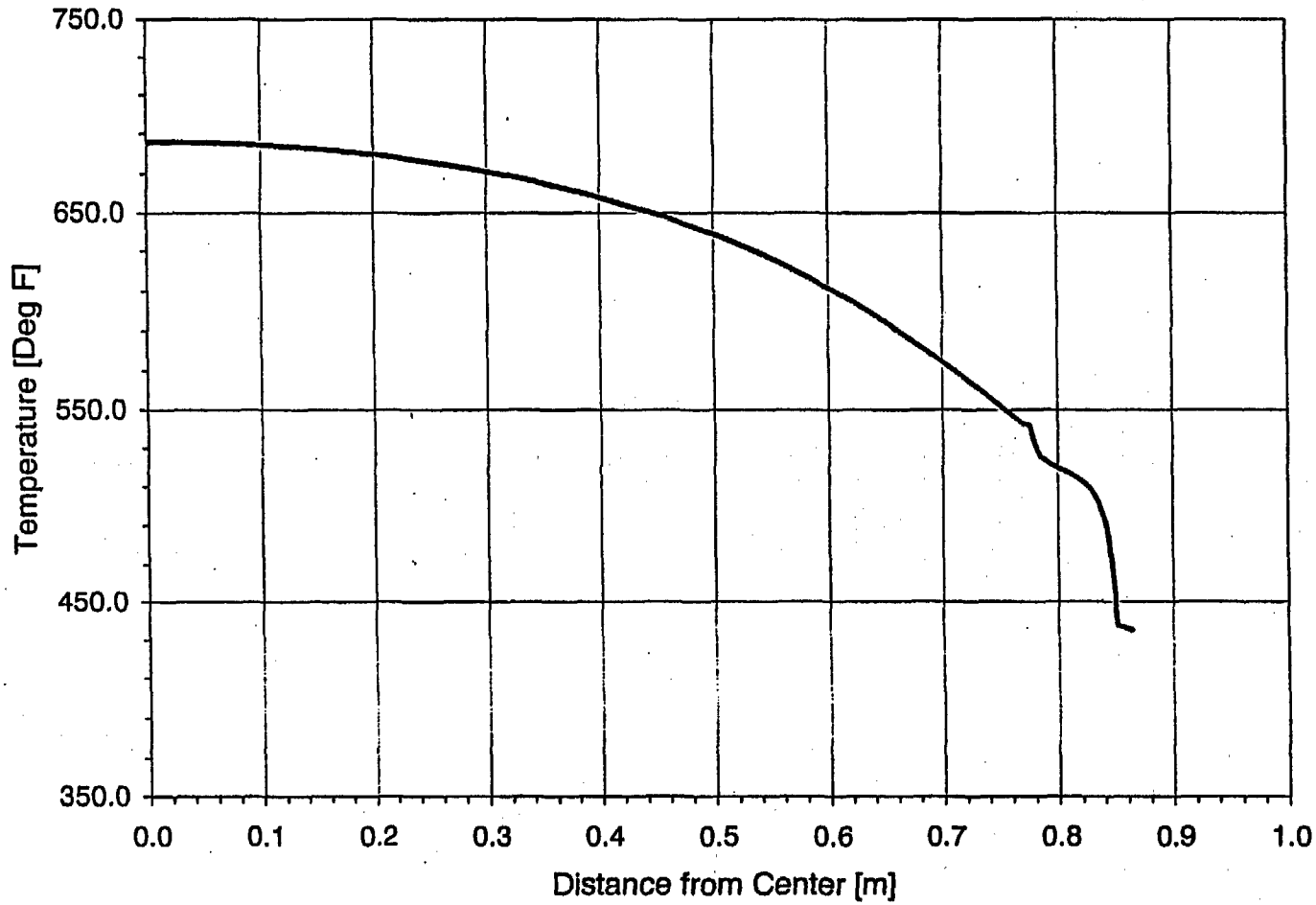


Figure 4.4.19: MPC-24 RADIAL TEMPERATURE PROFILE
(Hottest Basket Cross-Section)

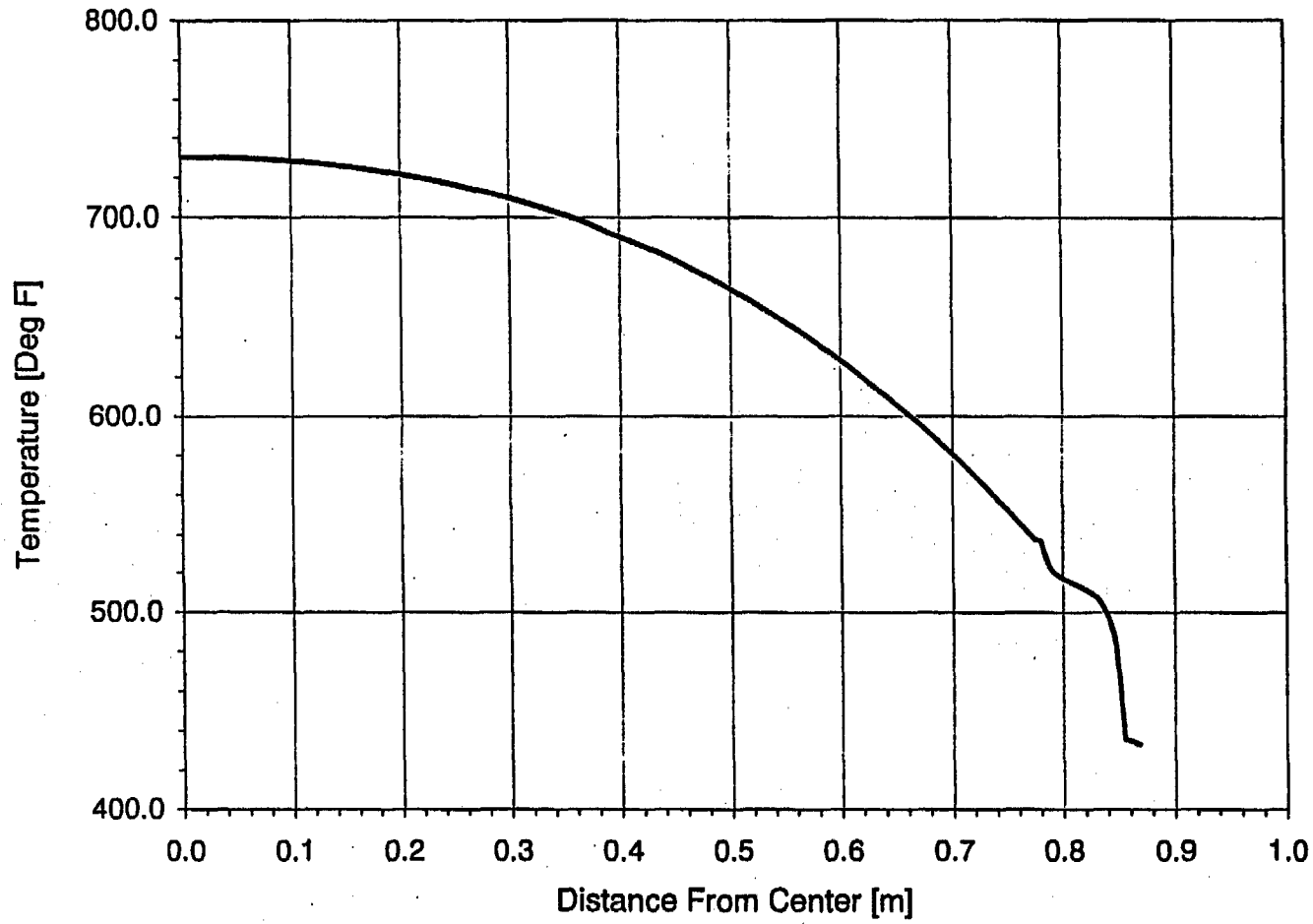


FIGURE 4.4.20: MPC-68 RADIAL TEMPERATURE PROFILE
(Hottest Basket Cross-Section)

FIGURE 4.4.21

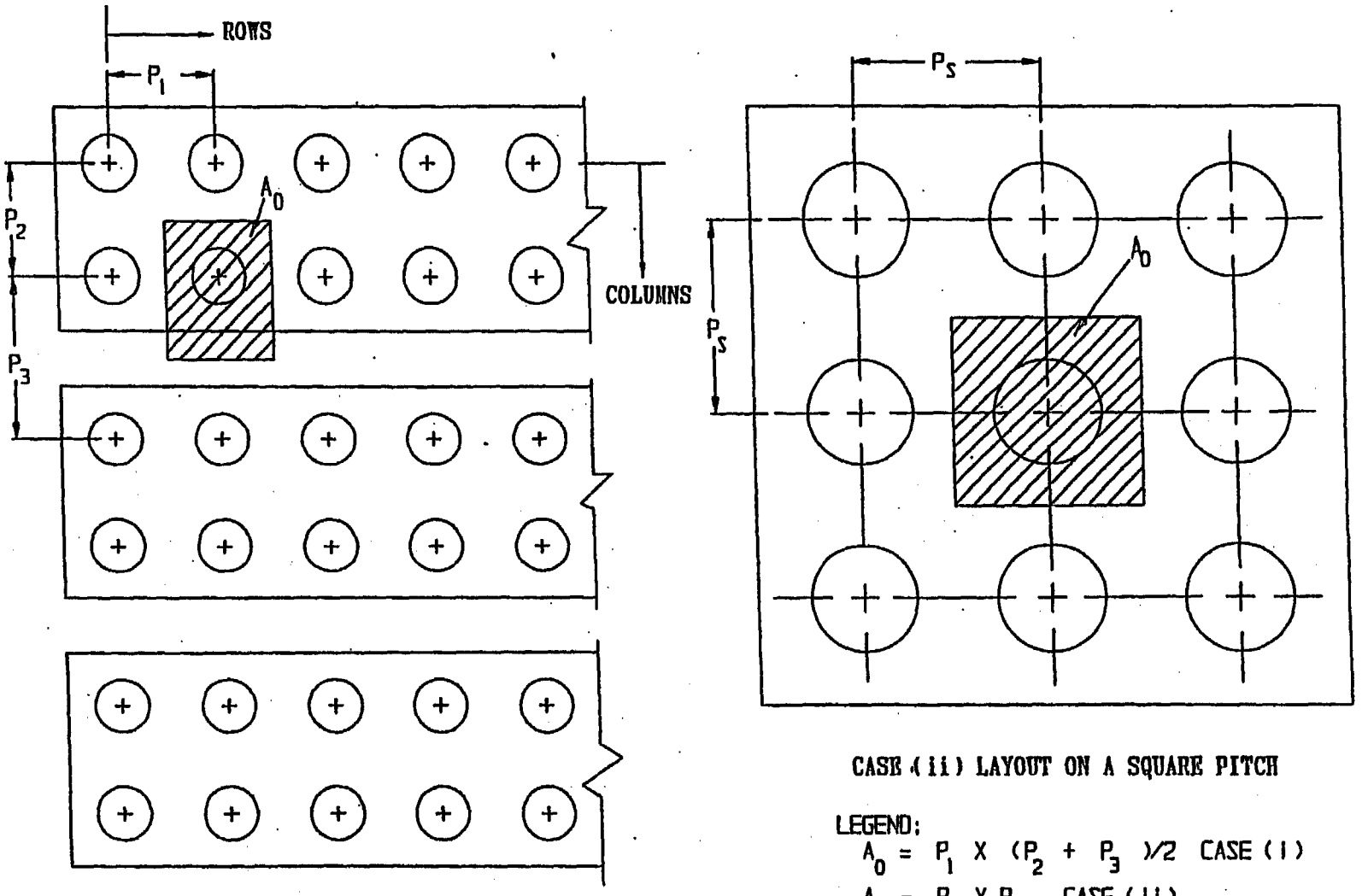
THIS FIGURE INTENTIONALLY DELETED.

FIGURE 4.4.22

THIS FIGURE INTENTIONALLY DELETED

FIGURE 4.4.23

THIS FIGURE INTENTIONALLY DELETED



CASE (i) LAYOUT ON A RECTANGULAR PITCH

CASE (ii) LAYOUT ON A SQUARE PITCH

LEGEND:

$A_0 = P_1 \times (P_2 + P_3)/2$ CASE (I)

$A_0 = P_S \times P_S$ CASE (II)

FIGURE 4.4.24; ILLUSTRATION OF MINIMUM AVAILABLE PLANAR AREA PER HI-STORM MODULE AT AN ISFSI.

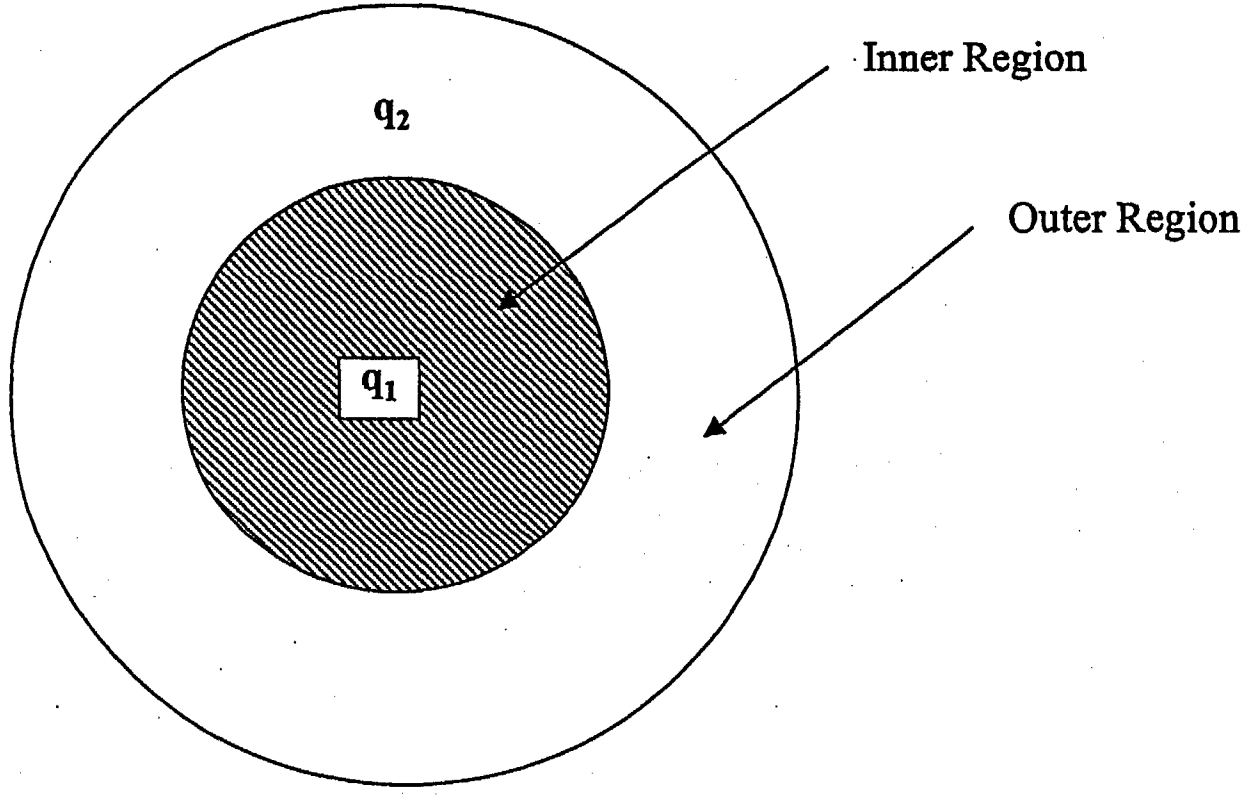


FIGURE 4.4.25: FUEL BASKET REGIONALIZED LOADING SCENARIO

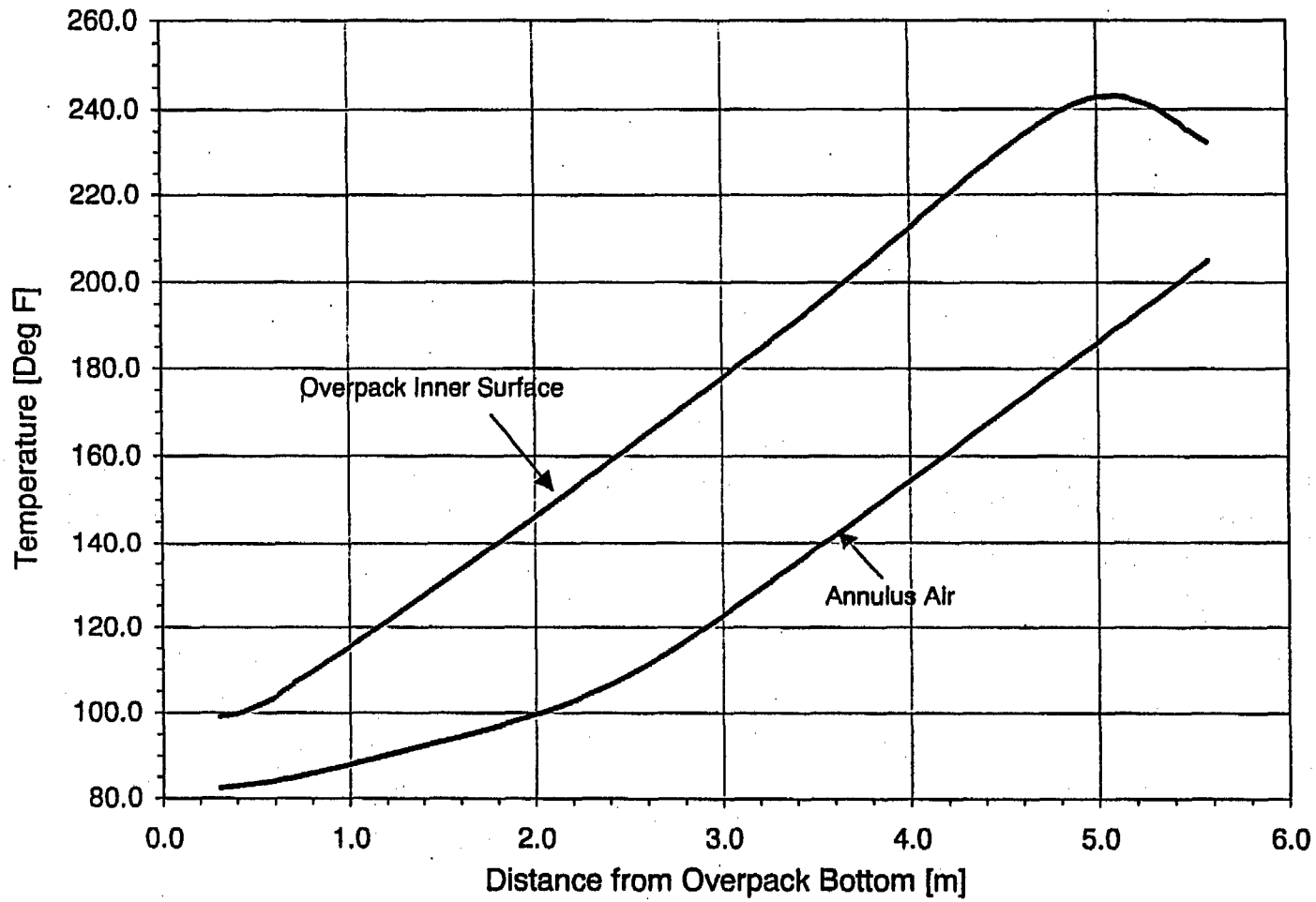


FIGURE 4.4.26: BOUNDING MPC-68 OVERPACK ANNULUS AXIAL PROFILES

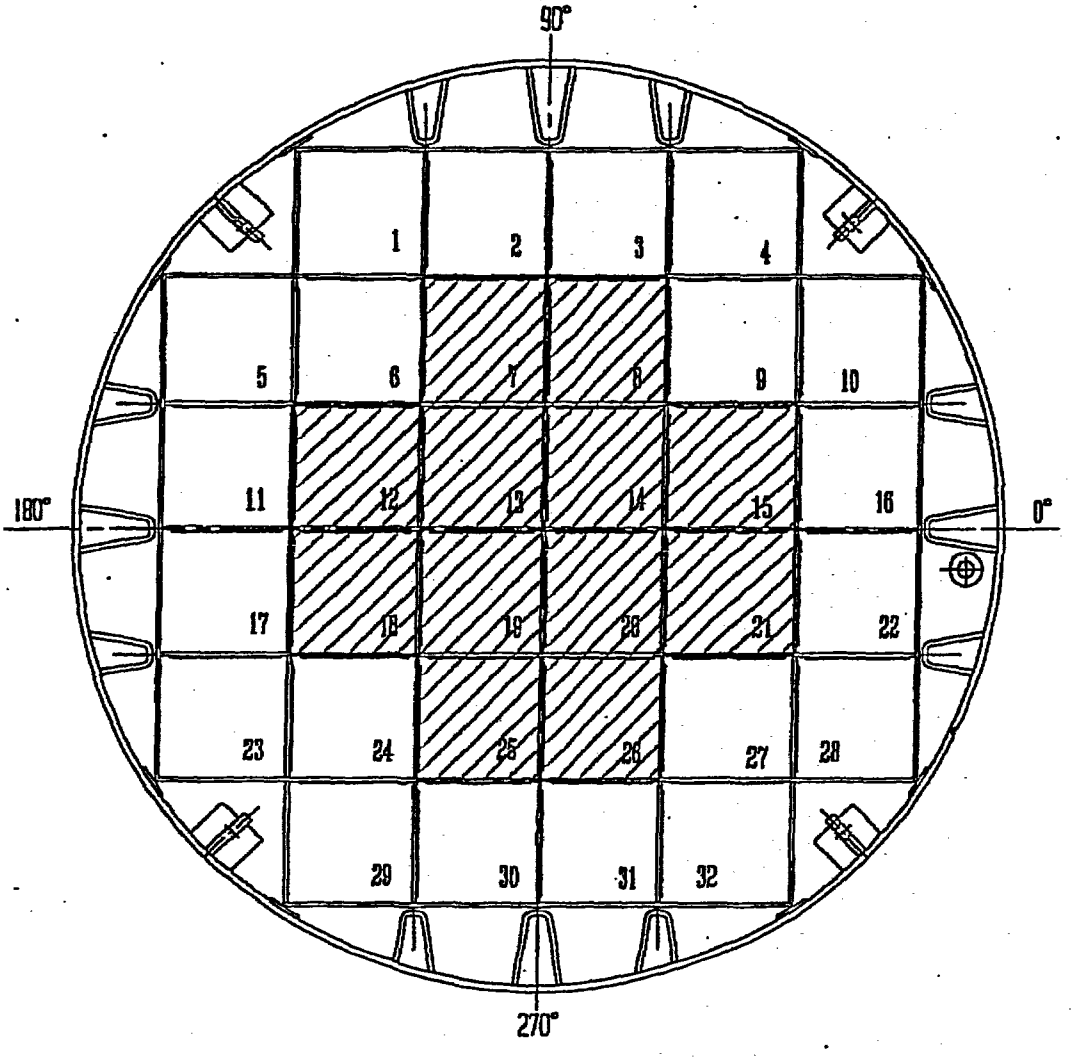


FIGURE 4.4.27: MPC-32 REGIONALIZED LOADING
(Region 1 Cells Shown Cross-Hatched)

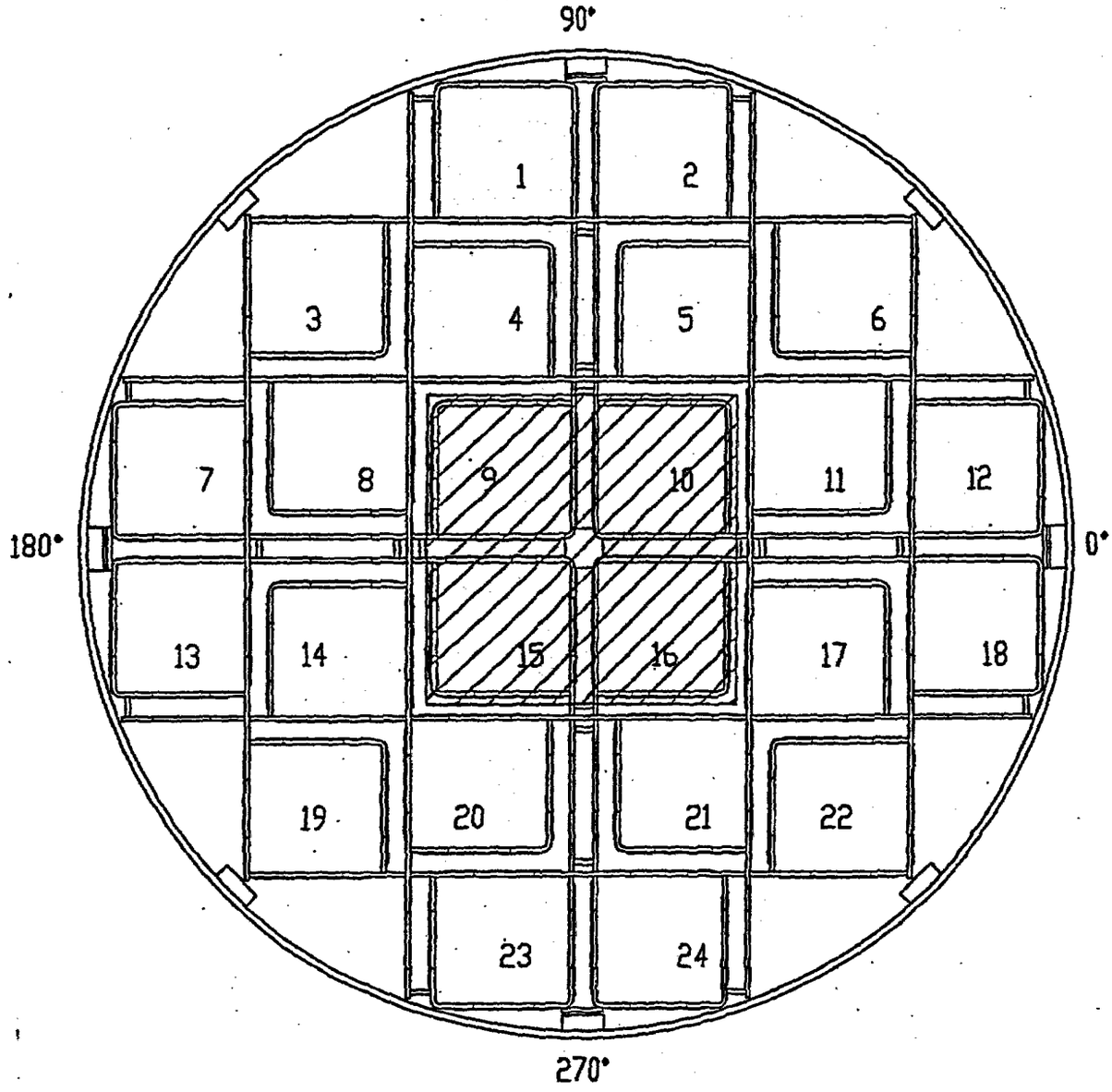


FIGURE 4.4.28: MPC-24 & MPC-24E REGIONALIZED LOADING
(Region 1 Cells Shown Cross-Hatched)

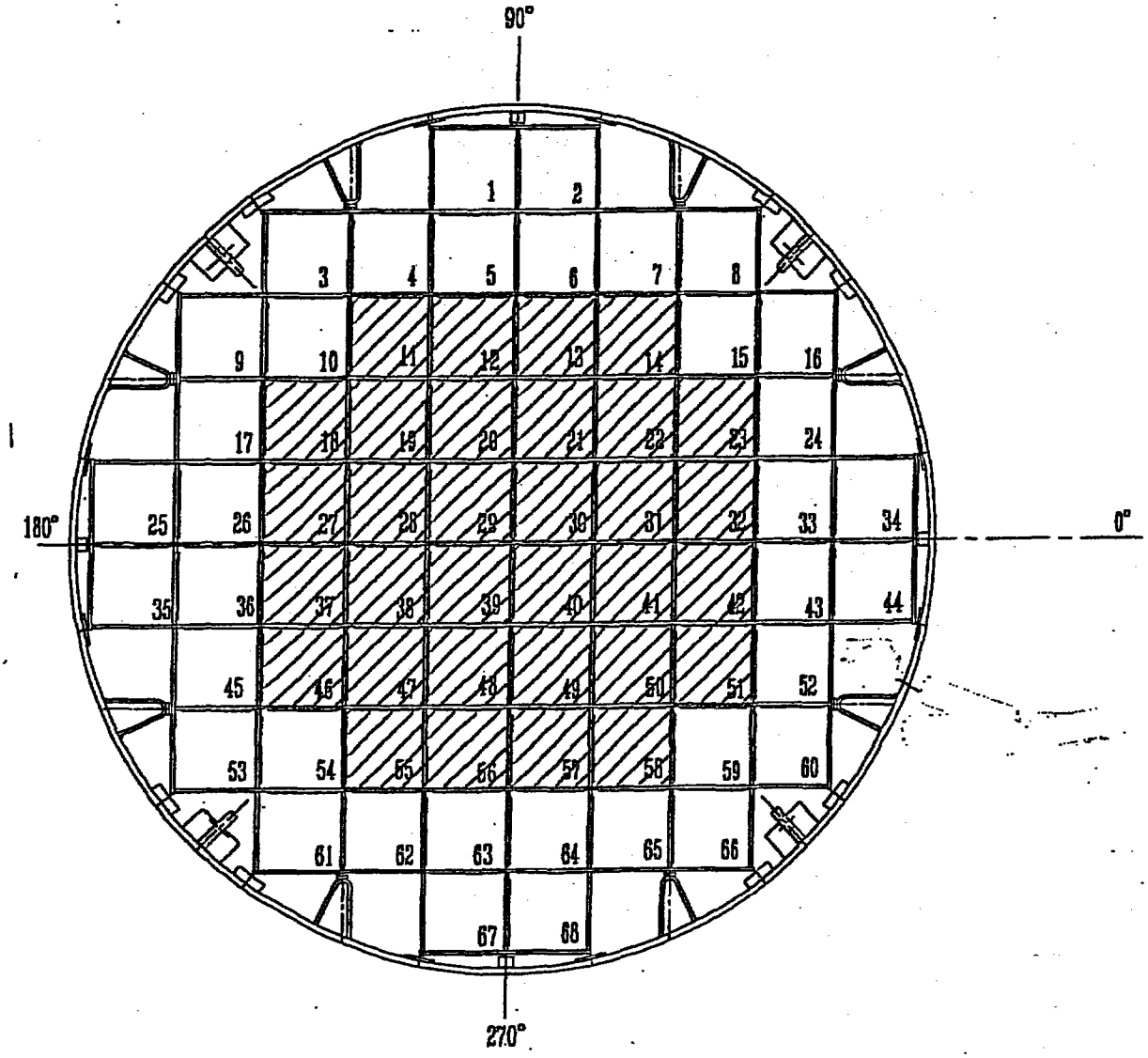


FIGURE 4.4.29: MPC-68 REGIONALIZED LOADING
(Region 1 Cells Shown Cross-Hatched)

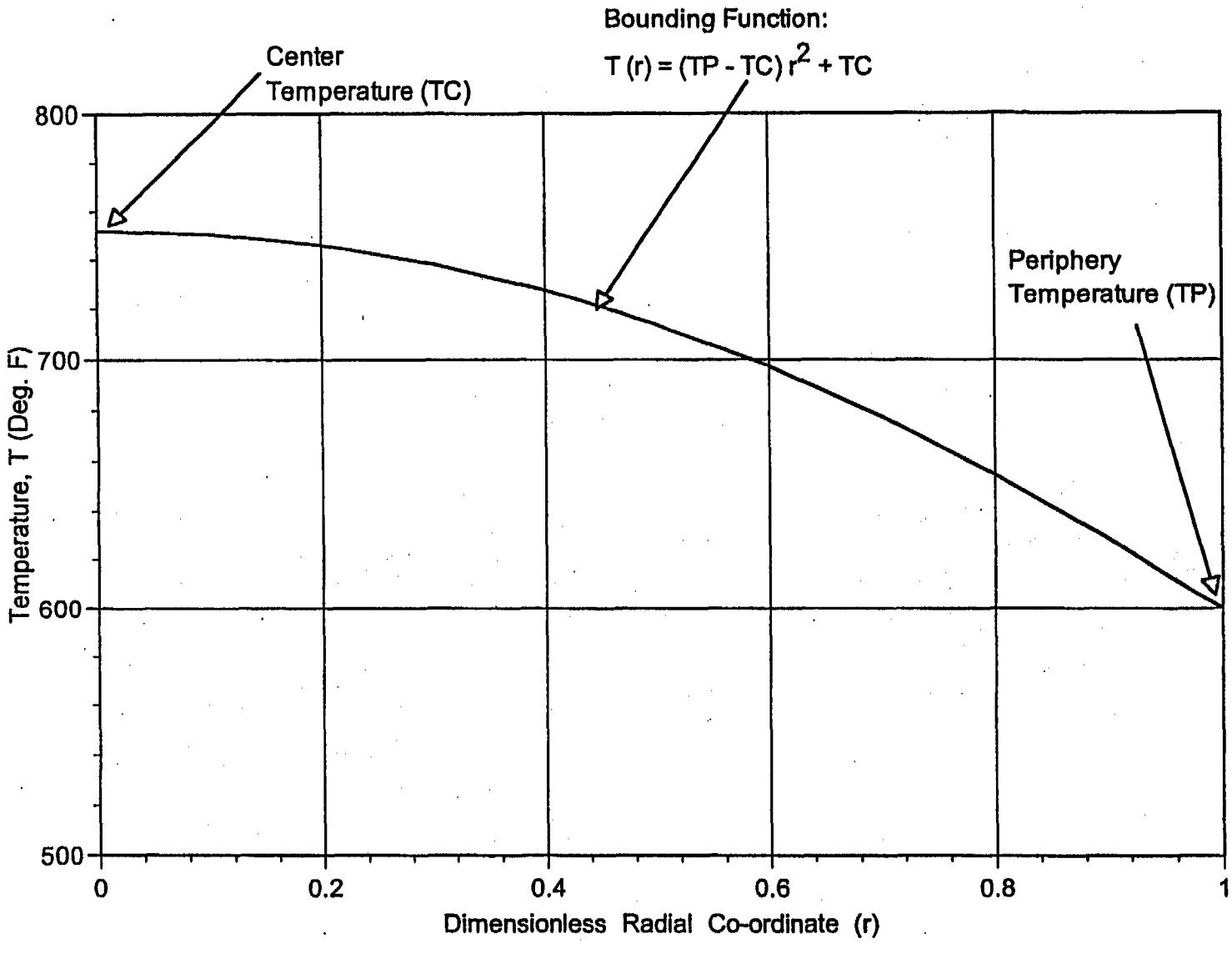


FIGURE 4.4.30: BOUNDING BASKET TEMPERATURE PROFILE FOR DIFFERENTIAL EXPANSION

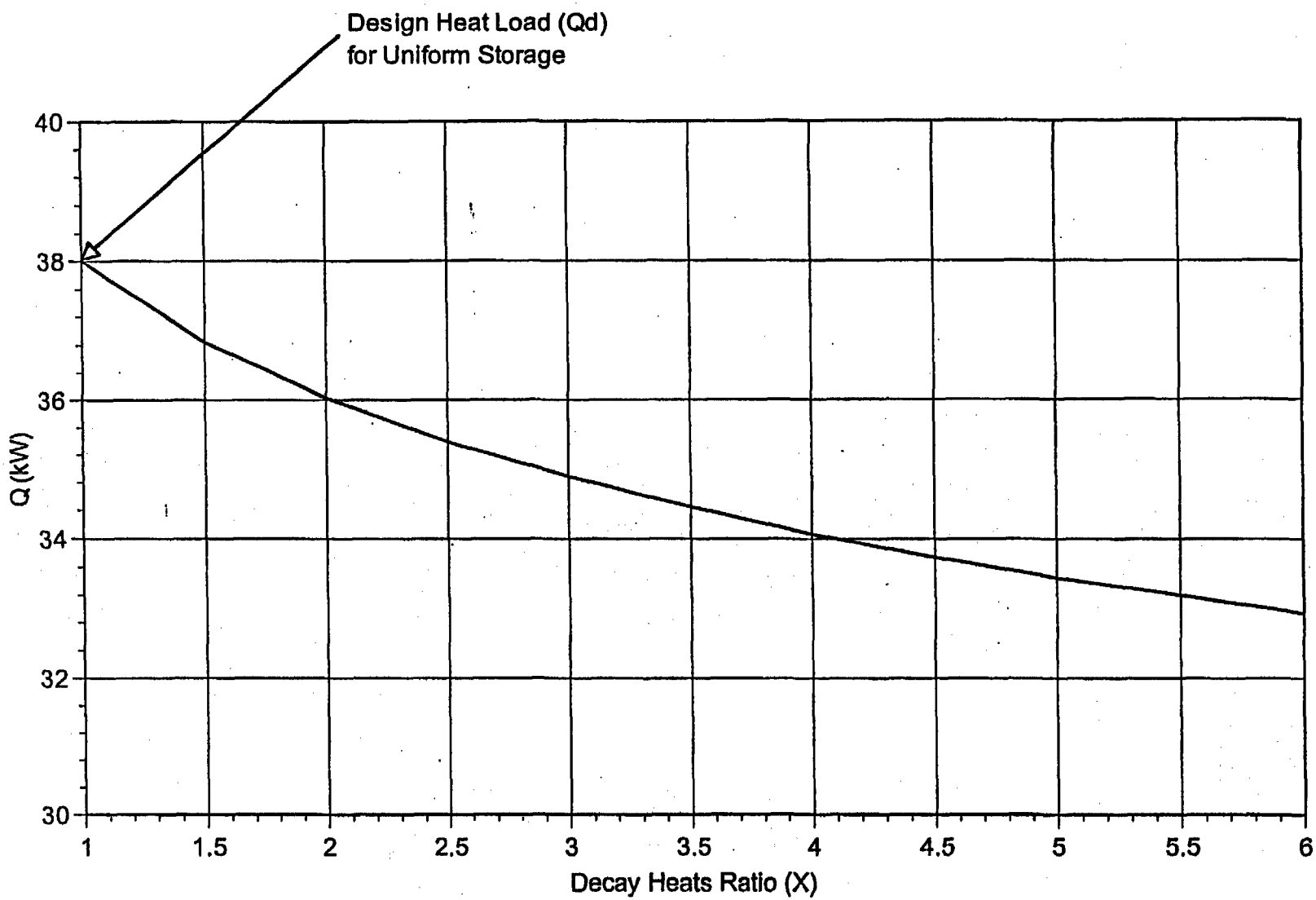


FIGURE 4.4.31: REGIONALIZED STORAGE DESIGN HEAT LOAD CURVE FOR PWR MPCs

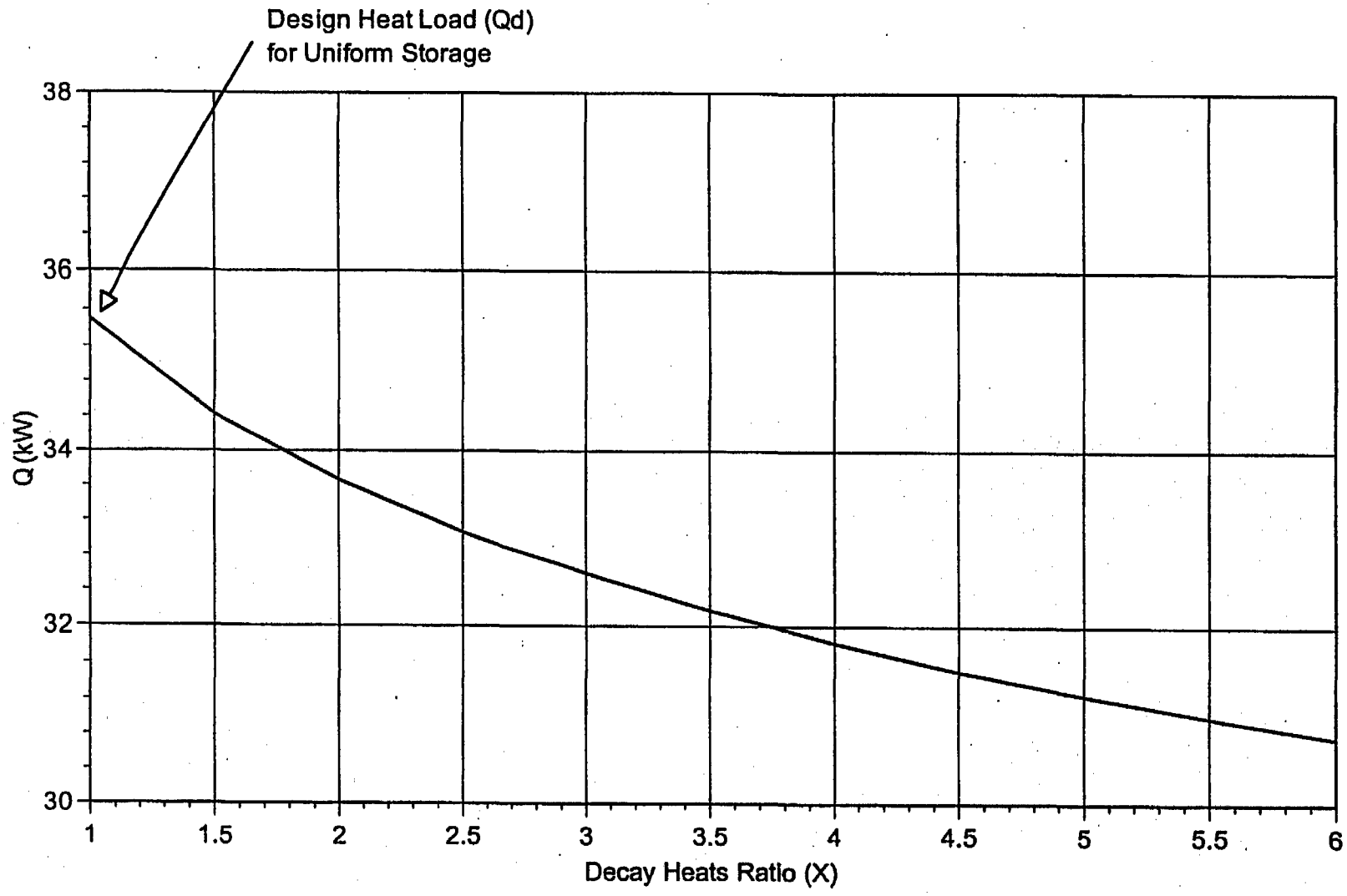


FIGURE 4.4.32: REGIONALIZED STORAGE DESIGN HEAT LOAD CURVE FOR BWR MPCs

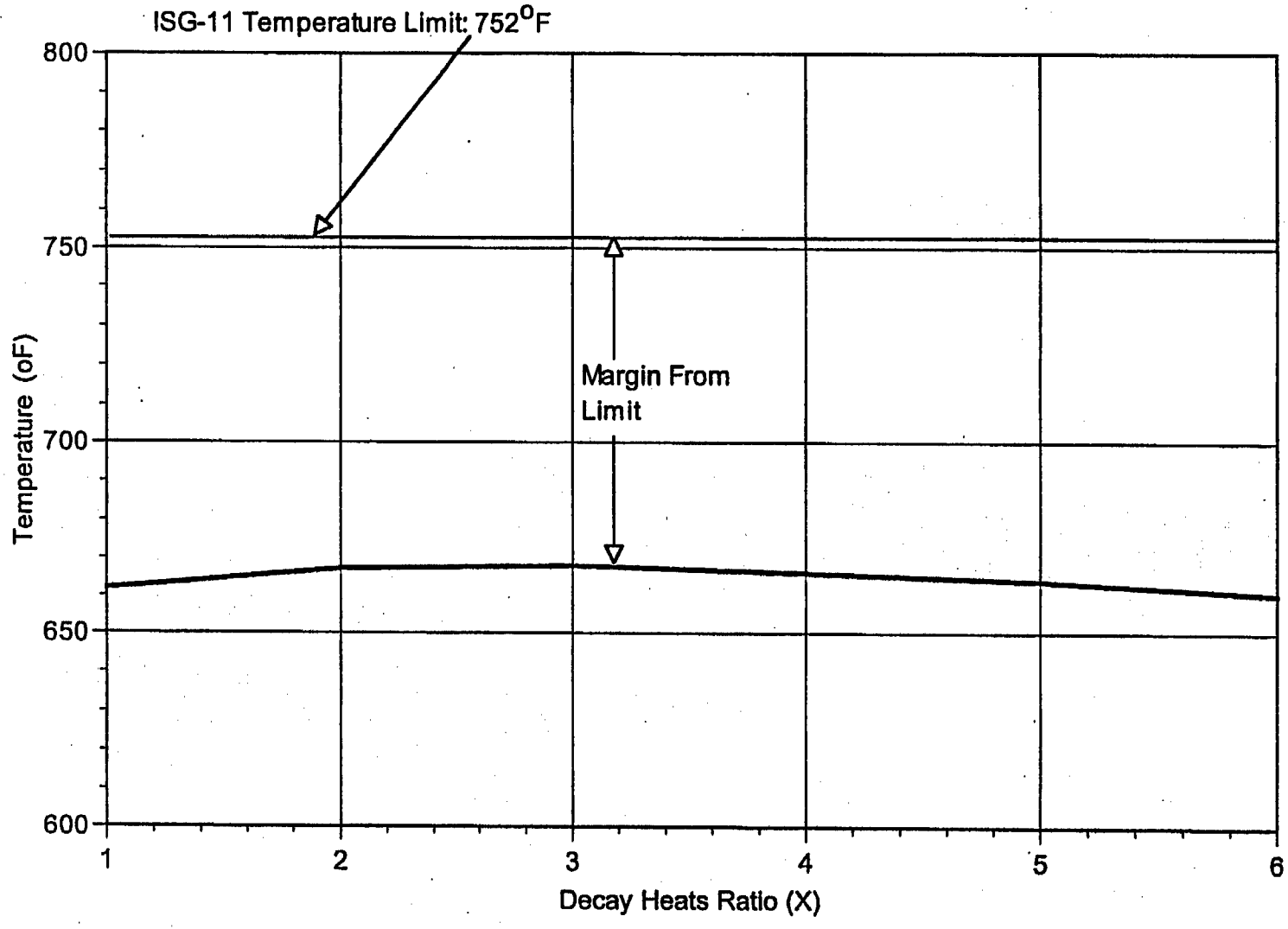


FIGURE 4.4.33: PEAK CLADDING TEMPERATURE VARIATION IN REGIONALIZED STORAGE (MPC-32)

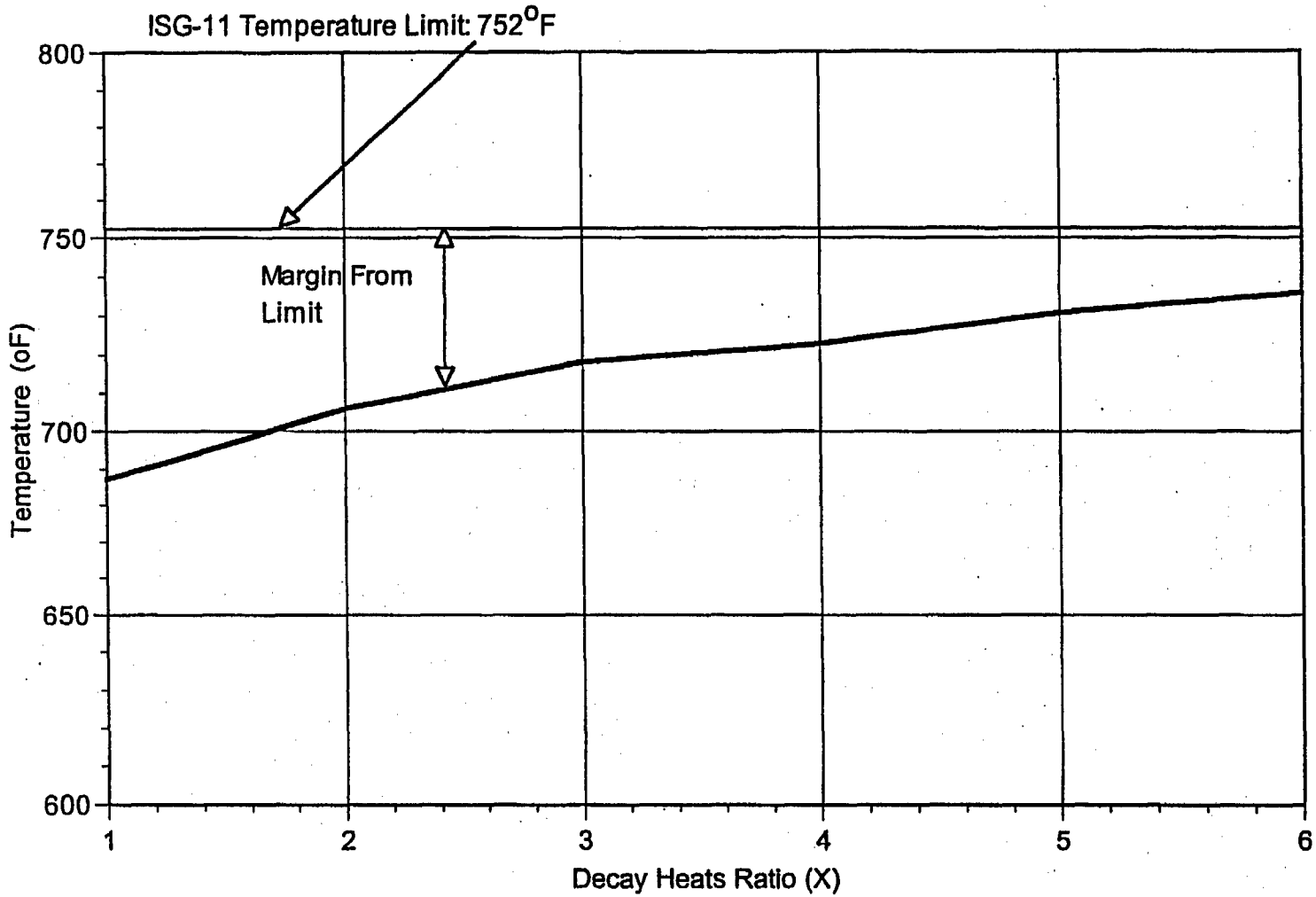


FIGURE 4.4.34: PEAK CLADDING TEMPERATURE VARIATION IN REGIONALIZED STORAGE (MPC-24/24E)

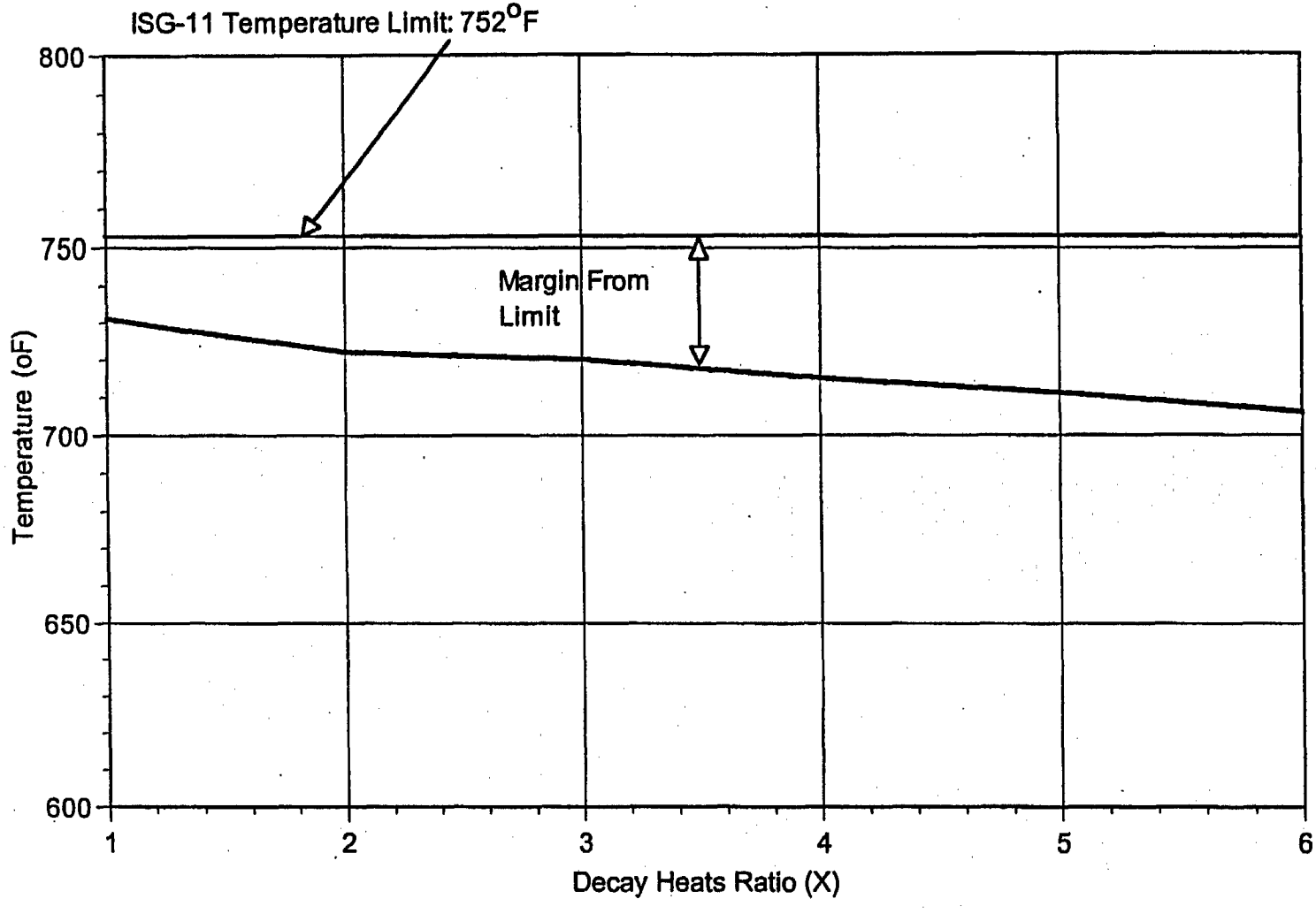


FIGURE 4.4.35: PEAK CLADDING TEMPERATURE VARIATION IN REGIONALIZED STORAGE (MPC-68)

4.5 THERMAL EVALUATION OF SHORT TERM OPERATIONS[‡]

4.5.1 Synopsis of Short Term Operations

Prior to placement in a HI-STORM overpack, an MPC must be loaded with fuel, outfitted with closures, dewatered, dried, backfilled with helium and transferred to the HI-STORM module. In the unlikely event that the fuel needs to be returned to the spent fuel pool, these steps must be performed in reverse. Finally, if required, transfer of a loaded MPC between HI-STORM overpacks or between a HI-STAR transport overpack and a HI-STORM storage overpack must be carried out in an assuredly safe manner. All of the above operations are short duration events that would likely occur no more than once or twice for an individual MPC. As stated in Chapter 2, ISG-11, Rev. 2 places a temperature limit on the fuel cladding temperature under all short term operations.

The device central to all of the above operations is the HI-TRAC transfer cask that, as stated in Chapter 1, is available in two anatomically identical design platforms denoted as HI-TRAC 100 and HI-TRAC 125. The HI-TRAC transfer cask is a short-term host for the MPC; therefore it is necessary to establish that during all thermally challenging operation events involving either of the two HI-TRAC designs, the permissible temperature limits specified in Reference [4.1.4], unless otherwise justified, are not exceeded. The following discrete thermal scenarios, all of short duration, involving the HI-TRAC transfer cask have been identified as warranting thermal analysis.

i. Loading Operations with Flooded MPC
ii. Drying of the MPC Cavity
iii. Onsite transport in HI-TRAC
iv. MPC Cooldown and Reflood for Defueling Operations

Each of the above conditions corresponds to a distinct thermal state for the spent fuel stored in the MPC. The first three conditions pertain to MPC loading operations; the last scenario is germane to the rare case when a loaded MPC needs to be defueled. Out of the four scenarios listed above, demoisturization using the vacuum drying method is thermally most challenging causing the greatest elevation in fuel cladding temperatures. On-site transport of the MPC generally occurs with the HI-TRAC in the vertical orientation, which preserves the thermosiphon action within the MPC. However, there may be a scenario wherein on-site transport of an MPC must occur with the HI-TRAC in the horizontal configuration. Horizontal transfer is thermally more adverse than its vertical configuration because of the suppression of the thermosiphon mode of cooling when HI-TRAC is horizontal. The above mentioned scenarios associated with fuel loading are carried out in reverse if the contents of an MPC have to be unloaded.

The fuel handling operation scenarios described above place a certain level of constraint to the dissipation of heat from the MPC relative to the normal storage condition. Consequently, for

[‡] Because of the significant quantity of new material required to satisfy ISG-11 (latest revision), this section has been rewritten in its entirety.

some scenarios, it is necessary to limit the maximum MPC heat generation rates to those at which the steady state fuel cladding temperature limits are approached. Henceforth, these limiting heat generation rates for each scenario will be referred to as the "threshold heat load" (Q_T) for that scenario. For certain scenarios the threshold heat loads may be more restrictive than the design basis heat load. The threshold heat loads to comply with the prescribed temperature limits for the short term operations are computed and reported in this chapter. Analyses are performed for short term operations, and threshold heat loads obtained and reported in Subsections 4.5.4 through 4.5.7.

4.5.2 Thermal Acceptance Criteria for Short Term Operations

As stated in Section 4.1, the HI-STORM FSAR seeks to establish complete compliance with the provisions of Reference [4.1.4]. This requires the maximum cladding temperature to not exceed ISG-11 limits (see Table 4.3.1) during short term operations for all high burnup fuel.

As stated in Section 4.1, for MPCs loaded with only Moderate Burnup Fuel (MBF), short term operations are permitted provided the following two criteria are satisfied[§]:

(i) The estimated cladding hoop stress, σ_{max}, does not exceed 90 MPa.
(ii) The peak cladding temperature is below 570°C (1058°F).

Because the cladding stress is a function of cladding thickness and cladding corrosion, the estimation of stress (Criterion (ii) above), of necessity, must be fuel-specific. The necessary computations are performed by a cask user opting for this criterion by employing the methodology described next.

The procedure for estimating σ_{max} to establish compliance with Criterion (ii) above is carried out in the following steps:

a. Determine the axial temperature distribution in the hottest rod using the HI-STORM thermal model presented in this chapter.

   The applicable heat load is based on the SNF batch to be loaded in the MPC. The maximum SNF heat generation rate (calculated using methods described in Chapter 5) is ascribed to every fuel assembly to bound the actual cumulative heat generation rate from all SNF in an MPC.

b. Compute the average gas temperature in the hottest rod

   For computing the average gas temperature in the fuel rod, two distinct axial zones in the fuel rod are identified. One zone is the fuel rod pellets stack region wherein the gas space consists of the annular gap between the pellet and the cladding inside surface. The other

§ For MBF that does not muster compliance with the σ_{max} criteria, the Reference [4.1.4] temperature limit shall apply.

region is the gas plenum space above the fuel pellets. T_{avg} is obtained by a gas volume weighted average of the mean rod temperature in the two regions. T_{avg} is computed for the hottest fuel rod by the following formula:

$$T_{avg} = \frac{v\frac{1}{a}\int_{o}^{a} T(z)dz + \frac{V}{L-a}\int_{a}^{L} T(z)dz}{v+V}$$

where: T(z): axial rod temperature profile
v: gas volume in pellet-to-clad gap
V: plenum gas volume
a: length of pellet stack region
L: fuel rod length

c. Compute the plenum gas pressure, P_v

The initial thermodynamic state of gas confined inside a fuel rod is specified by two parameters: Gas pressure (P_o) and a reference gas temperature (T_o) in absolute units. By the Ideal Gas Law, the rod gas pressure upon heat up under vacuum (P_v) is proportional to the average gas temperature of a fuel rod (T_{avg}). In other words,
$P_v = P_o (T_{avg}/T_o)$.

d. Compute the maximum cladding hoop stress, σ_{max}

The hoop stress (σ_{max}) developed in cladding is a function of rod internal diameter (d_i) cladding thickness (t) and an internal rod gas pressure (P_v). The stress is computed by the Lame formula given below:

$$\sigma_{max} = \frac{P_v d_i}{2t}$$

The cladding thickness should be taken as the nominal thickness less the estimated metal loss due to corrosion. The inside diameter of the cladding should be taken as the nominal diameter. Satisfying the above cited 90 MPa limit is an essential requirement for the MBF's cladding temperature limit to be set at 570°C for short term operations.

4.5.3 The HI-TRAC Thermal Model

4.5.3.1 Overview

The HI-TRAC transfer cask is used to load and unload the HI-STORM concrete storage overpack, including onsite transport of the MPCs from the loading facility to an ISFSI site . Section views of the HI-TRAC are provided in Chapter 1. Within a loaded HI-TRAC, heat generated in the MPC is transported from the contained fuel assemblies to the MPC shell in the

manner described in Section 4.4. From the outer surface of the MPC to the ambient air, heat is transported by a combination of conduction, thermal radiation and natural convection modes of heat transfer.

Two HI-TRAC transfer cask versions, namely HI-TRAC 100 and HI-TRAC 125 have the same design features but with lower member thicknesses in the 100 ton version to yield a lighter weight design. The analytical model developed for HI-TRAC thermal characterization is constructed to bound both 100 and 125 ton HI-TRAC transfer casks. In this manner, the HI-TRAC overpack resistance to heat transfer is overestimated, resulting in higher predicted MPC internals and fuel cladding temperature levels.

From the outer surface of the MPC to the ambient atmosphere, heat is transported within HI-TRAC through multiple concentric layers of air, steel and shielding materials. A small diametral gap exists between the outer surface of the MPC and the inner surface of the HI-TRAC overpack. Heat is transported across this gap by the parallel mechanisms of conduction and thermal radiation. Assuming that the MPC is centered and does not contact the transfer overpack walls conservatively minimizes heat transport across this gap. Additionally, thermal expansion that would minimize the gap is conservatively neglected. Heat is transported through the cylindrical wall of the HI-TRAC transfer overpack by conduction through successive layers of steel, lead and steel. A water jacket, which provides neutron shielding for the HI-TRAC overpack, surrounds the cylindrical steel wall. Each water jacket cavity is long and narrow having the cross section of an annular sector. Conduction heat transfer occurs through both the water cavities and the radial connectors . Heat is passively rejected to the ambient from the outer surface of the HI-TRAC transfer overpack by natural convection and thermal radiation.

In the vertical position, the bottom face of the HI-TRAC cask is in contact with a supporting surface. This face is conservatively modeled as an insulated surface. Because HI-TRAC is not used for long-term storage in an array, radiative blocking does not need to be considered. The HI-TRAC top lid is modeled as a surface with convection, radiative heat exchange with air and a constant maximum incident solar heat flux load. Insolation on cylindrical surfaces is conservatively based on 12-hour levels prescribed in 10CFR71 averaged on a 24-hour basis. Summary descriptions of the various components of HI-TRAC's thermal model are provided below.

4.5.3.2 Effective Thermal Conductivity of Water Jacket

The HI-TRAC water jacket is composed of an array of radial ribs equispaced around the HI-TRAC body and affixed to enclosure plates by welding to form discrete water compartments. Holes in the radial ribs connect all the individual compartments in the water jacket. Thus, the annular region between the HI-TRAC outer shell and the enclosure shell can be considered as an array of steel ribs and water spaces.

The effective radial thermal conductivity of this array of steel ribs and water spaces is determined by combining the heat transfer resistance of individual components in a parallel network. A bounding calculation is assured by using a minimum available metal thickness for radial heat

transfer. The thermal conductivity of the parallel steel ribs and water spaces is given by the following formula:

$$K_{re} = \frac{K_r (N_r t_r) \ln\left(\frac{r_o}{r_i}\right)}{2\pi L_R} + \frac{K_w (N_r t_w) \ln\left(\frac{r_o}{r_i}\right)}{2\pi L_R}$$

where:

K_{re} = effective radial thermal conductivity of water jacket
r_i = inner radius of water spaces
r_o = outer radius of water spaces
K_r = thermal conductivity of carbon steel ribs

$N_r t_r$ = Available metal thickness (product of number of radial ribs and rib thickness)
L_R = effective radial heat transport length through water spaces
K_w = thermal conductivity of water
$N_r t_w$ = Cumulative waterspaces width (between radial ribs)

4.5.3.3 Heat Rejection from Transfer Cask Exterior Surfaces

The following relationship for the surface heat flux from the outer surface of an isolated cask to the environment applied to the thermal model:

$$q_s = 0.19 (T_s - T_A)^{4/3} + 0.1714\varepsilon [(\frac{T_s + 460}{100})^4 - (\frac{T_A + 460}{100})^4]$$

where:

T_S = cask surface temperatures (°F)
T_A = ambient atmospheric temperature (°F)
q_s = surface heat flux (Btu/ft^2×hr)
ε = surface emissivity

The second term in this equation is the Stefan-Boltzmann formula for thermal radiation from an exposed surface to ambient. The first term is the natural convection heat transfer correlation recommended by Jacob and Hawkins [4.2.9]. This correlation is appropriate for turbulent natural convection from vertical surfaces, such as the vertical overpack wall. Although the ambient air is conservatively assumed to be quiescent, the natural convection is nevertheless turbulent.

Turbulent natural convection correlations are suitable for use when the product of the Grashof and Prandtl (Gr×Pr) numbers exceeds 10^9. This product can be expressed as $L^3 \times \Delta T \times Z$, where L is the characteristic length, ΔT is the surface-to-ambient temperature difference, and Z is a function of the surface temperature (defined in Section 4.2). The characteristic length of a vertically oriented HI-TRAC is its height of approximately 17 feet. The value of Z, conservatively taken at a bounding surface temperature of 340°F, is 2.6×10^5. Solving for the

value of ΔT that satisfies the equivalence $L^3 \times \Delta T \times Z = 10^9$ yields $\Delta T = 0.78°F$. For a horizontally oriented HI-TRAC the characteristic length is the diameter of approximately 7.6 feet (minimum of 100- and 125-ton versions), yielding $\Delta T = 8.76°F$. The natural convection will be turbulent, therefore, provided the surface to air temperature difference is greater than or equal to 0.78°F for a vertical orientation and 8.76°F for a horizontal orientation.

4.5.3.4 Determination of Solar Heat Input

As discussed in Section 4.4.1.1.8, the intensity of solar radiation incident on an exposed surface depends on a number of time varying terms. Twelve-hour averaged insolation levels are prescribed in 10CFR71. The HI-TRAC cask, however, possesses a considerable thermal inertia. This large thermal inertia precludes the HI-TRAC from reaching a steady-state thermal condition during a twelve-hour period. Thus, it is considered appropriate to use 24-hour averaged insolation levels.

4.5.3.5 Lead-to-Steel Interface

Lead, poured between the inner and outer shells of the HI-TRAC body, is utilized as a gamma shield material in the HI-TRAC transfer cask. Unlike many metal cask designs that utilize pre-fabricated lead "bricks", lead is installed in the HI-TRAC transfer cask in molten form. The lead pouring process is a mature technology and proven methods to preclude internal voids or gaps are well established in the industry.

To ensure a homogeneous lead pour, the HI-TRAC shell is pre-heated and molten lead is poured to fill the annular cavity. Ladling of the molten lead aided by the pressure from the column of molten lead helps ensure that internal voids or separation would not occur. The lead installation includes appropriate inspection measures, including gamma scan and weighing of the cask after lead placement, to confirm that the lead column installed in the HI-TRAC cask are without internal voids or gaps. Therefore, the lead-to-steel interface is assumed to be an uninterrupted continuum in the HI-TRAC thermal model.

4.5.4 Loading Operations with Flooded MPC

The HI-TRAC containing an MPC loaded with fuel and filled with water is removed from the cask pit (which may or may not be integral to the fuel pool) and placed in a designated space (typically on the pool deck) that is henceforth referred to as the Decontamination and Assembly Station (DAS). The early operations that occur at the DAS include welding of the main lid and testing of the welded joint using methods described in Chapter 8.

To minimize personnel dose and to keep the SNF in a cooled state, the SNF is kept submersed in water until the fuel drying operation (discussed in the next subsection) is initiated. The temperature of the fuel cladding is not a concern in the operational evolutions with the flooded MPC. In accordance with NUREG-1536, however, boiling of water in the MPC cavity is not permitted during wet loading operations. This requirement is met by imposing a limit on the

maximum allowable time duration for fuel to be submerged in water after a loaded HI-TRAC cask is removed from the pool and prior to the start of draining and fuel drying operations.

When the HI-TRAC transfer cask containing the loaded water-filled MPC is removed from the pool, or when the MPC lid is installed in the pool, the combined water, fuel mass, MPC, and HI-TRAC metal will absorb the decay heat emitted by the fuel assemblies. This results in a slow temperature rise of the entire system with time, starting from an initial temperature of the contents. The rate of temperature rise is limited by the thermal inertia of the HI-TRAC system. To enable a bounding heat-up rate determination for the HI-TRAC system, the following conservative assumptions are made:

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

ii. Design-basis maximum decay heat input from the loaded fuel assemblies is imposed on the HI-TRAC transfer cask.

iii. The smaller of the two (i.e., 100-ton and 125-ton) HI-TRAC transfer cask is credited in the analysis. The 100-ton version has a significantly smaller quantity of metal mass, which will result in a higher rate of temperature rise.

iv. A conservatively bounding MPC cavity free volume is considered for flooded water mass.

v. Only fifty percent of the water mass in the MPC cavity is credited towards water thermal inertia evaluation.

Table 4.5.5 summarizes the weights and thermal inertias of several components in the loaded HI-TRAC transfer cask. The rate of temperature rise of the HI-TRAC transfer cask and contents during an adiabatic heat-up is governed by the following equation:

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

where:

$Q =$ decay heat load (Btu/hr)

$C_h =$ combined thermal inertia of the loaded HI-TRAC transfer cask (Btu/°F) [see Table 4.5.5]

$T =$ temperature of the contents (°F)

$t =$ time after HI-TRAC transfer cask is removed from the pool (hr)

A heat-up rate for the HI-TRAC transfer cask contents is determined employing a bounding design heat load (Table 4.4.39, $Q = 1.3 \times 10^5$ Btu/hr) is determined as 4.99 °F/hr. From this

adiabatic rate of temperature rise estimate, the maximum allowable time duration (t_{max}) for fuel to be submerged in water is determined as:

$$t_{max} = \frac{T_{boil} - T_{initial}}{(dT/dt)}$$

where:

T_{boil} = boiling temperature of water (equal to 212°F at the water surface in the MPC cavity)

$T_{initial}$ = initial temperature of the HI-TRAC contents when the transfer cask is removed from the pool

Table 4.5.6 provides a summary of t_{max} at several representative HI-TRAC contents starting temperatures.

In a situation where the maximum allowable time provided in Table 4.5.6 is insufficient to complete all wet fuel handling operations, a suitable means of heat removal such as a forced water circulation shall be initiated and maintained to remove the decay heat from the MPC cavity. In this case, relatively cooler water is introduced via the MPC lid drain port connection and heated water exits from the vent port. The minimum water flow rate required to maintain the MPC cavity water temperature below boiling with an adequate subcooling margin is determined as follows:

$$M_W = \frac{Q}{C_{pW}(T_{max} - T_{in})}$$

where:

M_W = minimum water flow rate (lb/hr)

C_{pw} = water heat capacity (Btu/lb-°F)

T_{max} = maximum MPC cavity water mass temperature

T_{in} = temperature of pool water supply to MPC

As an illustrative example, if the MPC cavity water temperature is limited to 150°F, assuming an MPC inlet water temperature of 125°F and design maximum heat load, the water flow rate is computed as 5200 lb/hr (10.4 gpm). The required minimum flow rate shall be calculated using the actual MPC heat load and the temperature of the cooling water available for this operation.

4.5.5 MPC Drying

4.5.5.1 Drying Options

This FSAR provides for two methods for drying Commercial Spent Fuel (CSF) the MPC cavity, namely:

i. Forced Helium Dehydration
ii. Vacuum Drying

The methods are discussed next.

4.5.5.2 Forced Helium Dehydration

To reduce moisture to trace levels in the MPC using a Forced Helium Dehydration (FHD) system, a closed loop dehumidification system consisting of a condenser, a demoisturizer, a compressor, and a pre-heater is utilized to extract moisture from the MPC cavity through repeated displacement of its contained helium, accompanied by vigorous flow turbulation. Appendix 2.B contains detailed discussion of the design criteria and operation of the FHD system.

The FHD system provides concurrent fuel cooling during the moisture removal process through forced convective heat transfer. The attendant forced convection-aided heat transfer occurring during operation of the FHD system ensures that the fuel cladding temperature will remain below the applicable peak cladding temperature limit for normal conditions of storage, which is well below the high burnup cladding temperature limit in Table 4.3.1 for all combinations of SNF type, burnup, decay heat, and cooling time. Because the FHD operation induces a state of forced convection heat transfer in the MPC, (in contrast to the quiescent mode of natural convection in long term storage), it is readily concluded that the peak fuel cladding temperature under the latter condition will be greater than that during the FHD operation phase. In the event that the FHD system malfunctions, the forced convection state will degenerate to natural convection, which corresponds to the conditions of normal storage. As a result, the peak fuel cladding temperatures will approximate the values reached during normal storage as described elsewhere in this chapter.

4.5.5.3 Vacuum Drying

Because the vacuum drying method of demoisturization leads to a considerable rise in the fuel cladding temperature, threshold heat load limits** that are considerably lower than the MPC design basis heat load are computed for the vacuum drying evolution. The threshold heat loads are very low if one or more high burnup fuel (HBF) assemblies are included in the batch of fuel being loaded. If the fuel batch consists of only MBF, and the limitations on the maximum cladding stress described in Subsection 4.5.2 are met, then a higher threshold heat load meeting the less restrictive temperature limit is permitted.

(a) Analysis

The vacuum condition effective fuel assembly conductivity is determined by procedures discussed earlier (Subsection 4.4.1.1.2) with recognition of the attenuation of thermosiphon effect with the decrease in the quantity of helium and reduction in the conductivity of helium in a most conservative manner. For this purpose, the thermal conductivity of fluid media is set at a miniscule fraction of the helium conductivity. A direct result of this assumption is that the

** See threshold heat load discussion in Subsection 4.5.1

cladding temperatures are exaggerated in the thermal solutions and, correspondingly, threshold heat loads understated.. The MPC basket cross sectional effective conductivity is determined for vacuum conditions according to the procedure discussed in 4.4.1.1.4. Basket periphery-to-MPC shell heat transfer occurs through conduction and radiation. The heat transported to the MPC shell is dissipated from the external surface of the MPC shell to the annulus. It is recognized that the cladding temperature is directly affected by the temperature of the annulus. To ensure a robust margin in the cladding temperatures, vacuum drying is performed with a water cooled annulus. Two options are provided for annulus water cooling:

(i) Standing water in the annulus.
(ii) Annulus gap is water flushed.

An axisymmetric FLUENT thermal model of the MPC in a HI-TRAC is constructed, and peak cladding temperatures at threshold heat loads are obtained. The following conditions are applied to this evaluation:

i. The fuel temperatures rise to their asymptotic steady-state values.

ii. The outer surface of the MPC shell is postulated to be at a bounding temperature of 232°F[††] (standing water in annulus) or 125°F[‡‡] (continuously flushed with water).

iii. The bottom surface of the MPC is insulated.

(b) Results

Table 4.5.11 provides the threshold heat loads at or below which for which vacuum drying is permitted. For completeness, the threshold heat load under the FHD method of drying is also listed (it is equal to the design heat load). The threshold heat load under the vacuum drying condition is a function of two parameters:

i. Maximum burnup in the fuel batch stored
ii. The MPC-HI-TRAC annulus is stagnant or water flushed

As stated earlier, the permissible temperature for a fuel batch containing only MBF can be as high as 570°C if the clad stress criterion in Subsection 4.5.2 is met. The maximum fuel cladding temperature is quite obviously influenced by the thermal state in the annulus: continuous flushing helps reduce the peak cladding temperature. Table 4.5.11 accordingly provides discrete values of the threshold heat loads depending on annulus condition (standing

†† Saturation temperature of water at the bottom of a water filled HI-TRAC annulus.
‡‡ During vacuum operations, water must be circulated at a rate sufficient to ensure maximum annulus temperature is below 125°F. For example given water inlet at 100°F, Q=15 kW and a flush rate of 5 gpm, the maximum temperature (from an adiabatic heat balance for Q = 15 kW) is 120.5°F which is below 125°F.

water or annulus flushing) and fuel burnup.

4.5.6 On-site Transport in HI-TRAC

4.5.6.1 Analysis

An axisymmetric FLUENT thermal model of an MPC inside a HI-TRAC transfer cask was constructed to evaluate temperature distributions for onsite transport scenarios for two HI-TRAC annulus conditions:

(a) Static column of air
(b) Water cooled annulus

Steady-state analyses of the HI-TRAC transfer cask have been performed for the two annulus conditions under all on-site transport scenarios and threshold heat loads obtained. While the duration of onsite transport may be short enough to preclude the MPC and HI-TRAC from obtaining a steady-state, a steady-state analysis is conservative.

4.5.6.2 Results

As stated earlier, the threshold heat loads depend on the orientation of the HI-TRAC. The threshold heat load for vertical transport are greater than that for horizontal transport. Another variable that affects the computed threshold heat load for the on-site transport condition is the maximum burnup in the batch of fuel loaded in the MPC. Finally, if the actual heat generation rate in the MPC exceeds the threshold heat load permitted for the HI-TRAC orientation and burnup state of the CSF batch loaded then annulus cooling is required. Table 4.5.12 provides the threshold heat loads for all on-site transport scenarios.

For both horizontal and vertical mode of on-site transfer of the loaded MPC in the HI-TRAC transfer cask, threshold heat generation rates to meet the HBF fuel clad temperature limit are well below the design basis heat load for the MPC under the normal condition of storage. These threshold heat loads are provided in Table 4.5.12 for vertical and horizontal mode of on-site MPC transfer under steady state conditions. If the heat load of a canister exceeds the threshold value listed in Table 4.5.12, then supplemental cooling of the MPC must be provided to maintain the peak fuel cladding temperature below limit set forth in this FSAR.

Because of the narrow annular space between the MPC and HI-TRAC transfer cask and the availability of a threaded coupling connection in the bottom HI-TRAC lid, it is possible to provide augmented heat removal from the MPC by circulating a coolant through the annular space during MPC transfer operations. Calculations show that even when the threshold heat loads are substantially exceeded, a modest flow of water[§§] is all that is needed to extract sufficient amount of heat to ensure that the peak cladding temperature is below the ISG-11 Rev.

§§ For scenarios that exceed the threshold heat loads by modest amounts, a static column of water in the annulus is adequate.

2 limit adopted in this FSAR. For example, the principal variables and results from an evaluation performed for an MPC-32 at its design basis heat load are summarized in Table 4.5.13 to illustrate the concept. As shown in this table, the peak cladding temperature in the cooled MPC is much below the ISG-11 Rev. 2 limit.

Because the availability of utilities (demineralized water, compressed air, etc.) is plant-specific, it is not possible to design a standard ancillary that can be used at all sites. Nevertheless, the above example serves to demonstrate that the equipment required to effect the necessary heat removal to subcool the MPC during transfer operation for high heat load MPCs will be quite compact and operationally expedient.

It is seen from the above example that even a modest means to cool the external surface of the MPC during on-site transport is sufficient to create a substantial margin in the peak cladding temperature against the permissible limit. This margin may be necessary in certain plants to deal with a short-term handling step during the transfer operation when it may not be practical to maintain auxiliary cooling. Such a situation may occur at certain plants, for example, when the HI-TRAC transfer cask is being positioned over the HI-STORM overpack for the MPC's transfer. In the absence of the auxiliary cooling, if the canister heat load exceeds the Table 4.5.12 threshold heat load, then the fuel cladding temperature will begin to rise. To ensure that the amount of cladding temperature rise is not enough to cause an exceedance of the permissible temperature limit, the MPC must be sufficiently pre-cooled prior to the start of the transfer step when the external cooling becomes unavailable. Calculations show that, with appropriate pre-cooling, a reasonable amount of time to execute an operational step (with the external cooling turned off) can be provided.

To illustrate the MPC heat-up scenario, the water cooled MPC summarized in Table 4.5.13 is analyzed with the HI-TRAC rotated to the horizontal configuration (thermally most adverse configuration) and the auxiliary cooling system turned off. At the beginning of the thermal transient, the MPC is assumed to be in the thermal state condition analyzed above corresponding to the steady state condition with the HI-TRAC vertical and the external cooling operative. The rise in the peak cladding temperature as a function of time is shown in Figure 4.5.4 for this case. This figure shows that 7-1/2 hours are required for the peak cladding temperature to reach 752°F. If a longer period of time were warranted by the operational step, the auxiliary cooling system would be sized to ensure that, prior to initiation of the operation with auxiliary cooling withdrawn, the fuel cladding will be cooled below the applicable clad temperature limit by the required amount. The extent of required pre-cooling, of course, will depend on the MPC heat load, orientation of the HI-TRAC, and the expected duration of the operational step. However, as the above example illustrates, the approach to pre-cool the MPC to maintain the peak cladding temperature below the regulatory limit during a short-time, cooling-unaided operational step is quite feasible.

HI-TRAC cask temperature results are reported for thelimiting sceanario that obtains the highest cladding temperature (Condition 3 and 7, Table 4.5.12). The results are provided in Table 4.5.2

summarizing the maximum calculated temperatures in different parts of the HI-TRAC transfer cask and MPC.

4.5.7 MPC Cooldown and Reflooding for Defueling Operations

NUREG-1536 requires an evaluation of cask cooldown and reflood procedures to support fuel unloading from a dry condition. Past industry experience generally supports cooldown of cask internals and fuel from hot storage conditions by direct water quenching. For high heat load MPCs, the extremely rapid cooldown rates to which the hot MPC internals and the fuel cladding can be subjected during water injection may, however, result in high thermal stresses. Additionally, , water injection may result in large quantities of steam generation. To protect the fuel cladding from high thermal strains under direct water quenching the MPCs are cooled using appropriate means prior to the introduction of water in the MPC cavity space.

Because of the continuous gravity driven circulation of helium in the MPC which results in heated helium gas in sweeping contact with the underside of the top lid and the inner cylindrical surface of the enclosure vessel, utilizing an external cooling means to remove heat from the MPC is quite effective. The external cooling process can be completely non-intrusive such as extracting heat from the outer surface of the enclosure vessel using chilled water. Extraction of heat from the external surfaces of an MPC is very effective largely because of the thermosiphon induced internal transport of heat to the peripheral regions of the MPC. The non-intrusive means of heat removal is preferable to an intrusive process wherein helium is extracted and cooled using a closed loop system such as a Forced Helium Dehydrator (Appendix 2.B), because it eliminates the potential for any radioactive crud to exit the MPC during the cooldown process. Because the optimal method for MPC cooldown is heavily dependent on the location and availability of utilities at a particular nuclear plant, mandating a specific cooldown method cannot be prescribed in this FSAR. Simplified calculations are presented in the following to illustrate the feasibility and efficacy of utilizing an intrusive system such as a recirculating helium cooldown system.

Under a closed-loop forced helium circulation condition, the helium gas is cooled, via an external chiller. The chilled helium is then introduced into the MPC cavity from connections at the top of the MPC lid. The helium gas enters the MPC basket and moves through the fuel basket cells, removing heat from the fuel assemblies and MPC internals. The heated helium gas exits the MPC from the lid connection to the helium recirculation and cooling system. Because of the turbulation and mixing of the helium contents in the MPC cavity by the forced circulation, the MPC exiting temperature is a reliable measure of the thermal condition inside the MPC cavity. The objective of the cooldown system is to lower the bulk helium temperature in the MPC cavity to below the normal boiling temperature of water (212°F). For this purpose, the rate of helium circulation shall be sufficient to ensure that the helium exit gas temperature is below this threshold limit with a margin.

An example calulation for the required helium circulation rate is provided below to limit the helium temperature to 200°F. The calculation assumes no heat loss from the MPC boundaries

and a design maximum heat load*** (1.3×10^5 Btu/hr). Under these assumptions, the MPC helium is heated adiabatically by the MPC decay heat from a given inlet temperature (T1) to a temperature (T2). The required circulation rate to limit T2 to 200°F is computed as follows:

$$m = \frac{Q_d}{C_p(T2 - T1)}$$

where:

Q_d = Design maximum decay heat load (Btu/hr)
m = Minimum helium circulation rate (lb/hr)
Cp = Heat capacity of helium (1.24 Btu/lb-°F (Table 4.2.5))
T1 = Helium supply temperature (assumed 15°F in this example)

Substituting the values for the parameters in the equation above, m is computed as 567 lb/hr.

4.5.8 Minimum Temperatures for On-Site Transport

In Table 2.2.2, the minimum ambient temperature condition required to be considered for the HI-TRAC design is specified as 0°F. If, conservatively, a zero decay heat load (with no solar input) is applied to the stored fuel assemblies then every component of the system at steady state would be at this outside minimum temperature. Provided an antifreeze is added to the water jacket, all HI-TRAC materials will satisfactorily perform their intended functions at this minimum postulated temperature condition.

4.5.9 Evaluation of System Performance for Normal Conditions of Handling and Onsite Transport

The HI-TRAC transfer cask thermal analysis is based on a detailed heat transfer model that conservatively accounts for all modes of heat transfer in various portions of the MPC and HI-TRAC. The thermal model incorporates several conservative features, which are listed below:

i. The most severe levels of environmental factors - bounding ambient temperature (100°F) and constant solar flux - were coincidentally imposed on the thermal design. A bounding solar absorbtivity of 1.0 is applied to all insolation surfaces.

ii. The HI-TRAC cask-to-MPC annular gap is analyzed based on the nominal design dimensions. No credit is considered for the significant reduction in this radial gap that would occur as a result of differential thermal expansion with design basis fuel at hot conditions. The MPC is considered to be concentrically aligned with the cask cavity. This

*** Table 4.4.39 lists MPC design heat loads. From this Table, the maximum heat load (38 kW) is used in this evaluation.

is a worst-case scenario since any eccentricity will improve conductive heat transport in this region.

iii. No credit is considered for cooling of the HI-TRAC baseplate while in contact with a supporting surface. An insulated boundary condition is applied in the thermal model on the bottom baseplate face.

Temperature distribution results (Table 4.5.2) obtained from this highly conservative thermal model show that the short-term fuel cladding and cask component temperature limits are met with adequate margins. Expected margins during normal HI-TRAC use will be larger due to the many conservative assumptions incorporated in the analysis. Corresponding MPC internal pressure evaluation shows that the MPC confinement boundary remains well below the short-term condition design pressure. The maximum local axial neutron shield temperature is lower than design limits. Therefore, it is concluded that the HI-TRAC transfer cask thermal design is adequate to maintain fuel cladding integrity for short-term operations.

The water in the water jacket of the HI-TRAC provides necessary neutron shielding. During normal handling and onsite transfer operations this shielding water is contained within the water jacket, which is designed for an elevated internal pressure. It is recalled that the water jacket is equipped with pressure relief valves to retain pressure up to 60 psig thereby precluding boiling in the water jacket under normal conditions. Under normal handling and onsite transfer operations, the bulk temperature inside the water jacket reported in Table 4.5.2 is less than the coincident saturation temperature at 60 psig (307°F), so the shielding water remains in its liquid state. The bulk temperature is determined via a conservative analysis, presented earlier, with design-basis maximum decay heat load. One of the assumptions that render the computed temperatures extremely conservative is the stipulation of a 100°F steady-state ambient temperature. In view of the large thermal inertia of the HI-TRAC, an appropriate ambient temperature is the "time-averaged" temperature, formally referred to in this FSAR as the normal temperature.

Table 4.5.1

[INTENTIONALLY DELETED]

Table 4.5.2

HI-TRAC TRANSFER CASK STEADY-STATE
MAXIMUM TEMPERATURES FOR LIMITING SCENARIOS

Component	High Burnup Fuel (Conditions 3 and 7, Table 4.5.12) Temperature [°F]
Fuel Cladding	745
MPC Basket	728
Basket Periphery	620
MPC Outer Shell Surface	433
HI-TRAC Overpack Inner Surface	305
Water Jacket Inner Surface	286
Enclosure Shell Outer Surface	265
Water Jacket Bulk Water	246
Axial Neutron Shield†	288

† Local neutron shield section temperature.

Table 4.5.3 and 4.5.4

[INTENTIONALLY DELETED]

Table 4.5.5

SUMMARY OF LOADED 100-TON HI-TRAC TRANSFER CASK
BOUNDING COMPONENT
WEIGHTS AND THERMAL INERTIAS

Component	Weight (lbs)	Heat Capacity (Btu/lb-°F)	Thermal Inertia (Btu/°F)
Water Jacket	7,000	1.0	7,000
Lead	52,000	0.031	1,612
Carbon Steel	40,000	0.1	4,000
Alloy-X MPC (empty)	39,000	0.12	4,680
Fuel	40,000	0.056	2,240
MPC Cavity Water†	6,500	1.0	6,500
			26,032 (Total)

† Conservative lower bound water mass.

Table 4.5.6

MAXIMUM ALLOWABLE TIME DURATION FOR WET TRANSFER OPERATIONS

Initial Temperature††† (°F)	Time Duration (hr)
115	19.4
120	18.4
125	17.4
130	16.4
135	15.4
140	14.4
145	13.4
150	12.4

††† Pool water temperature during fuel loading.

Tables 4.5.7 through 4.5.10

INTENTIONALLY DELETED

Table 4.5.11

THRESHOLD HEAT LOADS FOR FUEL DRYING

Condition No.	Drying Method	Threshold Heat Load (Q_T)	Burnup State	Is Annulus Flush Required?	Is Hoop Stress Compliance Required?	Cladding Temperature Limit [°F]	Computed Maximum Cladding Temperature [°F]
1	FHD	Q_d‡‡‡	MBF	No	No	752	Note 1
2	FHD	Q_d	HBF	No	No	752	Note 1
3	VD	9 kW	HBF	No	No	752	729
4	VD	10 kW	HBF	Yes	No	752	740
5	VD	17 kW	MBF	No	Yes	1058	1042
6	VD	18 kW	MBF	Yes	Yes	1058	1056
7	VD	9 kW	MBF	No	No	752	729
8	VD	10 kW	MBF	Yes	No	752	740

Note 1: Under the FHD method for MPC drying, an externally driven circulation of helium ensures drying conditions in the MPC, which are in the neighborhood of the saturation temperature of water at the prevailing pressure (about 350°F). As such the operating clad temperatures are substantially below the cladding temperature limit (752°F) thereby ensuring a hospitable thermal environment for HBF.

Acronyms:
FHD – Forced Helium Dehydration
VD – Vacuum Drying
MBF – Moderate Burnup Fuel
HBF – High Burnup Fuel

‡‡‡ Design heat load (Table 4.4.39).

Table 4.5.12

PERMISSIBLE HEAT LOADS FOR ON-SITE TRANSPORT

Condition No.	HI-TRAC Orientation	Threshold Heat Load (Q_T)	Fuel burnup	Is Hoop Stress Compliance Required?	Is Annulus Cooling Required?	Temperature Limit (°F)	Computed Maximum Cladding Temperature [°F]
1	Horizontal	19 kW	HBF	No	No	752	735
2	Horizontal	Q_d§§§	HBF	No	Yes	752	Note 1
3	Vertical	23 kW	HBF	No	No	752	745
4	Vertical	Q_d	HBF	No	Yes	752	Note 1
5	Horizontal	19 kW	MBF	No	No	752	735
6	Horizontal	Q_d	MBF	No	Yes	752	Note 1
7	Vertical	23 kW	MBF	No	No	752	745
8	Vertical	Q_d	MBF	No	Yes	752	Note 1

Note 1: The maximum cladding temperature when annulus cooling is required will be dependent on site specifics such as decay heat of loaded SNF and availability of cooling fluid (air or water). Each user shall confirm that the annulus cooling provided is sufficient to keep the cladding temperatures below their applicable limits. An annulus cooling example is provided in Subsection 4.5.6.2.

Acronyms:
FHD – Forced Helium Dehydration
VD – Vacuum Drying
MBF – Moderate Burnup Fuel
HBF – High Burnup Fuel

§§§ Design heat load (Table 4.4.39).

Table 4.5.13

HI-TRAC ANNULUS COOLING EXAMPLE

MPC Type	MPC-32
Design Heat Load	38 kW
Transfer Cask Orientation	Vertical
Threshold Heat Load per Table 4.5.12	23kW
Coolant (Water) Flow Rate (assumed)	10 gpm
Coolant Inlet Temperature, °F (assumed)	80
Coolant Outlet Temperature, °F (calculated)	106
Peak Fuel Cladding Temperature, °F (calculated)	385

Figures 4.5.1 through 4.5.3 Intentionally Deleted

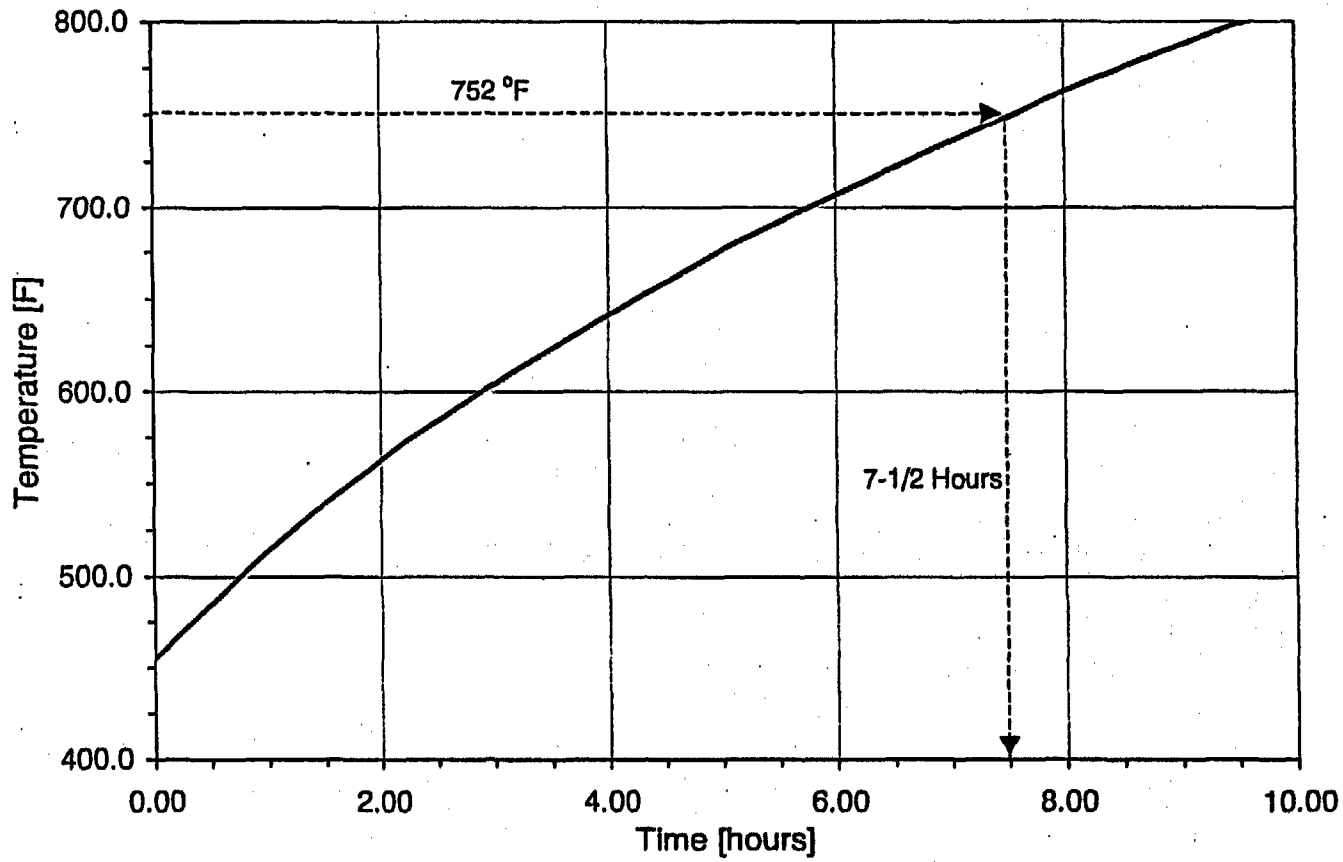


Figure 4.5.4: PEAK CLADDING TEMPERATURE IN A HORIZONTAL HI-TRAC CONFIGURATION.

4.6 REGULATORY COMPLIANCE

4.6.1 Normal Conditions of Storage

NUREG-1536 [4.4.10] *and ISG-11 [4.1.4]* defines several thermal acceptance criteria that must be applied to evaluations of normal conditions of storage. These items are addressed in Sections 4.1 through 4.4.5 *and results evaluated in Subsection 4.4.6.* Each of the pertinent criteria and the conclusion of the evaluations are summarized here.

As required by *ISG-11 [4.1.4]* ~~NUREG-1536 (4.0,IV,1)~~, the fuel cladding temperature at the beginning of dry cask storage is maintained below the anticipated damage-threshold temperatures for normal conditions *for the licensed life of the HI-STORM system.* ~~and a minimum of 20 years of cask storage.~~ Maximum clad temperatures for long-term storage conditions are reported in Section 4.4.2. ~~Anticipated damage-threshold temperatures, calculated as described in Section 4.3, are summarized in Table 2.2.3.~~

As required by NUREG-1536 (4.0,IV,3), the maximum internal pressure of the cask remains within its design pressure for normal, off-normal, and accident conditions, assuming rupture of 1 percent, 10 percent, and 100 percent of the fuel rods, respectively. Assumptions for pressure calculations include release of 100 percent of the fill gas and 30 percent of the significant radioactive gases in the fuel rods. Maximum internal pressures are reported in Section 4.4.4. Design pressures are summarized in Table 2.2.1.

As required by NUREG-1536 (4.0,IV,4), all cask and fuel materials are maintained within their minimum and maximum temperature for normal and off-normal conditions in order to enable components to perform their intended safety functions. Maximum and minimum temperatures for long-term storage conditions are reported in Sections 4.4.2 and 4.4.3, respectively. Design temperature limits are summarized in Table 2.2.3. HI-STORM System components defined as important to safety are listed in Table 2.2.6.

As required by NUREG-1536 (4.0,IV,5), the cask system ensures a very low probability of cladding breach during long-term storage. ~~Further, NUREG-1536 (4.0,IV,6) requires that the fuel cladding damage resulting from creep cavitation should be limited to 15 percent of the original cladding cross section area during dry storage.~~ *For long term normal and off-normal operations, the maximum CSF cladding temperature is below the ISG-11 [4.1.4] limit of 400°C (752°F).*
~~The calculation methodology, described in Section 4.3, for determining initial dry storage fuel clad temperature limits, ensures that both of these requirements are satisfied. Maximum fuel clad temperature limits are summarized in Table 2.2.3.~~

As required by NUREG-1536 (4.0,IV,7), the cask system is passively cooled. All heat rejection mechanisms described in this chapter, including conduction, natural convection, and thermal radiation, are completely passive.

As required by NUREG-1536 (4.0,IV,8), the thermal performance of the cask is within the allowable

design criteria specified in FSAR Chapters 2 and 3 for normal conditions. All thermal results reported in Sections 4.4.2 through 4.4.5 are within the design criteria allowable ranges for all normal conditions of storage.

4.6.2 ~~Normal Handling and Onsite Transfer~~ *Short Term Operations*

As discussed in Section 4.1, evaluation of short term operations is presented in Section 4.5. This section establishes complete compliance with the provisions of ISG-11 [4.1.4]. In particular the ISG-11 requirement to ensure that maximum cladding temperatures under all fuel loading and short term operations be below 400°C (752°F) is demonstrated as stated below.

~~NUREG-1536 [4.4.10] defines several thermal acceptance criteria that are addressed in Sections 4.5.1 through 4.5.5. Each of the pertinent criteria is summarized here.~~

~~As required by NUREG-1536 (4.0,IV,2),~~ *As required by ISG-11* the fuel cladding temperature is maintained below *400°C (752°F)* ~~570°C (1058°F)~~ for fuel transfer operations. Maximum clad temperatures for *short term operations (on-site transport and vacuum drying conditions)* ~~normal on-site transfer conditions~~ are reported in Sections *4.5.6 and 4.5.5 respectively.* ~~4.5.2. Maximum~~ *The* clad temperatures ~~for vacuum drying conditions are reported in Section 4.5.2.1 and~~ *for short term operations* comply *with the ISG-11 limits.* ~~within this limit by large conservative margins~~.

~~As required by NUREG-1536 (4.0,IV,3), the maximum internal pressure of the cask remains within its design pressure for normal and off-normal conditions, assuming rupture of 1 percent and 10 percent of the fuel rods, respectively. Assumptions for pressure calculations include release of 100 percent of the fill gas and 30 percent of the significant radioactive gases in the fuel rods. Maximum internal pressures are reported in Section 4.5.4. Design pressures are summarized in Table 2.2.1.~~

~~As required by NUREG-1536 (4.0,IV,4), all cask and fuel materials are maintained within their minimum and maximum temperature for normal (short-term) fuel handling operations in order to enable components to perform their intended safety functions. Maximum and minimum temperatures for fuel handling operations are reported in Sections 4.5.2 and 4.5.3, respectively. Design temperature limits are summarized in Table 2.2.3.~~

~~As required by NUREG-1536 (4.0,IV,7), the cask system is passively cooled. All heat rejection mechanisms described in this chapter, including conduction, natural convection, and thermal radiation, are completely passive.~~

~~As required by NUREG-1536 (4.0,IV,8), the thermal performance of the cask is within the allowable design criteria specified in FSAR Chapters 2 and 3 for normal (short-term) fuel handling operations. All thermal results reported in Sections 4.5.2 through 4.5.5 are within the design criteria allowable ranges for short-term conditions.~~

4.7 REFERENCES

[4.1.1] ANSYS Finite Element Modeling Package, Swanson Analysis Systems, Inc., Houston, PA, 1993.

[4.1.2] FLUENT Computational Fluid Dynamics Software, Fluent, Inc., Centerra Resource Park, 10 Cavendish Court, Lebanon, NH 03766.

[4.1.3] "The TN-24P PWR Spent-Fuel Storage Cask: Testing and Analyses," EPRI NP-5128, (April 1987).

[4.1.4] "Cladding Considerations for the Transportation and Storage of Spent Fuel", Interim Staff Guidance – 11, Revision 2, (7/30/02).

[4.1.5] "Topical Report on the HI-STAR/HI-STORM Thermal Model and its Benchmarking with Full-Size Cask Test Data", Holtec Report HI-992252, Rev. 1.

[4.2.1] Baumeister, T., Avallone, E.A. and Baumeister III, T., "Marks' Standard Handbook for Mechanical Engineers," 8th Edition, McGraw Hill Book Company, (1978).

[4.2.2] Rohsenow, W.M. and Hartnett, J.P., "Handbook of Heat Transfer," McGraw Hill Book Company, New York, (1973).

[4.2.3] Creer et al., "The TN-24P Spent Fuel Storage Cask: Testing and Analyses," EPRI NP-5128, PNL-6054, UC-85, (April 1987).

[4.2.4] Rust, J.H., "Nuclear Power Plant Engineering," Haralson Publishing Company, (1979).

[4.2.5] Kern, D.Q., "Process Heat Transfer," McGraw Hill Kogakusha, (1950).

[4.2.6] "A Handbook of Materials Properties for Use in the Analysis of Light Water Reactor Fuel Rod Behavior," NUREG/CR-0497, (August 1981).

[4.2.7] "Safety Analysis Report for the NAC Storable Transport Cask," Docket No. 71-9235.

[4.2.8] ASME Boiler and Pressure Vessel Code, Section II, Part D, (1995).

[4.2.9] Jakob, M. and Hawkins, G.A., "Elements of Heat Transfer," John Wiley & Sons, New York, (1957).

[4.2.10] ASME Steam Tables, 3rd Edition (1977).

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

[4.3.3] *Deleted* ~~"Spent Fuel Heat Generation in an Independent Spent Fuel Storage Installation," Regulatory Guide 3.54, Revision 1, (January 1999).~~

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

[4.3.5] *Deleted* ~~Schwartz, M.W., Witte, M.C., Lawrence Livermore National Laboratory, "Spent Fuel Cladding Integrity During Dry Storage," UCID-21181.~~

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

[4.3.7] *Deleted* ~~Schemel, J.H., "ASTM Manual on Zirconium and Hafnium," STP-639, American Society for Testing and Materials, (December 1977).~~

[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] Siegel, R. and Howell, J.R., "Thermal Radiation Heat Transfer," 2nd Edition, McGraw Hill (1981).

[4.4.4] *Deleted* ~~Holman, J.P., "Heat Transfer," 6th ed., McGraw-Hill Book Company, (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-1019, UC-820, page II-127, (November 1992).

[4.4.6] *Deleted* ~~Hewitt, G.F., Shires, G.L. and Bott, T.R., "Process Heat Transfer," CRC Press, (1994).~~

[4.4.7] *Deleted* ~~Hagrman, Reymann and Mason, "MATPRO-Version 11 (Revision 2) A Handbook of Materials Properties for Use in the Analysis of Light Water Reactor Fuel Rod Behavior," NUREG/CR-0497, Tree-1280, Rev. 2, EG&G Idaho, August 1981.~~

[4.4.8] *Deleted* ~~"Effective Thermal Conductivity and Edge Conductance Model for a Spent Fuel Assembly," R. D. Manteufel & N. E. Todreas, Nuclear Technology, 105, 421-440,~~

~~(March 1994).~~

[4.4.9] *Deleted* ~~"Spent Nuclear Fuel Effective Thermal Conductivity Report," US DOE Report BBA000000-01717-5705-00010 REV 0, (July 11, 1996).~~

[4.4.10] NUREG-1536, "Standard Review Plan for Dry Cask Storage Systems," USNRC, (January 1997).

[4.4.11] *Deleted* ~~"Fuel Cladding Cladding Temperatures in Transport and Storage Casks Development and Validation of a Computation Method," S. Anton, Ph.D. Thesis (German) RWTH Aachen, Germany, 1997.~~

[4.4.12] *Deleted* ~~"Topical Report on the HI-STAR/HI-STORM Thermal Model and its Benchmarking with Full-Size Cask Test Data", Holtec Report HI-992252, Rev. 1.~~

APPENDIX 4.A

[THIS APPENDIX HAS BEEN DELETED IN ITS ENTIRETY]

APPENDIX 4.B: CONSERVATISMS IN THE THERMAL ANALYSIS OF THE HI-STORM 100 SYSTEM

4.B.1 OVERVIEW OF CASK HEAT REMOVAL SYSTEM

The HI-STORM 100 overpack is a large, cylindrical structure with an internal cavity suited for emplacement of a cylindrical canister containing spent nuclear fuel (SNF). The canister is arrayed in an upright manner inside the vertically oriented overpack. The design of the system provides for a ~~small~~ *narrow* radial gap between the canister and the cylindrical overpack cavity. One principal function of a fuel storage system is to provide a means for ensuring fuel cladding integrity under long-term storage. ~~periods (20 years or more)~~. The HI-STORM 100 overpack is equipped with four large ducts near its bottom and top extremities. The ducted overpack construction, together with an engineered annular space between the MPC cylinder and internal cavity in the HI-STORM 100 overpack structure, ensures a passive means of heat dissipation from the stored fuel via ventilation action (i.e., natural circulation of air in the canister-to-overpack annulus). In this manner a large structure physically interposed between the hot canister and ambient air (viz. the concrete overpack engineered for radiation protection) is rendered as an air flow device for convective heat dissipation. The pertinent design features producing the air ventilation ("chimney effect") in the HI-STORM 100 cask are shown in Figure *4.0.1.* ~~4.B.1~~.

A great bulk of the heat emitted by the SNF is rejected to the environment (Q_1) by convective action. A small quantity of the total heat rejection occurs by natural convection and radiation from the surface of the overpack (Q_2), and an even smaller amount is dissipated by conduction to the concrete pad upon which the HI-STORM 100 overpack is placed (Q_3). From the energy conservation principle, the sum of heat dissipation to all sinks (convective cooling (Q_1), surface cooling (Q_2) and cooling to pad (Q_3)) equals the sum of decay heat emitted from the fuel stored in the canister (Q_d) and the heat deposited by insolation *(Q_s)*, ~~Q_s~~ (i.e., $Q_d + Q_s = Q_1 + Q_2 + Q_3$). This situation is illustrated in Figure 4.B.2. In the HI-STORM 100 System, Q_1 is by far the dominant mode of heat removal, accounting for well over 80% of the decay heat conveyed to the external environment. Figure 4.B.3 shows the relative portions of Q_d transferred to the environs via Q_1, Q_2, and Q_3 in the HI-STORM 100 System under the design basis heat load.

The heat removal through convection, Q_1, is similar to the manner in which a fireplace chimney functions: ~~Air~~ *air* is heated in the annulus between the canister and the overpack through contact with the canister's hot cylindrical surface causing it to flow upward toward the top (exit) ducts and inducing the suction of the ambient air through the bottom ducts. The flow of air sweeping past the cylindrical surfaces of the canister has sufficient velocity to create turbulence that aids in the heat extraction process. It is readily recognized that the chimney action relies on a fundamental and immutable property of air, namely that air becomes lighter (i.e., more buoyant) as it is heated. If the canister contained no heat emitting fuel, then there would be no means for the annulus air to heat and rise. Similarly, increasing the quantity of heat produced in the canister would make more heat available for heating of annulus air, resulting in a more vigorous chimney action. Because the heat energy of the spent nuclear fuel itself actuates the chimney action, ventilated overpacks of the HI-STORM 100 genre are considered absolutely safe against thermal

malfunction. While the removal of heat through convective mass transport of air is the dominant mechanism, other minor components, labeled Q_2 and Q_3 in the foregoing, are recognized and quantified in the thermal analysis of the HI-STORM 100 System.

Heat dissipation from the exposed surfaces of the overpack, Q_2, occurs principally by natural convection and radiation cooling. The rate of decay heat dissipation from the external surfaces is, of course, influenced by several factors, some of which aid the process (e.g., wind, thermal turbulation of air), while others oppose it (for example, radiant heating by the sun or blocking of radiation cooling by surrounding casks). In this appendix, the relative significance of Q_2 and Q_3 and the method to conservatively simulate their effect in the HI-STORM 100 thermal model is discussed.

The thermal problem posed for the HI-STORM 100 System *thermal design* ~~in the system's Final Safety Analysis Report (FSAR)~~ is as follows: Given a specified maximum fuel cladding temperature, T_c, and a specified ambient temperature, T_a what is the maximum permissible heat generation rate Q_d, in the canister under steady state conditions? Of course, in the real world, the ambient temperature, T_a, varies continuously, and the cask system is rarely in a steady state (i.e., temperatures vary with time). Fortunately, fracture mechanics of spent fuel cladding instruct us that it is the time-integrated effect of elevated temperature, rather than an instantaneous peak value, that determines whether fuel cladding would rupture. The most appropriate reference ambient temperature for cladding integrity evaluation, therefore, is the <u>average</u> ambient temperature for the <u>entire</u> duration of dry storage. For conservatism, the reference ambient temperature is, however, selected to be the maximum yearly average for the ISFSI site. In the general certification of HI-STORM 100, the reference ambient temperature (formally referred to as the normal temperature) is set equal to 80°F, which is greater than the annual average for any power plant location in the U.S.*

The thermal analysis of the cask system leads to a computed value of the fuel cladding temperature greater than T_a by an amount C. In other words, $T_c = T_a + C$, where C decreases slightly as T_a (~~assumed~~ ambient temperature) is increased. The thermal analysis of HI-STORM 100 is carried out to compute C in a most conservative manner. In other words, the mathematical model seeks to calculate an upper bound on the value of C.

Dry storage scenarios are characterized by relatively large temperature elevations (C) above ambient (650°F or so). The cladding temperature rise is the cumulative sum of temperature increments arising from individual elements of thermal resistance. To protect cladding from overheating, analytical assumptions adversely impacting heat transfer are chosen with particular attention given to those temperature increments which form the bulk of the temperature rise. In this appendix, the principal conservatisms in the thermal modeling of the HI-STORM 100 System and their underlying theoretical bases are presented. This overview is intended to provide

* According to the U.S. National Oceanic and Atmospheric Administration (NOAA) publication, "Comparative Climatic Data for the United States through 1998", the highest annual average temperature for any location in the continental U.S. is 77.8°F in Key West, Florida.

a physical understanding of the large margins ~~buried~~ *embedded* in the HI-STORM 100 design which are summarized in Section 4.4.6 of this FSAR.

4.B.2 CONSERVATISM IN ENVIRONMENTAL CONDITION SPECIFICATION

The ultimate heat sink for decay heat generated by stored fuel is ambient air. ~~The HI-STORM 100 System defines three ambient temperatures as the environmental conditions for thermal analysis. These are, the Normal (80°F), the Off Normal (100°F) and Extreme Hot (125°F) conditions.~~ Two factors dictate the stipulation of an ambient temperature for cladding integrity calculations. One factor is that ambient temperatures are constantly cycling on a daily basis (night and day). Furthermore, there are seasonal variations (summer to winter). The other factor is that cladding degradation is an incremental process that, over a long period of time (20 years), has an accumulated damage resulting from an "averaged-out" effect of the environmental temperature history. The 80°F normal temperature stated in the HI-STORM 100 FSAR is defined as the highest annual average temperature at a site established from past records. This is a principal design parameter in the HI-STORM 100 analysis because it establishes the basis for demonstrating long-term SNF integrity. The choice of maximum annual average temperature is conservative *as* ~~for a 20 year period. Based~~ *based* on meteorological data, the 80°F ~~is chosen to~~ bound*s* annual average temperatures reported within the continental US.

For short periods, it is recognized that ambient temperature excursions above 80°F are possible. Two scenarios are postulated and analyzed in the FSAR to bound such transient events. The Off-Normal (100°F) and Extreme Hot (125°F)* cases are postulated as continuous (72-hour average) conditions. Both cases are analyzed as steady-state conditions (i.e., thermal inertia of the considerable concrete mass, fuel and metal completely neglected) occurring at the start of dry storage when the decay heat load to the HI-STORM 100 System is at its peak value with fuel emitting heat at its design basis maximum level.

4.B.3 CONSERVATISM IN MODELING THE ISFSI ARRAY

Traditionally, in the classical treatment of the ventilated storage cask thermal problem, the cask to be analyzed (the subject cask) is modeled as a stand-alone component that rejects heat to the ambient air through chimney action (Q_1), by natural convection to quiescent ambient air and radiation to the surrounding open spaces (Q_2), and finally, a small amount through the concrete pad into the ground (Q_3). The contributing effect of the sun (addition of heat) is considered, but the dissipative effect of wind is neglected. The interchange of radiative heat between proximate casks is also neglected (the so-called "cask-to-cask interactions"). In modeling the HI-STORM 100 System, Holtec International extended the classical cask thermal model to include the effect of the neighboring casks in a most conservative manner. This model represents the flow of supply air to the inlet ducts for the subject cask by erecting a cylinder around the subject cask. The model blocks all lateral flow of air from the surrounding space into the subject cask's inlet ducts. This mathematical artifice is illustrated in Figure 4.B.4, where the lateral air flow arrows

* According to NOAA, the highest daily mean temperature for any location in the continental U.S. is 93.7°F, which occurred in Yuma, Arizona.

are shown "dotted" to indicate that the mathematical cylinder constructed around the cask has blocked off the lateral flow of air. Consequently, the chimney air must flow down the annulus from the air plenum space above the casks, turn around at the bottom and enter the inlet ducts. Because the vertical downflow of air introduces additional resistance to flow, an obvious effect of the hypothetical enclosing cylinder construct is an increased total resistance to the chimney flow which, it is recalled, is the main heat conveyance mechanism in a ventilated cask. Throttling of the chimney flow by the hypothetical enclosing cylinder is an element of conservatism in the HI-STORM modeling.

~~Thus, whereas air flows toward the bottom ducts from areas of supply which are scattered in a three dimensional continuum with partial restriction from neighboring casks, the analytical model blocks the air flow completely from areas outside the hypothetical cylinder. This is illustrated in Figure 4.B.4 in which an impervious boundary is shown to limit HI-STORM 100 cask access to fresh air from an annular opening near the top.~~

Thus, in the HI-STORM model, the feeder air to the HI-STORM 100 System must flow down the hypothetical annulus sweeping past the external surface of the cask. The ambient air, assumed to enter this hypothetical annulus at the assumed environmental temperature, heats by convective heat extraction from the overpack before reaching the bottom (inlet) ducts. In this manner, the temperature of the feeder air into the ducts is maximized. In reality, the horizontal flow of air in the vicinity of the inlet ducts, suppressed by the enclosed cylinder construct (as shown in Figure 4.B.4) would act to mitigate the pre-heating of the feeder air. By maximizing the extent of air preheating, the computed value of ventilation flow is underestimated in the simulation.

4.B.4 CONSERVATISM IN RADIANT HEAT LOSS

In an array of casks, the external (exposed) cask surfaces have a certain "view" of each other. The extent of view is a function of relative geometrical orientation of the surfaces and presence of other objects between them. The extent of view influences the rate of heat exchange between surfaces by thermal radiation. The presence of neighboring casks also partially blocks the escape of radiant heat from a cask thus affecting its ability to dissipate heat to the environment. This aspect of Radiative Blocking (RB) is illustrated for a reference cask (shown shaded) in Figure 4.B.5. It is also apparent that a cask is a recipient of radiant energy from adjacent casks (Radiant Heating (RH)). Thus, a thermal model representative of a cask array must address the RB and RH effects in a conservative manner. To bound the physical situation, a Hypothetical Reflecting Boundary (HRB) modeling feature is introduced in the thermal model. The HRB feature surrounds the HI-STORM 100 overpack with a reflecting cylindrical surface with the boundaries insulated.

In Figures 4.B.6 and 4.B.7 the inclusion of RB and RH effects in the HI-STORM 100 modeling is graphically illustrated. Figure 4.B.6 shows that an incident ray of radiant energy leaving the cask surface bounces back from the HRB thus preventing escape (i.e., RB effect maximized). The RH effect is illustrated in Figure 4.B.7 by superimposing on the physical model reflected images of HI-STORM 100 cask surrounding the reference cask. A ray of radiant energy from an adjacent cask directed toward the reference cask (AA) is duplicated by the model via another ray

of radiant energy leaving the cask (BB) and being reflected back by the HRB (BA'). A significant feature of this model is that the reflected ray (BA') *is* initiated from a cask surface (reference cask) assumed to be loaded with design basis maximum heat (hottest surface temperature). As the strength of the ray is directly proportional to the fourth power of surface temperature, radiant energy emission from an adjacent cask at a lower heat load will be overestimated by the HRB construct. In other words, the reference cask is assumed to be in an array of casks all producing design basis maximum heat. Clearly, it is physically impossible to load every location of every cask with fuel emitting heat at design basis maximum. Such a spent fuel inventory does not exist. This bounding assumption has the effect of maximizing cask surface temperature as the possibility of "hot" (design basis) casks being radiatively cooled by adjacent casks is precluded. The HRB feature included in the HI-STORM 100 model thus provides a bounding effect of an infinite array of casks, all at design basis maximum heat loads. No radiant heat is permitted to escape the reference cask (bounding effect) and the reflecting boundary mimics incident radiation toward the reference casks around the 360° circumference (bounding effect).

4.B.5 CONSERVATISM IN ~~REPRESENTING~~ *MODELING THE MPC TOP PLENUM*~~BASKET AXIAL RESISTANCE~~

The top region of the MPC contains an open space between the underside of the MPC lid and the top of the fuel basket. In addition the top of the fuel basket contains mouseholes. For conservatism, the open space portion of the top plenum is neglected in the MPC thermal model. This results in added resistance to the thermosiphon action in the plenum region, which overstates the cladding temperatures in the thermal models.

~~As stated earlier, the largest fraction of the total resistance to the flow of heat from the spent nuclear fuel (SNF) to the ambient is centered in the basket itself. Out of the total temperature drop of approximately 650°F (C=650°F) between the peak fuel cladding temperature and the ambient, over 400°F occurs in the fuel basket. Therefore, it stands to reason that conservatism in the basket thermal simulation would have a pronounced effect on the conservatism in the final solution. The thermal model of the fuel basket in the HI-STORM 100 FSAR was accordingly constructed with a number of conservative assumptions that are described in the HI-STORM 100 FSAR. We illustrate the significance of the whole array of conservatisms by explaining one in some detail in the following discussion.~~

~~It is recognized that the heat emission from a fuel assembly is axially non-uniform. The maximum heat generation occurs at about the mid-height region of the enriched uranium column, and tapers off toward its extremities. The axial heat conduction in the fuel basket would act to diffuse and levelize the temperature field in the basket. The axial conductivity of the basket, quite clearly, is the key determinant in how well the thermal field in the basket would be homogenized. It is also evident that the conduction of heat along the length of the basket occurs in an uninterrupted manner in a HI-STORM 100 basket because of its continuously welded honeycomb geometry. On the other hand, the in-plane transfer of heat must occur through the physical gaps that exist between the fuel rods, between the fuel assembly and the basket walls and between the basket and the MPC shell. These gaps depress the in-plane conductivity of the basket. However, in the interest of conservatism, only a small fraction of the axial conductivity of~~

~~the basket is included in the HI-STORM 100 thermal model. This assumption has the direct effect of throttling the axial flow of heat and thus of elevating the computed value of mid-height cladding temperature (where the peak temperature occurs) above its actual value. In actuality, the axial conductivity of the fuel basket is much greater than the in-plane conductivity due to the continuity of the fuel and basket structures in that direction. Had the axial conductivity of the basket been modeled less conservatively in the HI-STORM 100 thermal analysis, then the temperature distribution in the basket will be more uniform, i.e., the bottom region of the basket would be hotter than that computed. This means that the temperature of the MPC's external surface in the bottom region is hotter than computed in the HI-STORM 100 analysis. It is a well-known fact in ventilated column design that the lower the location in the column where the heat is introduced, the more vigorous the ventilation action. Therefore, the conservatism in the basket's axial conductivity assumption has the net effect of reducing the computed ventilation rate.~~

~~To estimate the conservatism in restricting the basket axial resistance, we perform a numerical exercise using mathematical perturbation techniques. The axial conductivity (K_z) of the MPC is, as explained previously, much higher than the in-plane (K_r) conductivity. The thermal solution to the MPC anisotropic conductivities problem (i.e. K_z and K_r are not equal) is mathematically expressed as a sum of a baseline isotropic solution T_o (setting $K_z = K_r$) and a perturbation T^* which accounts for anisotropic effects. From Fourier's Law of heat conduction in solids, the perturbation equation for T' is reduced to the following form:~~

$$\cancel{K_z \frac{d^2T^*}{dz^2} = -\Delta K \frac{d^2T_o}{dz^2}}$$

~~Where, ΔK is the perturbation parameter (i.e. axial conductivity offset $\Delta K = K_z - K_r$). The boundary conditions for the perturbation solution are zero slope at peak cladding temperature location ($dT^*/dz = 0$) (which occurs at about the top of the active fuel height) and $T^* = 0$ at the bottom of the active fuel length. The object of this calculation is to compute T^* where the peak fuel cladding temperature is reached. To this end, the baseline thermal solution T_o (i.e. HI-STORM isotropic modeling solution) is employed to compute an appropriate value for d^2T_o/dz^2 which characterizes the axial temperature rise over the height of the active fuel length in the hottest fuel cell. This is computed as ($\Delta T_{ax}/L^2$) where ΔT_{ax} is the fuel cell temperature rise and L is the active fuel length. Conservatively postulating a lower bound ΔT_{ax} of 200°F and L of 12 ft, d^2T_o/dz^2 is computed as 1.39°F/ft^2. Integrating the perturbation equation shown above, the following formula for T^* is obtained:~~

$$\cancel{T^* = \left(\frac{\Delta K}{K_z}\right)\frac{d^2T_o}{dz^2}L^2}$$

~~Employing a conservative low value for the ($\Delta K/K_z$) parameter of 0.15, T^* is computed as 30°F. In other words, the baseline HI-STORM solution over predicts the peak cladding temperature by approximately 30°F.~~

4.B.6 HEAT DISSIPATION UNDERPREDICTION IN THE MPC DOWNCOMER

Internal circulation of helium in the sealed MPC is modeled as flow in a porous medium in the fueled region containing the SNF (including top and bottom plenums). The basket-to-MPC shell

clearance space is modeled as a helium filled radial gap to include the downcomer flow in the thermal model. The downcomer region, as illustrated in Figure 4.4.2, consists of an azimuthally varying gap formed by the square-celled basket outline and the cylindrical MPC shell. At the locations of closest approach a differential expansion gap (a small clearance on the order of 1/10 of an inch) is engineered to allow free thermal expansion of the basket. At the widest locations, the gaps are on the order of the fuel cell opening (~6" (BWR) and ~9" (PWR) MPCs). It is heuristically evident that heat dissipation by conduction is maximum at the closest approach locations (low thermal resistance path) and that convective heat transfer is highest at the widest gap locations (large downcomer flow). In the FLUENT thermal model, a radial gap that is large compared to the basket-to-shell clearance and small compared to the cell opening is used. As a relatively large gap penalizes heat dissipation by conduction and a small gap throttles convective flow, the use of a single gap in the FLUENT model understates both conduction and convection heat transfer in the downcomer region. ~~Furthermore, heat dissipation by the aluminum heat conduction elements, if used, is conservatively neglected in the thermosiphon models employed in the HI-STORM modeling.~~

In previous revisions of this FSAR, the downcomer area was grossly understated in the FLUENT models. In Revision 2 of the FSAR, the downcomer area is still slightly understated for all MPC geometries (see table below), but the extent of conservatism has been moderated.

Comparison of the Actual and Assumed MPCs Downcomer Flow Areas			
	Actual (Based on drawings provided in Section 1.5)	*Assumed in the FLUENT Model (Revision 1)*	*Assumed in the FLUENT Model (Revision 2)*
MPC-24	*700.6*	*517.1*	*641.4*
MPC-24E & MPC-24EF	*664.9*	*517.1*	*641.4*
MPC-32 & MPC-32F	*773.3*	*517.1*	*746.1*
MPC-68, MPC-68F & MPC-68FF	*629.9*	*370.6*	*601.1*

Heat dissipation in the downcomer region is the sum of ~~five~~ *four* elements, viz. convective heat transfer (C1), helium conduction heat transfer (C2), basket-to-shell contact heat transfer (C3), *and* radiation heat transfer (C4). ~~and aluminum conduction elements (if used) heat transfer (C5).~~ In the HI-STORM thermal modeling, ~~two elements of heat transfer (C3 and C5) are~~ *C3 is* completely neglected, C2 is severely penalized and C1 is underpredicted. In other words the HI-STORM thermosiphon model has choked the radial flow of heat in the downcomer space. This has the direct effect of raising the temperature of fuel in the thermal solutions.

4.B.7 CONSERVATISM IN MPC EXTERNAL HEAT DISSIPATION TO CHIMNEY AIR

The principle means of decay heat dissipation to the environment is by cooling of the MPC surface by chimney air flow. Heat rejection from the MPC surface is by a combination of

convective heat transfer to a through flowing fluid medium (air), *and conduction through the overpack structure.* ~~natural convection cooling at the outer overpack surface, and by radiation heat transfer.~~ Because the temperature of the fuel stored in the MPC is directly affected by the rate of heat dissipation from the canister external surface, heat transfer correlations with robust conservatisms are employed in the HI-STORM simulations. The FLUENT computer code deployed for the modeling employs a so called "wall-functions" approach for computing the transfer of heat from solid surfaces to fluid medium. This approac has the desired effect of computing heat dissipation in a ~~most~~ conservative manner. ~~As this default approach has been employed in the thermal modeling, it is contextually relevant to quantify the conservatism in a classical setting to provide an additional level of assurance in the HI-STORM results. To do this, we have posed a classical heat transfer problem of a heated square block cooled in a stream of upward moving air. The problem is illustrated in Figure 4.B.8. From the physics of the problem, the maximum steady state solid interior temperature (T_{max}) is computed as:~~

~~$$T_{max} = T_{sink} + \Delta T_{air} + \Delta T_s$$~~

~~where, T_{sink} = Sink temperature (mean of inlet and outlet air temperature)~~
~~ΔT_{air} = Solid surface to air temperature difference~~
~~ΔT_s = Solid block interior temperature elevation~~

~~The sink temperature is computed by first calculating the air outlet temperature from energy conservation principles. Solid to air heat transfer is computed using classical natural convection correlation proposed by Jakob and Hawkins ("Elements of Heat Transfer", John Wiley & Sons, 1957) and ΔTs is readily computed by an analytical solution to the equation of heat conduction in solids. By solving this same problem on the FLUENT computer code using the in built "wall-functions", in excess of 100°F conservative margin over the classical result for T_{max} is established.~~

4.B.8 ~~MISCELLANEOUS~~ *OTHER* CONSERVATISMS

Section 4.4.6 of the FSAR lists ~~eleven elements~~ *an array* of conservatism*s*, of which certain non-transparent and individually significant items are discussed in detail in this appendix. *These conservatisms are primarily intrinsic to the modeling methodology or are product of assumptions in the input data. Examples in the latter category are values assumed in the thermal analysis for key inputs such as insolation heat, ambient temperature and emissivity of the heat exchange surfaces. Conservatisms in the former category includes implicit assumptions to under represent heat transfer, an example of this being the assumption that fuel pellets do not contribute to axial heat dissipation in the MPC. A listing of conservatisms is provided below:*

i) Axial heat transfer through fuel pellets is neglected.
ii) The upflow of helium through the MPCs is assumed to be laminar (high flow resistance, low heat transfer).
iii) Heat dissipation by grid spacers, top & bottom fittings is ignored.
iv) Insolation heating assumed with a bounding absorbtivity of 1.0.
v) Contact between fuel and basket and between basket and supports neglected.

vi) *MPC is assumed to be loaded with the most thermally resistive fuel type in its category (BWR or PWR) as applicable.*

Finally, it should be noted that the computed peak cladding temperatures for all MPCs are also lower than the 400°C limit by varying amounts, which can be viewed as an additional thermal margin in the system. The assumptions inherent to the FLUENT solution methodology and to the solution process, in conjunction with those in the input data, are estimated to have an aggregate effect of overestimating cladding temperatures by a considerable amount, as estimated in Table 4.B.1.

~~Out of the balance of conservatisms, the one of notable mention is the conservatism in fuel decay heat generation stipulation based on the most heat emissive fuel assembly type. This posture imputes a large conservatism for certain other fuel types, which have a much lower quantity of Uranium fuel inventory relative to the design basis fuel type. Combining this with other miscellaneous conservatisms, an aggregate effect is to overestimate cladding temperatures by about 15°F to 50°F.~~

~~4.B.9 CONCLUSIONS~~

~~The foregoing narrative provides a physical description of the many elements of conservatism in the HI-STORM 100 thermal model. The conservatisms may be broadly divided into two categories:~~

~~1. Those intrinsic to the FLUENT modeling process.~~

~~2. Those arising from the input data and on the HI-STORM 100 thermal modeling.~~

~~The conservatism in Category (1) may be identified by reviewing the Holtec International Benchmark Report [4.B.1], which shows that the FLUENT solution methodology, when applied to the prototype cask (TN 24P) over predicts the peak cladding temperature by as much as 79 °F. and as much as 37°F relative to the PNNL results (see Attachment 1 to Reference [4.B.1]) from their COBRA-SFS solution as compared against Holtec's FLUENT solution.~~

~~Category (2) conservatisms are those that we have deliberately embedded in the HI-STORM 100 thermal model to ensure that the computed value of the peak fuel cladding temperature is further over-stated. Table 4.B.1 contains a listing of the major conservatisms in the HI-STORM 100 thermal model, along with an estimate of the effect (increase) of each on the computed peak cladding temperature.~~

Table 4.B.1

Conservatism in the HI-STORM 100 Thermal Model

MODELING ELEMENT	*ESTIMATED* CONSERVATISM *IN THE COMPUTED MAX. CLADDING TEMPERATURE* [°F]
Long Term Ambient Temperature	2 to 30
Hypothetical Cylinder Construct	~5
~~Axial Heat Dissipation Restriction~~	~~30~~
MPC *Plenum Flow* ~~Downcomer Heat Dissipation~~ Restriction	~~50~~ *5*
~~MPC External Heat Dissipation Under prediction~~ *Use of SNF Bounding Axial Flow Resistance*	50
~~Miscellaneous~~ *Other* Conservatisms	~~15 to 50~~ *15*

~~4.B.9 REFERENCES~~

~~[4.B.1] "Topical Report on the HI-STAR/HI-STORM Thermal Model and its Benchmarking with Full-Size Cask Test Data", Holtec Report HI-992252, Rev. 1.~~

Figure 4.B.1

[Intentionally Deleted]

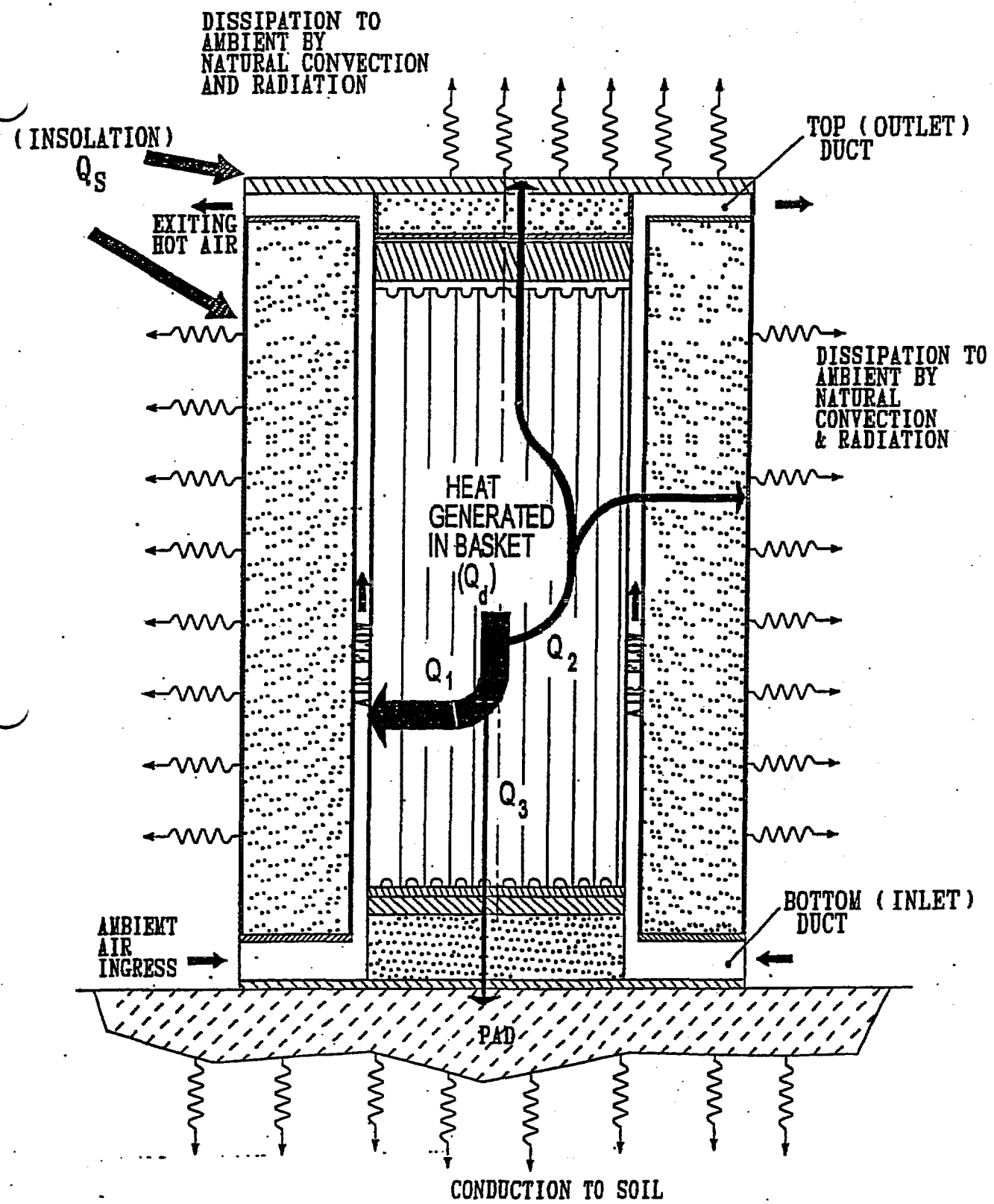


FIGURE 4.B.2: DEPICTION OF THE HI-STORM VENTILATED CASK HEAT DISSIPATION ELEMENTS

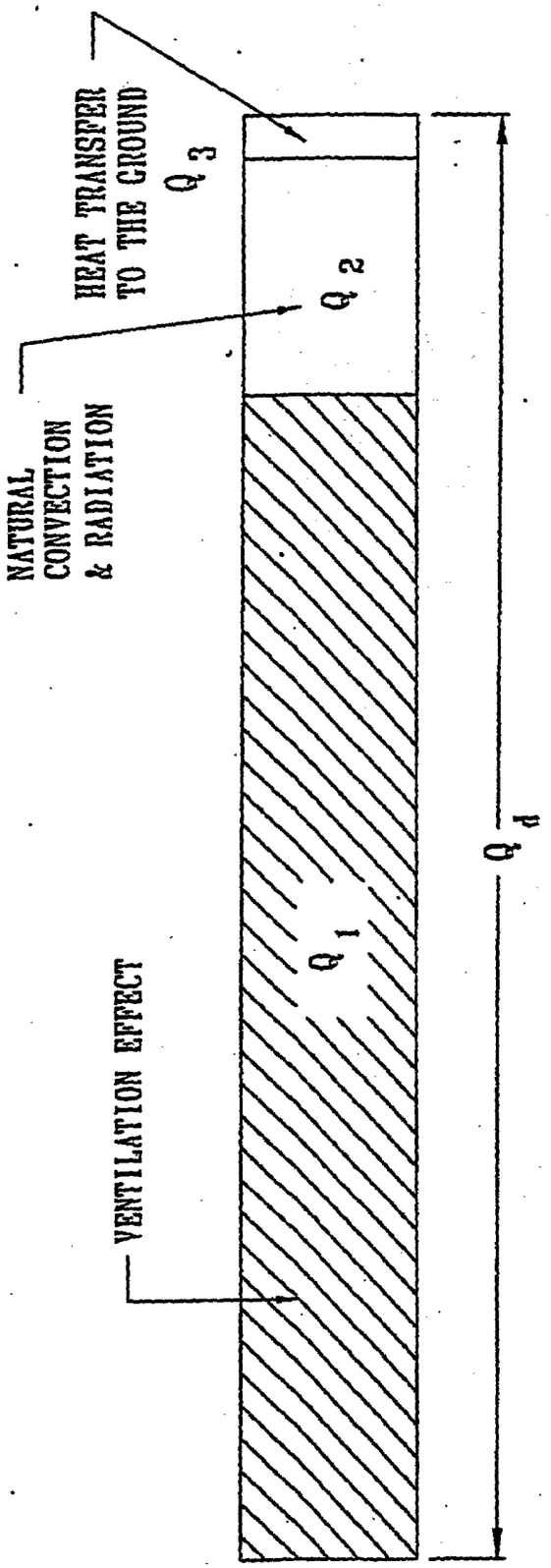


FIGURE 4.B.3: RELATIVE SIGNIFICANCE OF HEAT DISSIPATION ELEMENTS IN THE HI-STORM 100

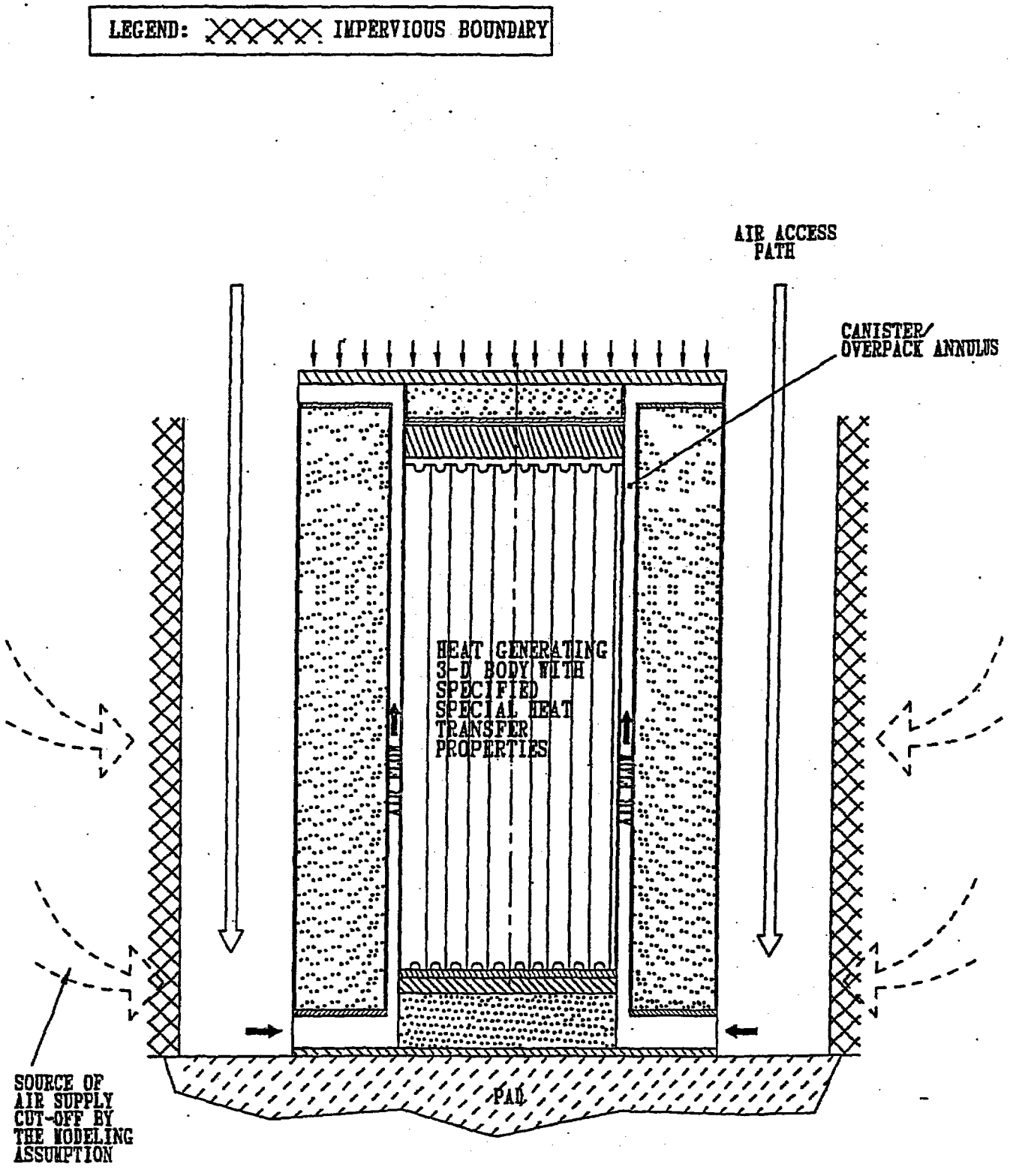


FIGURE 4.B.4: AIR ACCESS RESTRICTIONS IN THE HI-STORM THERMAL MODEL

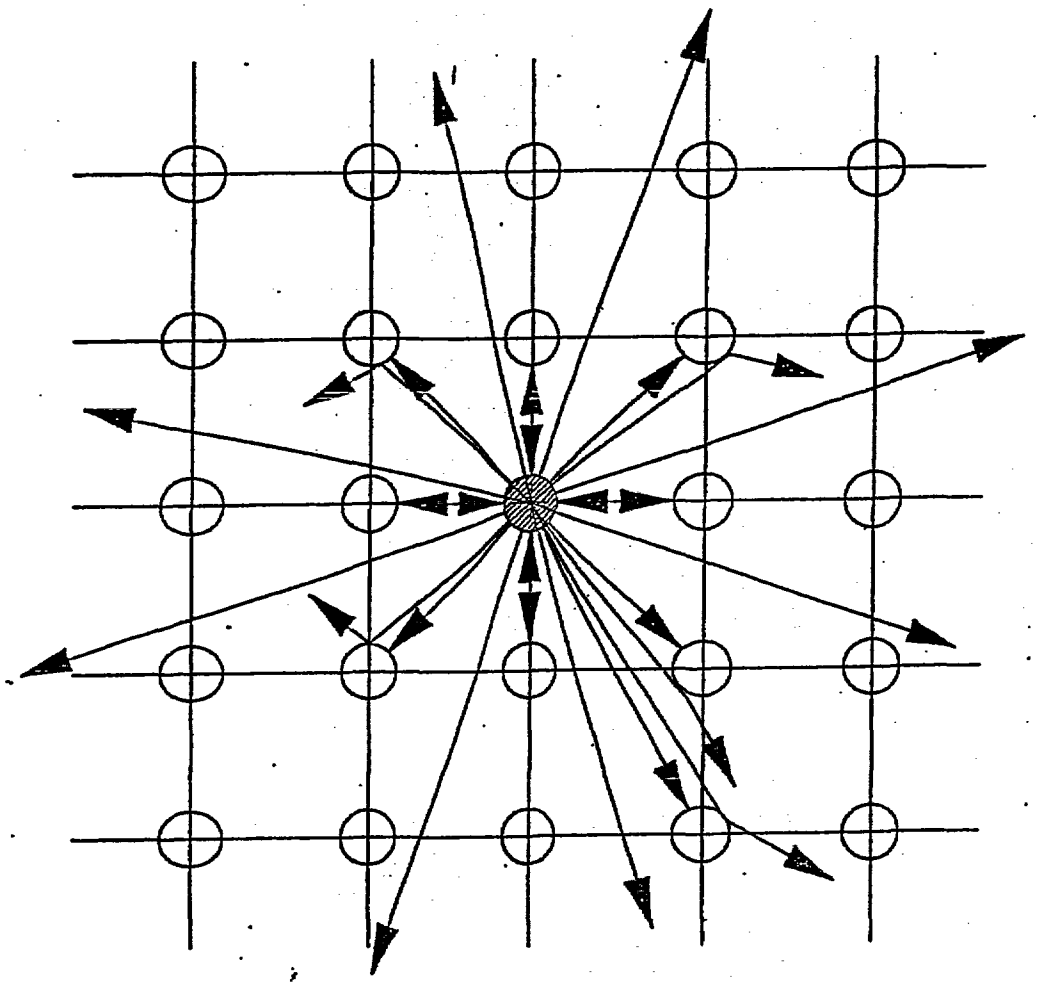


FIGURE 4.B.5: IN-PLANE RADIATIVE COOLING OF A HI-STORM CASK IN AN ARRAY

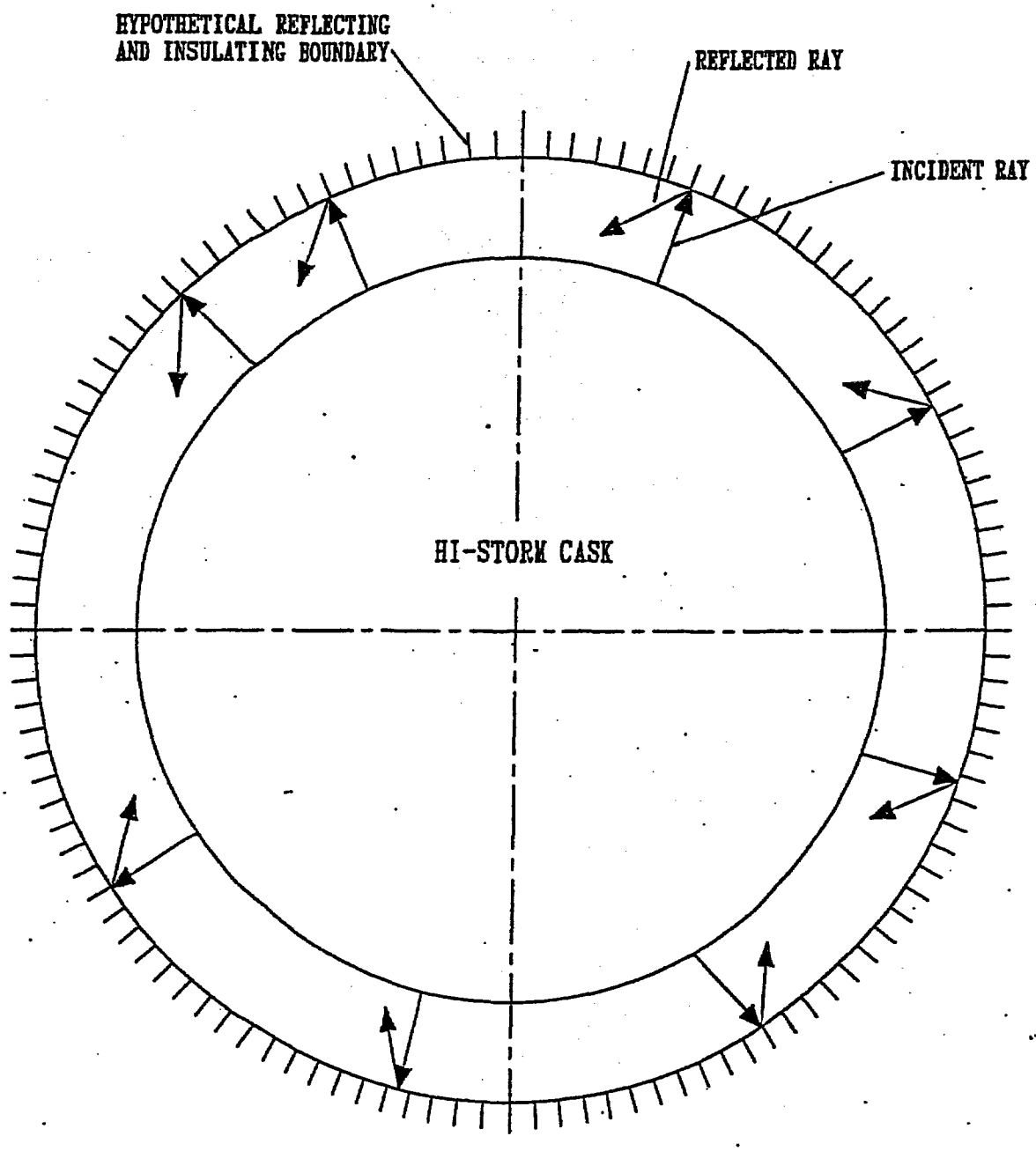


FIGURE 4.B.6: IN-PLANE RADIATIVE BLOCKING OF A HI-STORM CASK BY HYPOTHETICAL REFLECTING BOUNDARY

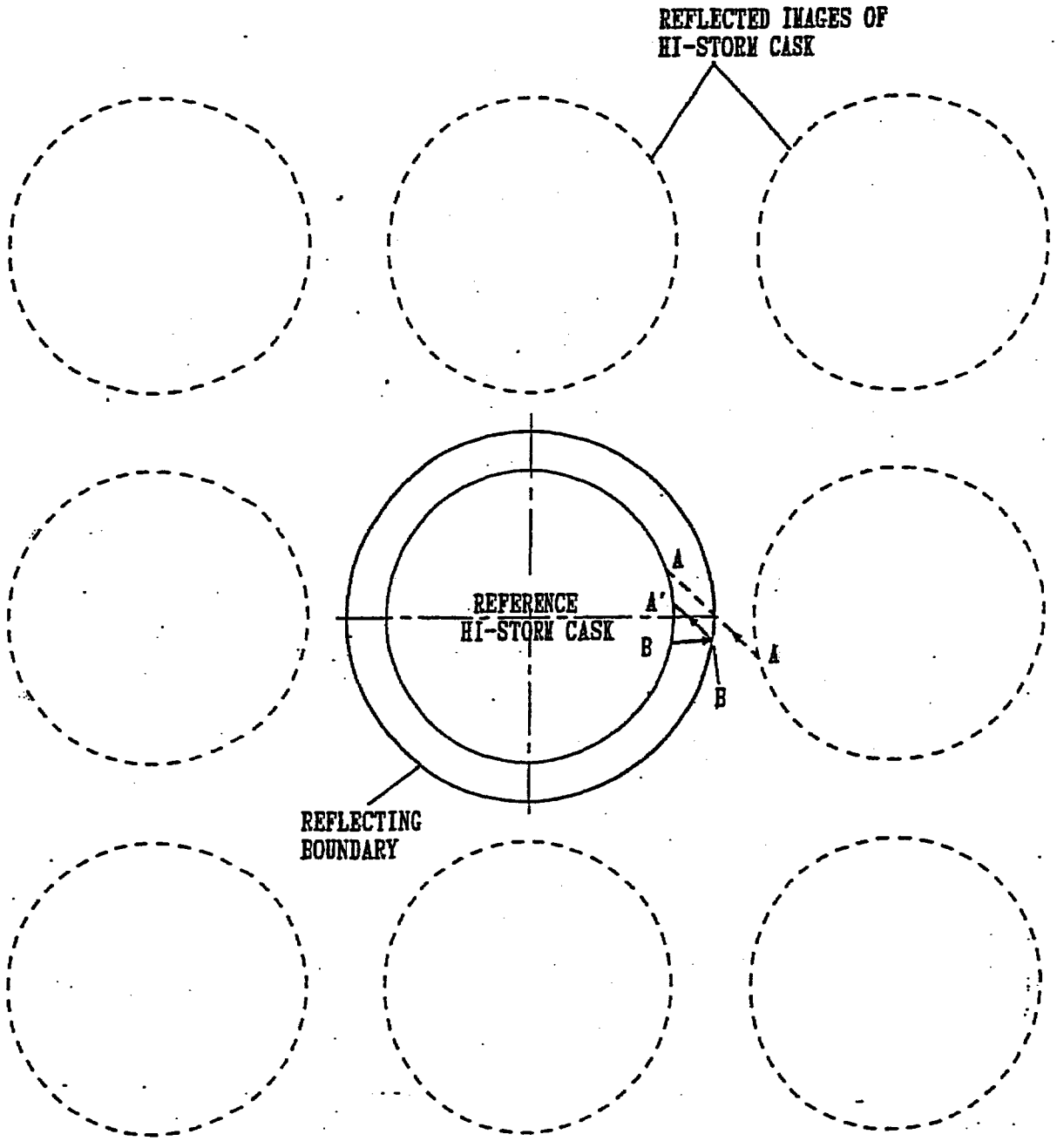


FIGURE 4.B.7: RADIATIVE HEATING OF REFERENCE HI-STORM CASK BY SURROUNDING CASKS

Figure 4.B.8

[Intentionally Deleted]

CHAPTER 5[†]: SHIELDING EVALUATION

5.0 INTRODUCTION

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

In addition to storing intact PWR and BWR fuel assemblies, the HI-STORM 100 System is designed to store BWR and PWR damaged fuel assemblies and fuel debris. Damaged fuel assemblies and fuel debris are defined in Sections 2.1.3 and ~~the approved contents section of Appendix B to the CoC~~*2.1.9*. Both damaged fuel assemblies and fuel debris are required to be loaded into Damaged Fuel Containers (DFCs) prior to being loaded into the MPC. DFCs containing BWR fuel debris must be stored in the MPC-68F or MPC-68FF. DFCs containing BWR damaged fuel assemblies may be stored in either the MPC-68, the MPC-68F, or the MPC-68FF. DFCs containing PWR fuel debris must be stored in the MPC-24EF *or MPC-32F* while DFCs containing PWR damaged fuel assemblies may be stored in either the MPC-24E, ~~or~~ MPC-24EF, *MPC-32, or MPC-32F.*

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

---

† This chapter has been prepared in the format and section organization set forth in Regulatory Guide 3.61. However, the material content of this chapter also fulfills the requirements of NUREG-1536. Pagination and numbering of sections, figures, and tables are consistent with the convention set down in *Chapter 1,* Section 1.0, herein. Finally, all terms-of-art used in this chapter are consistent with the terminology of the glossary (Table 1.0.1) and component nomenclature of the Bill-of-Materials (Section 1.5).

PWR fuel assemblies may contain burnable poison rod assemblies (BPRAs), thimble plug devices (TPDs), control rod assemblies (CRAs) or axial power shaping rod assemblies (APSRs) or similarly named devices. These non-fuel hardware devices are an integral yet removable part of PWR fuel assemblies and therefore the HI-STORM 100 System has been designed to store PWR fuel assemblies with or without these devices. Since each device occupies the same location within a fuel assembly, a single PWR fuel assembly will not contain multiple devices.

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

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

Acceptance Criteria

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

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

3. Dose rates from the cask must be consistent with a well established "as low as reasonably achievable" (ALARA) program for activities in and around the storage site.

4. After a design-basis accident, an individual at the boundary or outside the controlled area shall not receive a dose greater than the limits specified in 10CFR 72.106.

5. The proposed shielding features must ensure that the dry cask storage system meets the regulatory requirements for occupational and radiation dose limits for individual members of the public, as prescribed in 10 CFR Part 20, Subparts C and D.

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

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

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

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

Chapter 10, Radiation Protection, contains the following information:

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

5.1 DISCUSSION AND RESULTS

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

- Gamma radiation originating from the following sources

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

- Neutron radiation originating from the following sources

  1. Spontaneous fission
  2. α,n reactions in fuel materials
  3. Secondary neutrons produced by fission from subcritical multiplication
  4. γ,n reactions (this source is negligible)
  5. Dresden Unit 1 antimony-beryllium neutron sources

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

The shielding analyses were performed with MCNP-4A [5.1.1] developed by Los Alamos National Laboratory (LANL). The source terms for the design basis fuels were calculated with the SAS2H and ORIGEN-S sequences from the SCALE 4.3 system [5.1.2, 5.1.3]. A detailed description of the MCNP models and the source term calculations are presented in Sections 5.3 and 5.2, respectively.

The design basis zircaloy clad fuel assemblies used for calculating the dose rates presented in this chapter are B&W 15x15 and the GE 7x7, for PWR and BWR fuel types, respectively. The design basis intact 6x6 and mixed oxide (MOX) fuel assemblies are the GE 6x6. The GE 6x6 is also the design basis damaged fuel assembly for the Dresden Unit 1 and Humboldt Bay array classes. ~~Table 2.1.6~~*Section 2.1.9* specifies the acceptable intact zircaloy clad fuel characteristics ~~for storage. Table 2.1.7 specifies~~*and* the acceptable damaged fuel characteristics ~~for storage~~.

The design basis stainless steel clad fuels are the WE 15x15 and the A/C 10x10, for PWR and BWR fuel types, respectively. ~~Table 2.1.8~~*Section 2.1.9* specifies the acceptable fuel characteristics of stainless steel clad fuel for storage.

The MPC-24, MPC-24E, MPC-24EF, MPC-32, *MPC-32F,* MPC-68, and MPC-68FF are qualified for storage of SNF with different combinations of maximum burnup levels and minimum cooling times. ~~The approved contents section of Appendix B to the CoC~~*Section 2.1.9* specifies the acceptable maximum burnup levels and minimum cooling times for storage of zircaloy clad fuel in these MPCs. ~~Appendix B to the CoC~~*Section 2.1.9* also specifies the acceptable maximum burnup levels and minimum cooling times for storage of stainless steel clad fuel. The *burnup and cooling time* values in ~~Appendix B to the CoC~~*Section 2.1.9, which differ by array class,* were chosen based on an analysis of the maximum decay heat load that could be accommodated within each MPC. *Section 5.2 of this chapter describes the choice of the design basis fuel assembly based on a comparison of source terms and also provides a description of how the allowable burnup and cooling times were derived. Since for a given cooling time, different array classes have different allowable burnups in Section 2.1.9, burnup and cooling times that bound array classes 14x14A and 9x9G were used for the analysis in this chapter since these array class burnup and cooling time combinations bound the combinations from the other PWR and BWR array classes. Section 5.2.5 describes how this results in a conservative estimate of the maximum dose rates.*

Section 2.1.9 specifies that the maximum assembly average burnup for PWR and BWR fuel is 68,200 and 65,000 MWD/MTU, respectively. The analysis in this chapter conservatively considers burnups up to 75,000 and 70,000 MWD/MTU for PWR and BWR fuel, respectively.

The dose rates surrounding the HI-STORM overpack are very low, and thus, the shielding analysis of the HI-STORM overpack conservatively considered the burnup and cooling time combinations listed below, which bound the acceptable burnup levels and cooling times from ~~Appendix B to the CoC~~*Section 2.1.9*. This large conservatism is included in the analysis of the HI-STORM overpack to unequivocally demonstrate that the HI-STORM overpack meets the Part 72 dose requirements.

Zircaloy Clad Fuel		
MPC-24	**MPC-32**	**MPC-68**
~~52~~*47*,500 MWD/MTU ~~5~~*-3* year cooling	~~45~~*35*,000 MWD/MTU ~~5~~*-3* year cooling	~~47,500~~ *40,000* MWD/MTU ~~5~~*-3* year cooling
Stainless Steel Clad Fuel		
MPC-24	**MPC-32**	**MPC-68**
40,000 MWD/MTU 8 year cooling	40,000 MWD/MTU 9 year cooling	22,500 MWD/MTU 10 year cooling

The burnup and cooling time combinations analyzed for zircaloy clad fuel produce dose rates at the midplane of the HI-STORM overpack which bound all uniform and regionalized loading burnup and cooling time combinations listed in ~~Appendix B to the CoC~~*Section 2.1.9*. Therefore, the HI-STORM shielding analysis presented in this chapter is conservatively bounding for the MPC-24, MPC-32, and MPC-68.

The dose rates surrounding the HI-TRAC transfer cask are significantly higher than the dose rates surrounding the HI-STORM overpack, and although no specific regulatory limits are defined, dose rates are based on the ALARA principle. Therefore, the cited dose rates were based on ~~the actual~~ burnups and cooling times ~~requested in Appendix B to the CoC~~*closer to the combinations in Section 2.1.9*. Two different burnup and cooling times, listed below, were analyzed for the MPC-24, MPC-32, and the MPC-68 in the 100-ton HI-TRAC. *The burnup and cooling time combinations were chosen for the minimum cooling time and a bounding burnup corresponding to the 14x14A in the MPC-24 and MPC-32 and the 9x9G fuel assembly in the MPC-68.* The burnups corresponding to ~~5~~*3*-year cooling times produce dose rates at 1 meter from the *radial* surface of the overpack, for the locations reported in this chapter, which bound the dose rates from all other uniform loading burnup and cooling time combinations ~~listed~~ in ~~Appendix B to the CoC~~*Section 2.1.9*. ~~Since it is reasonable to assume that the majority of fuel which will be loaded in casks will be 10 years or older, the dose rates from conservative burnups for 10 year cooling are also presented in this chapter.~~

100-ton HI-TRAC		
MPC-24	**MPC-32**	**MPC-68**
~~42,500~~*46,000* MWD/MTU ~~5~~*3* year cooling	~~32,500~~*35,000* MWD/MTU ~~5~~*3* year cooling	~~40~~*39*,000 MWD/MTU ~~5~~*3* year cooling
~~52,500~~*75,000* MWD/MTU ~~10~~*5* year cooling	~~45,000~~*75,000* MWD/MTU ~~10~~*8* year cooling	~~50~~*70*,000 MWD/MTU ~~10~~*6* year cooling

The 100-ton HI-TRAC with the MPC-24 has higher dose rates at the mid-plane than the 100-ton HI-TRAC with the MPC-32 or the MPC-68. Therefore, the MPC-24 results for ~~5~~*3*-year cooling are presented in this section and the MPC-24 was used for the dose exposure estimates in Chapter 10. The MPC-32 results, MPC-68 results, and additional MPC-24 results are provided in Section 5.4 for comparison.

The 100-ton HI-TRAC dose rates bound the HI-TRAC 125 and 125D dose rates for the same burnup and cooling time combinations. Therefore, for illustrative purposes, the MPC-24 was the only MPC analyzed in the HI-TRAC 125 and 125D. Since the HI-TRAC 125D has fewer radial ribs, the dose rate at the midplane of the HI-TRAC 125D is higher than the dose rate at the midplane of the HI-TRAC 125. Therefore, the results on the radial surface are only presented for the HI-TRAC 125D in this chapter. Dose rates are presented for two different burnup and cooling time combinations for the MPC-24 in the HI-TRAC 125D *which bound the allowable contents Section 2.1.9*: ~~42,500~~*46,000* MWD/MTU with ~~5~~*3*-year cooling and ~~57,500~~*75,000*

MWD/MTU with ~~125~~-year cooling. The dose rates for the later combination are presented in this section because it produces the highest dose rate at the cask midplane. Dose rates for the other burnup and cooling time combination are presented in Section 5.4.

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

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

5.1.1 Normal and Off-Normal Operations

Chapter 11 discusses the potential off-normal conditions and their effect on the HI-STORM 100 System. None of the off-normal conditions have any impact on the shielding analysis. Therefore, off-normal and normal conditions are identical for the purpose of the shielding evaluation.

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

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

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

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

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

The HI-STORM overpack dose rates presented in this section are conservatively evaluated for the MPC-32, the MPC-68, and the MPC-24. All burnup and cooling time combinations analyzed bound the allowable burnup and cooling times specified in ~~Appendix B to the CoC~~*Section 2.1.9*.

Figure 5.1.1 and 5.1.12 identify the locations of the dose points referenced in the dose rate summary tables for the HI-STORM 100 and HI-STORM 100S overpacks, respectively. Dose

Points #1 and #3 are the locations of the inlet and outlet air ducts, respectively. The dose values reported for these locations (adjacent and 1 meter) were averaged over the duct opening. Dose Point #4 is the peak dose location above the overpack shield block. For the adjacent top dose, this dose point is located over the air annulus between the MPC and the overpack. Dose Point #4a in Figure 5.1.12 is located directly above the exit duct and next to the concrete shield block. The dose values reported at the locations shown on Figure 5.1.1 and 5.1.12 are averaged over a region that is approximately 1 foot in width.

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

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

Although the dose rates for the MPC-32 in HI-STORM 100s are ~~equivalent to or~~ greater than those for the MPC-24 in HI-STORM 100 *at the ventilation ducts*, as shown in Tables 5.1.1, 5.1.2, 5.1.4, and 5.1.5, the MPC-24 was used in the calculations for the dose rates at the controlled area boundary. *This is acceptable because the vents are a small fraction of the radial surface area. As such, the dominant effect on the dose at distance is the radial portion of the overpack between the vents which comprises approximately 91% of the total radial surface area compared to approximately 1.3% for the vents.* The MPC-24 was chosen because, for a given cooling time, the MPC-24 has a higher allowable burnup than the MPC-32 or the MPC-68 (see ~~Appendix B to the CoC~~*Section 2.1.9*). Consequently, for the allowable burnup and cooling times, the MPC-24 will have dose rates that are greater than or equivalent to those from the MPC-68 and MPC-32. The dose rates at the controlled area boundary were calculated for the HI-STORM 100 overpack rather than the HI-STORM 100S overpack. The difference in height will have little impact on the dose rates at the controlled area boundary since the surface dose rates are very similar. The controlled area boundary dose rates were also calculated ~~without~~ including *the BPRA* non-fuel hardware *source. In the site specific dose analysis, users should perform an analysis which properly bounds the fuel to be stored including BPRAs if present.* ~~This is acceptable because the dose rates for the HI-STORM 100 overpack calculated in Table 5.1.2 without BPRAs are conservative enough to bound the dose rates for actual burnup and cooling times from Appendix B to the CoC including BPRAs.~~

Table 5.1.7 provides dose rates adjacent to and one meter from the 100-ton HI-TRAC. Table 5.1.8 provides dose rates adjacent to and one meter from the 125-ton HI-TRACs. Figures 5.1.2 and 5.1.4 identify the locations of the dose points referenced in Tables 5.1.7 and 5.1.8 for the HI-TRAC 125 and 100 transfer casks, respectively. The dose rates listed in Tables 5.1.7 and 5.1.8

correspond to the normal condition in which the MPC is dry and the HI-TRAC water jacket is filled with water. The dose rates below the HI-TRAC (Dose Point #5) are provided for two conditions. The first condition is when the pool lid is in use and the second condition is when the transfer lid is in use. The HI-TRAC 125D does not utilize the transfer lid, rather it utilizes the pool lid in conjunction with the mating device. Therefore the dose rates reported for the pool lid are applicable to both the HI-TRAC 125 and 125D while the dose rates reported for the transfer lid are applicable only to the HI-TRAC 125. The calculational model of the 100-ton HI-TRAC included a concrete floor positioned 6 inches (the typical carry height) below the pool lid to account for ground scatter. As a result of the modeling, the dose rate at 1 meter from the pool lid for the 100-ton HI-TRAC was not calculated. The dose rates provided in Tables 5.1.7 and 5.1.8 are for the MPC-24 with design basis fuel at burnups and cooling times, based on the allowed burnup and cooling times specified in ~~Appendix B to the CoC~~*Section 2.1.9*, that result in dose rates that are generally higher in each of the two HI-TRAC designs. The burnup and cooling time combination used for both the 100-ton and 125-ton HI-TRAC was chosen ~~based onto~~ *bound* the allowable burnup and cooling times in ~~Appendix B to the CoC~~*Section 2.1.9*. Results for other burnup and cooling times and for the MPC-68 and MPC-32 are provided in Section 5.4.

Because the dose rates for the 100-ton HI-TRAC transfer cask are significantly higher than the dose rates for the 125-ton HI-TRACs or the HI-STORM overpack, it is important to understand the behavior of the dose rates surrounding the external surface. To assist in this understanding, several figures, showing the dose rate profiles on the top, bottom and sides of the 100-ton HI-TRAC transfer cask, are presented below. The figures discussed below were all calculated without the gamma source from BPRAs and were calculated for an earlier design of the HI-TRAC which utilized 30 steel fins 0.375 inches thick compared to 10 steel fins 1.25 inches thick. The change in rib design only affects the magnitude of the dose rates presented for the radial surface but does not affect the conclusions discussed below.

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

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

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

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

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

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

The dose to any real individual at or beyond the controlled area boundary is required to be below 25 mrem per year. The minimum distance to the controlled area boundary is 100 meters from the ISFSI. As mentioned, only the MPC-24 was used in the calculation of the dose rates at the controlled area boundary. Table 5.1.9 presents the annual dose to an individual from a single HI-STORM cask and various storage cask arrays, assuming an 8760 hour annual occupancy at the dose point location. The minimum distance required for the corresponding dose is also listed. These values were conservatively calculated for a burnup of ~~52~~*47*,500 MWD/MTU and a ~~53~~-year cooling time. In addition, the annual dose was calculated for ~~a~~ burnup*s* of 45,000 *and 52,500* MWD/MTU *with corresponding cooling times of*~~and a~~ *9 and 5* years *respectively.* ~~cooling time.~~ BPRAs were ~~not~~ included in these dose estimates. It is noted that these data are provided for illustrative purposes only. A detailed site-specific evaluation of dose at the controlled area boundary must be performed for each ISFSI in accordance with 10CFR72.212, as

stated in Chapter 12, "Operating Controls and Limits". The site-specific evaluation will consider dose from other portions of the facility and will consider the actual conditions of the fuel being stored (burnup and cooling time).

Figure 5.1.3 is an annual dose versus distance graph for the cask array configurations provided in Table 5.1.9. This curve, which is based on an 8760 hour occupancy, is provided for illustrative purposes only and will be re-evaluated on a site-specific basis.

Section 5.2 lists the gamma and neutron sources for the design basis fuels. Since the source strengths of the GE 6x6 intact and damaged fuel and the GE 6x6 MOX fuel are significantly smaller in all energy groups than the intact design basis fuel source strengths, the dose rates from the GE 6x6 fuels for normal conditions are bounded by the MPC-68 analysis with the design basis intact fuel. Therefore, no explicit analysis of the MPC-68 with either GE 6x6 intact or damaged or GE 6x6 MOX fuel for normal conditions is required to demonstrate that the MPC-68 with GE 6x6 fuels will meet the normal condition regulatory requirements. Section 5.4.2 evaluates the effect of generic damaged fuel in the MPC-24E, *MPC-32* and the MPC-68.

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

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

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

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

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

5.1.2 Accident Conditions

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

> Any individual located on or beyond the nearest boundary of the controlled area may not receive from any design basis accident the more limiting of a total effective dose equivalent of 5 Rem, or the sum of the deep-dose equivalent and the committed dose equivalent to any individual organ or tissue (other than the lens of the eye) of 50 Rem.

The lens dose equivalent shall not exceed 15 Rem and the shallow dose equivalent to skin or to any extremity shall not exceed 50 rem. The minimum distance from the spent fuel or high-level radioactive waste handling and storage facilities to the nearest boundary of the controlled are shall be at least 100 meters.

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

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

Throughout all design basis accident conditions the axial location of the fuel will remain fixed within the MPC because of the fuel spacers. The HI-STAR 100 System (Docket Number 72-1008) documentation provides analysis to demonstrate that the fuel spacers will not fail under any normal, off-normal, or accident condition of storage. Chapter 3 also shows that the HI-TRAC inner shell, lead, and outer shell remain intact throughout all design basis accident conditions. Localized damage of the HI-TRAC outer shell could be experienced. However, the localized deformations will have only a negligible impact on the dose rate at the boundary of the controlled area.

The complete loss of the HI-TRAC neutron shield significantly affects the dose at mid-height (Dose Point #2) adjacent to the HI-TRAC. Loss of the neutron shield has a small effect on the dose at the other dose points. To illustrate the impact of the design basis accident, the dose rates at Dose Point #2 (see Figures 5.1.2 and 5.1.4) are provided in Table 5.1.10. The normal condition dose rates are provided for reference. Table 5.1.10 provides a comparison of the normal and accident condition dose rates at one meter from the HI-TRAC. The burnup and cooling time combinations used in Table 5.1.10 were the combinations that resulted in the highest post-accident condition dose rates. These burnup and cooling time combinations do not necessarily correspond to the burnup and cooling time combinations that result in the highest dose rate during normal conditions. Scaling this accident dose rate by the dose rate reduction seen in HI-STORM yields a dose rate at the 100 meter controlled area boundary that would be approximately ~~1.48~~*4.06*† mrem/hr for the HI-TRAC accident condition. At this dose rate, it would take ~~3378~~*1231* hours (~~~14~~*51* days) for the dose at the controlled area boundary to reach 5 Rem. *Assuming a 30 day accident duration, the accumulated dose at the controlled area boundary would be 2.92 Rem.* Based on this dose rate and the short duration of use for the

† ~~2098.54~~*5927.95* mrem/hr (Table 5.1.10) x [~~129~~*289* mrem/yr (Table 5.4.7) / 8760 hrs / ~~20.9~~*48.16* mrem/hr (Table 5.1.5)]

loaded HI-TRAC transfer cask, it is evident that the dose as a result of the design basis accident cannot exceed 5 Rem at the controlled area boundary for the short duration of the accident.

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

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

Table 5.1.1

DOSE RATES ADJACENT TO HI-STORM 100S OVERPACK
FOR NORMAL CONDITIONS
MPC-32 DESIGN BASIS ZIRCALOY CLAD FUEL AT BOUNDING
BURNUP AND COOLING TIME
~~45~~35,000 MWD/MTU AND ~~5~~3-YEAR COOLING

Dose Point† Location	Fuel Gammas†† (mrem/hr)	^{60}Co Gammas (mrem/hr)	Neutrons (mrem/hr)	Totals (mrem/hr)	Totals with BPRAs (mrem/hr)
1	*15.16*	*18.14*	*3.44*	*36.75*	*37.68*
2	*84.79*†††	*0.05*	*1.02*	*85.86*	*92.07*
3	*15.88*	*18.95*	*2.71*	*37.54*	*45.75*
4	*3.22*	*1.18*	*0.95*	*5.36*	*6.10*
4a	*7.12*	*10.46*	*13.26*	*30.83*	*35.71*

† Refer to Figure 5.1.12.

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

††† The cobalt activation of incore grid spacers accounts for ~~*8.5*~~*4.1* % of this dose rate.

Table 5.1.2

DOSE RATES ADJACENT TO HI-STORM 100 OVERPACK
FOR NORMAL CONDITIONS
MPC-24 DESIGN BASIS ZIRCALOY CLAD FUEL AT BOUNDING
BURNUP AND COOLING TIME
~~52~~47,500 MWD/MTU AND ~~5~~3-YEAR COOLING

Dose Point† Location	Fuel Gammas†† (mrem/hr)	^{60}Co Gammas (mrem/hr)	Neutrons (mrem/hr)	Totals (mrem/hr)	Totals with BPRAs (mrem/hr)
1	*11.14*	*6.61*	*3.70*	*21.46*	*21.84*
2	*88.86*†††	*0.04*	*2.52*	*91.41*	*96.85*
3	*7.51*	*4.36*	*1.84*	*13.71*	*15.38*
4	*1.74*	*0.49*	*4.82*	*7.05*	*7.51*

† Refer to Figure 5.1.1.

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

††† The cobalt activation of incore grid spacers accounts for ~~8.04~~ % of this dose rate.

Table 5.1.3

DOSE RATES ADJACENT TO HI-STORM 100S OVERPACK FOR NORMAL CONDITIONS
MPC-68 DESIGN BASIS ZIRCALOY CLAD FUEL AT BOUNDING BURNUP AND COOLING TIME
~~47,500~~ *40,000* MWD/MTU AND ~~5~~*3*-YEAR COOLING

Dose Point† Location	Fuel Gammas†† (mrem/hr)	^{60}Co Gammas (mrem/hr)	Neutrons (mrem/hr)	Totals (mrem/hr)
1	*15.26*	*14.43*	*5.79*	*35.48*
2	*77.57*	*0.01*	*1.76*	*79.35*
3	*6.40*	*18.63*	*2.58*	*27.62*
4	*1.81*	*1.42*	*0.94*	*4.17*
4a	*1.77*	*11.45*	*12.55*	*25.77*

† Refer to Figure 5.1.12.

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

Table 5.1.4

DOSE RATES AT ONE METER FROM HI-STORM 100S OVERPACK
FOR NORMAL CONDITIONS
MPC-32 DESIGN BASIS ZIRCALOY CLAD FUEL AT BOUNDING
BURNUP AND COOLING TIME
~~45~~35,000 MWD/MTU AND ~~5~~3-YEAR COOLING

Dose Point† Location	Fuel Gammas†† (mrem/hr)	^{60}Co Gammas (mrem/hr)	Neutrons (mrem/hr)	Totals (mrem/hr)	Totals with BPRAs (mrem/hr)
1	*10.50*	*6.08*	*0.50*	*17.07*	*17.89*
2	*44.21*†††	*0.39*	*0.43*	*45.02*	*48.25*
3	*8.31*	*5.33*	*0.44*	*14.08*	*16.77*
4	*0.83*	*0.37*	*0.42*	*1.62*	*1.83*

† Refer to Figure 5.1.12.

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

††† The cobalt activation of incore grid spacers accounts for ~~8.~~*64.1* % of this dose rate.

Table 5.1.5

DOSE RATES AT ONE METER FROM HI-STORM 100 OVERPACK
FOR NORMAL CONDITIONS
MPC-24 DESIGN BASIS ZIRCALOY CLAD FUEL AT BOUNDING
BURNUP AND COOLING TIME
~~52~~47,500 MWD/MTU AND ~~5~~3-YEAR COOLING

Dose Point† Location	Fuel Gammas†† (mrem/hr)	^{60}Co Gammas (mrem/hr)	Neutrons (mrem/hr)	Totals (mrem/hr)	Totals with BPRAs (mrem/hr)
1	*11.15*	*3.94*	*0.72*	*15.82*	*16.36*
2	*46.78*†††	*0.33*	*1.04*	*48.16*	*50.95*
3	*6.51*	*2.84*	*0.28*	*9.64*	*10.87*
4	*0.84*	*0.22*	*1.47*	*2.53*	*2.66*

† Refer to Figure 5.1.1.

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

††† The cobalt activation of incore grid spacers accounts for ~~8.04~~ % of this dose rate.

Table 5.1.6

DOSE RATES AT ONE METER FROM HI-STORM 100S OVERPACK
FOR NORMAL CONDITIONS
MPC-68 DESIGN BASIS ZIRCALOY CLAD FUEL AT BOUNDING
BURNUP AND COOLING TIME
~~47,500~~ *40,000* MWD/MTU AND ~~5~~*3*-YEAR COOLING

Dose Point† Location	Fuel Gammas†† (mrem/hr)	^{60}Co Gammas (mrem/hr)	Neutrons (mrem/hr)	Totals (mrem/hr)
1	*10.65*	*4.55*	*0.77*	*15.96*
2	*39.27*	*0.33*	*0.74*	*40.34*
3	*4.02*	*5.70*	*0.42*	*10.14*
4	*0.45*	*0.44*	*0.40*	*1.28*

† Refer to Figure 5.1.12.

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

Table 5.1.7

DOSE RATES FROM THE 100-TON HI-TRAC FOR NORMAL CONDITIONS
MPC-24 DESIGN BASIS ZIRCALOY CLAD FUEL
~~42,500~~*46,000* MWD/MTU AND ~~5~~*3*-YEAR COOLING

Dose Point Location	Fuel Gammas (mrem/hr)	(n,γ) Gammas (mrem/hr)	^{60}Co Gammas (mrem/hr)	Neutrons (mrem/hr)	Totals (mrem/hr)	Totals with BPRAs (mrem/hr)
ADJACENT TO THE 100-TON HI-TRAC						
1	*106.76*	*17.29*	*849.14*	*244.86*	*1218.05*	*1226.59*
2	*2673.26*†	*70.39*	*0.85*	*129.91*	*2874.41*	*3121.64*
3	*31.55*	*3.39*	*468.20*	*204.87*	*708.01*	*856.53*
3 (temp)	*14.08*	*6.03*	*217.01*	*3.29*	*240.41*	*308.56*
4	*67.59*	*1.34*	*376.81*	*252.20*	*697.94*	*822.44*
4 (outer)	*20.45*	*0.85*	*93.82*	*170.24*	*285.36*	*316.69*
5 (pool lid)	*704.26*	*22.94*	*4298.12*	*1518.06*	*6543.38*	*6608.15*
5 (transfer)	*1015.91*	*1.35*	*6375.30*	*941.78*	*8334.34*	*8431.18*
5(t-outer)	*262.72*	*0.46*	*617.08*	*372.07*	*1252.32*	*1273.80*
ONE METER FROM THE 100-TON HI-TRAC						
1	*354.02*	*9.30*	*126.22*	*39.80*	*529.34*	*561.82*
2	*1170.82*†	*21.52*	*9.99*	*48.71*	*1251.03*	*1360.51*
3	*148.77*	*5.18*	*104.85*	*19.11*	*277.92*	*327.35*
3 (temp)	*147.95*	*5.56*	*89.31*	*7.23*	*250.05*	*294.61*
4	*23.46*	*0.23*	*116.33*	*62.83*	*202.86*	*241.43*
5 (transfer)	*453.62*	*0.25*	*2604.33*	*262.81*	*3321.01*	*3360.14*
5(t-outer)	*62.33*	*0.80*	*234.75*	*75.45*	*373.34*	*377.23*

Notes:

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

† The cobalt activation of incore grid spacers accounts for ~~12.3~~*6.3%* of the surface and one-meter dose rates.

Table 5.1.8

DOSE RATES FROM THE 125-TON HI-TRACS FOR NORMAL CONDITIONS
MPC-24 DESIGN BASIS ZIRCALOY CLAD FUEL
~~57,500~~*75,000* MWD/MTU AND ~~125~~-YEAR COOLING

Dose Point Location	Fuel Gammas (mrem/hr)	(n,γ) Gammas (mrem/hr)	^{60}Co Gammas (mrem/hr)	Neutrons (mrem/hr)	Totals (mrem/hr)	Totals with BPRAs (mrem/hr)
ADJACENT TO THE 125-TON HI-TRACs						
1	*6.32*	*61.85*	*100.63*	*415.90*	*584.70*	*585.42*
2	*113.33*†	*183.20*	*0.01*	*287.94*	*584.49*	*600.36*
3	*1.41*	*6.55*	*62.26*	*663.65*	*733.88*	*753.59*
4	*41.57*	*8.40*	*340.67*	*767.94*	*1158.58*	*1274.01*
4 (outer)	*4.84*	*6.00*	*42.31*	*16.11*	*69.26*	*83.45*
5 (pool)	*54.77*	*3.67*	*454.56*	*2883.53*	*3396.53*	*3404.24*
5 (transfer)	*65.81*	*4.78*	*601.40*	*440.29*	*1112.28*	*1117.76*
ONE METER FROM THE 125-TON HI-TRACs						
1	*14.93*	*24.68*	*12.90*	*68.44*	*120.95*	*122.99*
2	*50.47*†	*59.39*	*0.52*	*98.23*	*208.61*	*215.68*
3	*5.66*	*13.95*	*12.58*	*61.07*	*93.26*	*98.17*
4	*11.54*	*2.03*	*82.02*	*79.09*	*174.68*	*202.33*
5 (transfer)	*25.98*	*0.92*	*290.76*	*76.26*	*393.92*	*396.85*

Notes:

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

† The cobalt activation of incore grid spacers accounts for ~~15.5~~*9.4%* of the surface and one-meter dose rates.

Table 5.1.9

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

Array Configuration	1 cask	2x2	2x3	2x4	2x5
47,500 MWD/MTU AND 3-YEAR COOLING					
Annual Dose (mrem/year)†	*24.10*	*18.07*	*15.86*	*21.15*	*16.29*
Distance to Controlled Area Boundary (meters)††,†††	*250*	*350*	*400*	*400*	*450*
52,500 MWD/MTU AND 5-YEAR COOLING					
Annual Dose (mrem/year)†	*22.88*	*14.34*	*21.52*	*16.79*	*20.99*
Distance to Controlled Area Boundary (meters) ††	200	*300*	300	350	350
45,000 MWD/MTU AND 9-YEAR COOLING					
Annual Dose (mrem/year) †	*22.20*	*23.41*	*16.77*	*22.36*	*14.91*
Distance to Controlled Area Boundary (meters) ††	150	200	250	250	*300*

† 8760 hr. annual occupancy is assumed.

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

††† Actual controlled area boundary dose rates will be lower because the maximum permissible burnup for ~~5~~3-year cooling, as specified in the ~~Appendix B to the CoC~~*Section 2.1.9*, is lower than the burnup used for this analysis.

Table 5.1.10

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

Dose Point† Location	Fuel Gammas†† (mrem/hr)	^{60}Co Gammas (mrem/hr)	Neutrons (mrem/hr)	Totals (mrem/hr)	Totals with BPRAs (mrem/hr)
125-TON HI-TRACs					
~~57,500~~*75,000* MWD/MTU AND ~~125~~-YEAR COOLING					
2 (Accident Condition)	*92.26*	*1.02*	*3476.98*	*3570.26*	*3583.16*
2 (Normal Condition)	*109.86*	*0.52*	*98.23*	*208.61*	*215.68*
100-TON HI-TRAC					
~~57,500~~*75,000* MWD/MTU AND ~~125~~-YEAR COOLING					
2 (Accident Condition)	*1354.67*	*17.88*	*4359.16*	*5731.72*	*5927.95*
2 (Normal Condition)	*829.09*	*9.90*	*168.82*	*1007.81*	*1117.29*

† Refer to Figures 5.1.2 and 5.1.4.

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

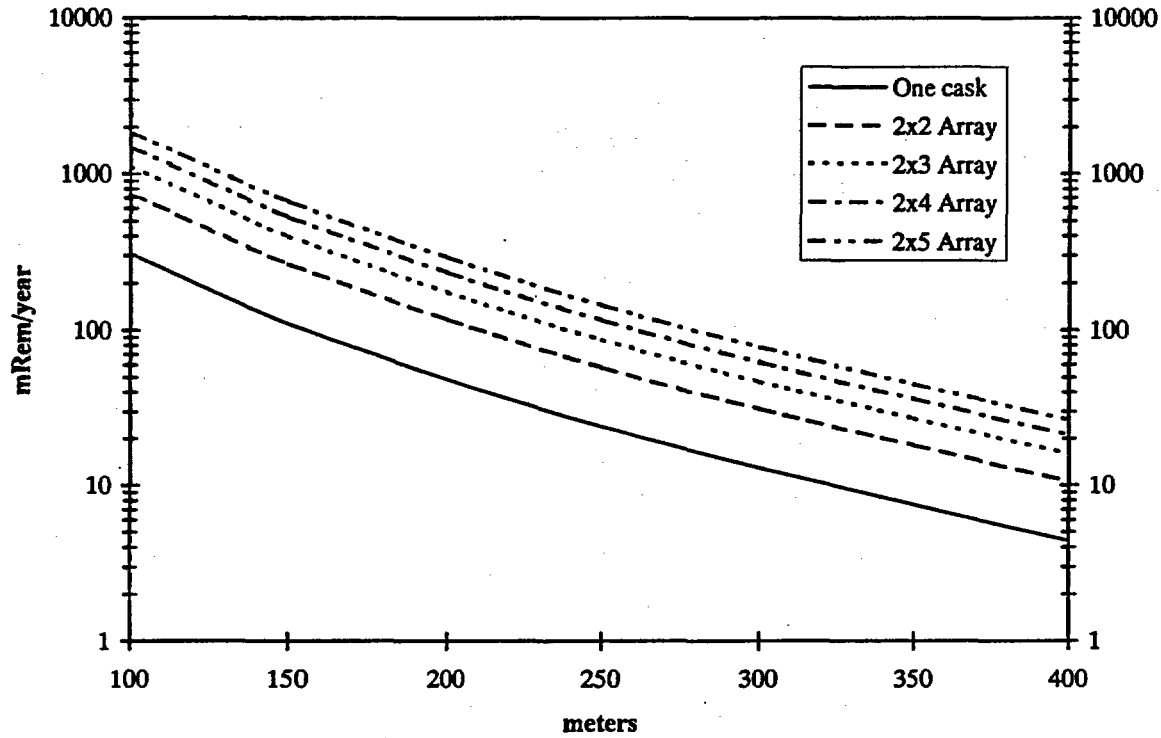


FIGURE 5.1.3; ANNUAL DOSE VERSUS DISTANCE FOR VARIOUS CONFIGURATIONS OF THE MPC-24 FOR 47,500 MWD/MTU AND 3-YEAR COOLING (8760 HOUR OCCUPANCY ASSUMED)

5.2 SOURCE SPECIFICATION

The neutron and gamma source terms, decay heat values, and quantities of radionuclides available for release were calculated with the SAS2H and ORIGEN-S modules of the SCALE 4.3 system [5.1.2, 5.1.3]. SAS2H has been extensively compared to experimental isotopic validations and decay heat measurements. References [5.2.8] through [5.2.12] *and [5.2.15]* present isotopic comparisons for PWR and BWR fuels for burnups ranging to 47 GWD/MTU and reference [5.2.13] presents results for BWR measurements to a burnup of 57 GWD/MTU. A comparison of calculated and measured decays heats is presented in reference [5.2.14]. All of these studies indicate good agreement between SAS2H and measured data. Additional comparisons of calculated values and measured data are being performed by various institutions for high burnup PWR and BWR fuel. These new results, when published, are expected to further confirm the validity of SAS2H for the analysis of PWR and BWR fuel.

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

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

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

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

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

In performing the SAS2H and ORIGEN-S calculations, a single full power cycle was used to achieve the desired burnup. This assumption, in conjunction with the above-average specific powers listed in Tables 5.2.1, 5.2.2, 5.2.3, and 5.2.21 resulted in conservative source term calculations.

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

5.2.1 Gamma Source

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

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

The primary source of activity in the non-fuel regions of an assembly arises from the activation of ^{59}Co to ^{60}Co. The primary source of ^{59}Co in a fuel assembly is impurities in the steel structural material above and below the fuel. The zircaloy in these regions is neglected since it does not have a significant ^{59}Co impurity level. Reference [5.2.2] indicates that the impurity level in steel is 800 ppm or 0.8 gm/kg. Conservatively, the impurity level of ^{59}Co was assumed to be 1000 ppm or 1.0 gm/kg. Therefore, Inconel and stainless steel in the non-fuel regions are both conservatively assumed to have the same 1.0 gm/kg impurity level.

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

Some of the PWR fuel assembly designs (B&W and WE 15x15) utilized inconel in-core grid spacers while other PWR fuel designs use zircaloy in-core grid spacers. In the mid 1980s, the fuel assembly designs using inconel in-core grid spacers were altered to use zircaloy in-core grid spacers. Since both designs may be loaded into the HI-STORM 100 system, the gamma source for the PWR zircaloy clad fuel assembly includes the activation of the in-core grid spacers. Although BWR assembly grid spacers are zircaloy, some assembly designs have inconel springs in conjunction with the grid spacers. The gamma source for the BWR zircaloy clad fuel assembly includes the activation of these springs associated with the grid spacers.

The non-fuel data listed in Table 5.2.1 were taken from References [5.2.2], [5.2.4], and [5.2.5]. As stated above, a Cobalt-59 impurity level of 1 gm/kg (0.1 wt%) was used for both inconel and stainless steel. Therefore, there is little distinction between stainless steel and inconel in the source term generation and since the shielding characteristics are similar, stainless steel was used in the MCNP calculations instead of inconel. The BWR masses are for an 8x8 fuel assembly. These masses are also appropriate for the 7x7 assembly since the masses of the non-fuel hardware from a 7x7 and an 8x8 are approximately the same. The masses listed are those of the steel components. The zircaloy in these regions was not included because zircaloy does not produce significant activation. The masses are larger than most other fuel assemblies from other manufacturers. This, in combination with the conservative ^{59}Co impurity level and the use of conservative flux weighting fractions (discussed below) results in an over-prediction of the non-fuel hardware source that bounds all fuel for which storage is requested.

The masses in Table 5.2.1 were used to calculate a ^{59}Co impurity level in the fuel assembly 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 5.2.10. These scaling factors were taken from Reference [5.2.3].

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

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

There is some uncertainty associated with the ORIGEN-S calculations due to uncertainty in the physics data (e.g. cross sections, decay constants, etc.) and the modeling techniques. References [5.2.9], [5.2.10], and [5.2.15] perform comparisons between calculations and experimental isotopic measurement data. These comparisons indicate that calculated to measured ratios for Cs-134 and Eu-154, two of the major contributors to the gamma source, range from 0.79 to 1.009 and 0.79 to 0.98, respectively. These values provide representative insight into the entire range of possible error in the source term calculations. However, any non-conservatism associated with the uncertainty in the source term calculations is offset by the conservative nature of the source term and shielding calculations performed in this chapter, and therefore no adjustments were made to the calculated values.

5.2.2 Neutron Source

It is well known that the neutron source strength increases as enrichment decreases, for a constant burnup and decay time. This is due to the increase in Pu content in the fuel, which increases the inventory of other transuranium nuclides such as Cm. The gamma source also varies with enrichment, although only slightly. Because of this effect and in order to obtain conservative source terms, low initial fuel enrichments were chosen for the BWR and PWR design basis fuel assemblies. The enrichments are appropriately varied as a function of burnup. Table 5.2.24 presents the ^{235}U initial enrichments for various burnup ranges from 20,000 - ~~70~~75,000 MWD/MTU for PWR and *20,000 - 70,000 MWD/MTU for* BWR zircaloy clad fuel. These enrichments are based on References [5.2.6] and [5.2.7]. Table 8 of reference [5.2.6] presents average enrichments for burnup ranges. The initial enrichments chosen in Table 5.2.24, for burnups up to 50,000 MWD/MTU, are approximately the average enrichments from Table 8 of reference [5.2.6] for the burnup range that is 5,000 MWD/MTU less than the ranges listed in

Table 5.2.24. These enrichments are below the enrichments typically required to achieve the burnups that were analyzed. For burnups greater than 50,000 MWD/MTU, the data on historical and projected burnups available in the LWR Quantities Database in reference [5.2.7] *and some additional data from nuclear plants* was reviewed and conservatively low enrichments were chosen for each burnup range above 50,000 MWD/MTU.

Inherent to this approach of selecting minimum enrichments that bound the vast majority of discharged fuel is the fact that a small number of atypical assemblies will not be bounded. However, these atypical assemblies are very few in number (as evidenced by the referenced discharge data), and thus, it is unlikely that a single cask would contain several of these outlying assemblies. Further, because the approach is based on using minimum enrichments for given burnup ranges, any atypical assemblies that may exist are expected to have enrichments that are very near to the minimum enrichments used in the analysis. Therefore, the result is an insignificant effect on the calculated dose rates. Consequently, the minimum enrichment values used in the *shielding* analysis are adequate to bound the fuel authorized by the limits in ~~the CoC~~*Section 2.1.9* for loading in the HI-STORM system. ~~Therefore a minimum enrichment is not specified in the limits in the CoC.~~ Since the enrichment does affect the source term evaluation, it is recommended that the site-specific dose evaluation consider the enrichment for the fuel being stored.

The neutron source calculated for the design basis fuel assemblies for the MPC-24, MPC-32, and MPC-68 and the design basis 6x6 fuel are listed in Tables 5.2.15 through 5.2.18 in neutrons/s for varying burnup and cooling times. Table 5.2.23 provides the neutron source in neutrons/sec for the design basis MOX fuel assembly. ^{244}Cm accounts for approximately ~~96~~*92-97%* of the total number of neutrons produced~~, with slightly over 2% originating from (α,n)~~ *Alpha,n* reactions *in isotopes other than ^{244}Cm account for approximately 0.3-2% of the neutrons produced while spontaneous fission in isotopes other than ^{244}Cm account for approximately 2-8% of the neutrons produced* 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.

There is some uncertainty associated with the ORIGEN-S calculations due to uncertainty in the physics data (e.g. cross sections, decay constants, etc.) and the modeling techniques. References [5.2.9], [5.2.10], and [5.2.15] perform comparisons between calculations and experimental isotopic measurement data. These comparisons indicate that calculated to measured ratios for Cm-244 ranges from 0.81 to 0.95. These values provide representative insight into the entire range of possible error in the source term calculations. However, any non-conservatism associated with the uncertainty in the source term calculations is offset by the conservative nature of the source term and shielding calculations performed in this chapter, and therefore no adjustments were made to the calculated values.

5.2.3 Stainless Steel Clad Fuel Source

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

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

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

5.2.4 Non-fuel Hardware

Burnable poison rod assemblies (BPRAs), thimble plug devices (TPDs), control rod assemblies (CRAs), and axial power shaping rods (APSRs) are permitted for storage in the HI-STORM 100

System as an integral part of a PWR fuel assembly. BPRAs and TPDs may be stored in any fuel location while CRAs and APSRs are restricted to the inner four fuel storage locations in the MPC-24, MPC-24E, and the MPC-32.

5.2.4.1 BPRAs and TPDs

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

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

BPRAs are made of stainless steel in the region above the active fuel zone and may contain a small amount of inconel in this region. Within the active fuel zone the BPRAs may contain 2-24 rodlets which are burnable absorbers clad in either zircaloy or stainless steel. The stainless steel clad BPRAs create a significant radiation source (Co-60) while the zircaloy clad BPRAs create a negligible radiation source. Therefore the stainless steel clad BPRAs are bounding.

SAS2H and ORIGEN-S were used to calculate a radiation source term for the TPDs and BPRAs. In the ORIGEN-S calculations the cobalt-59 impurity level was conservatively assumed to be 0.8 gm/kg for stainless steel and 4.7 gm/kg for inconel. These calculations were performed by irradiating the appropriate mass of steel and inconel using the flux calculated for the design basis B&W 15x15 fuel assembly. The mass of material in the regions above the active fuel zone was scaled by the appropriate scaling factors listed in Table 5.2.10 in order to account for the reduced flux levels above the fuel assembly. The total curies of cobalt were calculated for the TPDs and BPRAs as a function of burnup and cooling time. For burnups beyond 45,000 MWD/MTU, it was assumed, for the purpose of the calculation, that the burned fuel assembly was replaced with a fresh fuel assembly every 45,000 MWD/MTU. This was achieved in ORIGEN-S by resetting the flux levels and cross sections to the 0 MWD/MTU condition after every 45,000 MWD/MTU.

Since the HI-STORM 100 cask system is designed to store many varieties of PWR fuel, a bounding TPD and BPRA had to be determined for the purposes of the analysis. This was

accomplished by analyzing all of the BPRAs and TPDs (Westinghouse and B&W 14x14 through 17x17) found in references [5.2.5] and [5.2.7] to determine the TPD and BPRA which produced the highest Cobalt-60 source term and decay heat for a specific burnup and cooling time. The bounding TPD was determined to be the Westinghouse 17x17 guide tube plug and the bounding BPRA was actually determined by combining the higher masses of the Westinghouse 17x17 and 15x15 BPRAs into a singly hypothetical BPRA. The masses of this TPD and BPRA are listed in Table 5.2.30. As mentioned above, reference [5.2.5] describes the Westinghouse 14x14 water displacement guide tube plug as having a steel portion which extends into the active fuel zone. This particular water displacement guide tube plug was analyzed and determined to be bounded by the design basis TPD and BPRA.

Once the bounding BPRA and TPD were determined, the allowable Co-60 source *and decay heat* from the BPRA and TPD were specified *as*: 50 curies Co-60 *and 0.77 watts* for each TPD and ~~831~~ *895* curies Co-60 *and 14.4 watts* for each BPRA. Table 5.2.31 shows the curies of Co-60 that were calculated for BPRAs and TPDs in each region of the fuel assembly (e.g. incore, plenum, top). An allowable burnup and cooling time, separate from the fuel assemblies, is used for BPRAs and TPDs. These burnup and cooling times assure that the Cobalt-60 activity remains below the allowable levels specified above. It should be noted that at very high burnups, greater than 200,000 MWD/MTU the TPD Co-60 source actually decreases as the burnup continues to increase. This is due to a decrease in the Cobalt-60 production rate as the initial Cobalt-59 impurity is being depleted. Conservatively, a constant cooling time has been specified for burnups from 180,000 to 630,000 MWD/MTU for the TPDs.

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

5.2.4.2 CRAs and APSRs

Control rod assemblies (CRAs) (including control element assemblies and rod cluster control assemblies) and axial power shaping rod assemblies (APSRs) are an integral portion of a PWR fuel assembly. These devices are utilized for many years (upwards of 20 years) prior to discharge into the spent fuel pool. The manner in which the CRAs are utilized vary from plant to plant. Some utilities maintain the CRAs fully withdrawn during normal operation while others may operate with a bank of rods partially inserted (approximately 10%) during normal operation. Even when fully withdrawn, the ends of the CRAs are present in the upper portion of the fuel assembly since they are never fully removed from the fuel assembly during operation. The result of the different operating styles is a variation in the source term for the CRAs. In all cases, however, only the lower portion of the CRAs will be significantly activated. Therefore, when the CRAs are stored with the PWR fuel assembly, the activated portion of the CRAs will be in the lower portion of the cask. CRAs are fabricated of various materials. The cladding is typically stainless steel, although inconel has been used. The absorber can be a single material or a combination of materials. AgInCd is possibly the most common absorber although B_4C in aluminum is used, and hafnium has also been used. AgInCd produces a noticeable source term in

the 0.3-1.0 MeV range due to the activation of Ag. The source term from the other absorbers is negligible, therefore the AgInCd CRAs are the bounding CRAs.

APSRs are used to flatten the power distribution during normal operation and as a result these devices achieve a considerably higher activation than CRAs. There are two types of B&W stainless steel clad APSRs: gray and black. According to reference [5.2.5], the black APSRs have 36 inches of AgInCd as the absorber while the gray ones use 63 inches of inconel as the absorber. Because of the cobalt-60 source from the activation of inconel, the gray APSRs produce a higher source term than the black APSRs and therefore are the bounding APSR.

Since the level of activation of CRAs and APSRs can vary, the quantity that can be stored in an MPC is being limited to four CRAs and/or APSRs. These four devices are required to be stored in the inner four locations in the MPC-24, MPC-24E, MPC-24EF, and MPC-32 as outlined in ~~Appendix B to the CoC~~*Section 2.1.9*.

In order to determine the impact on the dose rates around the HI-STORM 100 System, source terms for the CRAs and APSRs were calculated using SAS2H and ORIGEN-S. In the ORIGEN-S calculations the cobalt-59 impurity level was conservatively assumed to be 0.8 gm/kg for stainless steel and 4.7 gm/kg for inconel. These calculations were performed by irradiating 1 kg of steel, inconel, and AgInCd using the flux calculated for the design basis B&W 15x15 fuel assembly. The total curies of cobalt for the steel and inconel and the 0.3-1.0 MeV source for the AgInCd were calculated as a function of burnup and cooling time to a maximum burnup of 630,000 MWD/MTU. For burnups beyond 45,000 MWD/MTU, it was assumed, for the purpose of the calculation, that the burned fuel assembly was replaced with a fresh fuel assembly every 45,000 MWD/MTU. This was achieved in ORIGEN-S by resetting the flux levels and cross sections to the 0 MWD/MTU condition after every 45,000 MWD/MTU. The sources were then scaled by the appropriate mass using the flux weighting factors for the different regions of the assembly to determine the final source term. Two different configurations were analyzed for both the CRAs and APSRs with an additional third configuration analyzed for the APSRs. The configurations, which are summarized below, are described in Tables 5.2.32 for the CRAs and Table 5.2.33 for the APSR. The masses of the materials listed in these tables were determined from a review of [5.2.5] with bounding values chosen. The masses listed in Tables 5.2.32 and 5.2.33 do not match exact values from [5.2.5] because the values in the reference were adjusted to the lengths shown in the tables.

<u>Configuration 1: CRA and APSR</u>

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

Configuration 2: CRA and APSR
This configuration represents a fully removed CRA or APSR during normal core operations. The activated portion corresponds to the upper portion of a fuel assembly above the active fuel length with the appropriate flux weighting factors used.

Configuration 3: APSR
This configuration represents a fully inserted gray APSR during normal core operations. The region in full flux was assumed to be the 63 inches of the absorber.

Tables 5.2.34 and 5.2.35 present the source terms, *including decay heat,* that were calculated for the CRAs and APSRs respectively. The only significant source from the activation of inconel or steel is Co-60 and the only significant source from the activation of AgInCd is from 0.3-1.0 MeV. The source terms for CRAs, Table 5.2.34, were calculated for a maximum burnup of 630,000 MWD/MTU and a minimum cooling time of 5 years. Because of the significant source term in APSRs that have seen extensive in-core operations, the source term in Table 5.2.35 was calculated to be a bounding source term for a variable burnup and cooling time as outlined in ~~Appendix B to the CoC~~*Section 2.1.9*. The very larger Cobalt-60 activity in configuration 3 in Table 5.2.35 is due to the assumed Cobalt-59 impurity level of 4.7 gm/kg. If this impurity level were similar to the assumed value for steel, 0.8 gm/kg, this source would decrease by approximately a factor of 5.8.

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

5.2.5 Choice of Design Basis Assembly

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

5.2.5.1 PWR Design Basis Assembly

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

Table 5.2.25 presents the characteristics of the fuel assemblies analyzed to determine the design basis zircaloy clad PWR fuel assembly. *The corresponding fuel assembly array class from Section 2.1.9 is also listed in the table.* The fuel assembly listed for each class is the assembly with the highest UO_2 mass. The St. Lucie and Ft. Calhoun classes are not present in Table 5.2.25. These assemblies are shorter versions of the CE 16x16 and CE 14x14 assembly classes, respectively. Therefore, these assemblies are bounded by the CE 16x16 and CE 14x14 classes and were not explicitly analyzed. Since the Indian Point 1, Haddam Neck, and San Onofre 1 classes are stainless steel clad fuel, these classes were analyzed separately and are discussed below. All fuel assemblies in Table 5.2.25 were analyzed at the same burnup and cooling time. The initial enrichment used in the analysis is consistent with Table 5.2.24. The results of the comparison are provided in Table 5.2.27. These results indicate that the B&W 15x15 fuel assembly has the highest radiation source term of the zircaloy clad fuel assembly classes considered in Table 2.1.1. This fuel assembly also has the highest UO_2 mass (see Table 5.2.25) which confirms that, for a given initial enrichment, burnup, and cooling time, the assembly with the highest UO_2 mass produces the highest radiation source term. *The power/assembly values used in Table 5.2.25 were calculated by dividing 110% of the thermal power for commercial PWR reactors using that array class by the number of assemblies in the core. The higher thermal power, 110%, was used to account for potential power uprates. The power level used for the B&W15 is an additional 17% higher for consistency with previous revisions of the FSAR which also used this assembly as the design basis assembly.*

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

As discussed below in Section 5.2.5.3, the allowable burnup limits in Section 2.1.9 were calculated for different array classes rather than using the design basis assembly to calculate the allowable burnups for all array classes. As mentioned above, the design basis assembly has the highest neutron and gamma source term of the various array classes for the same burnup and cooling time. In order to account for the fact that different array classes have different allowable burnups for the same cooling time, burnups which bound the 14x14A array class were used with the design basis assembly for the analysis in this chapter because those burnups bound the burnups from all other PWR array classes. This approach assures that the calculated source terms and dose rates will be conservative.

5.2.5.2 BWR Design Basis Assembly

Table 2.1.2 lists the BWR fuel assembly classes that were evaluated to determine the design basis BWR fuel assembly. Since there are minor differences between the array types in the GE

BWR/2-3 and GE BWR/4-6 assembly classes, these assembly classes were not considered individually but rather as a single class. Within that class, the array types, 7x7, 8x8, 9x9, and 10x10 were analyzed to determine the bounding BWR fuel assembly. Since the Humboldt Bay 7x7 and Dresden 1 8x8 are smaller versions of the 7x7 and 8x8 assemblies they are bounded by the 7x7 and 8x8 assemblies in the GE BWR/2-3 and GE BWR/4-6 classes. Within each array type, the fuel assembly with the highest UO_2 mass was analyzed. Since the variations of fuel assemblies within an array type are very minor, it is conservative to choose the assembly with the highest UO_2 mass. For a given array type of assemblies, the one with the highest UO_2 mass will produce the highest radiation source because, for a given burnup (MWD/MTU) and enrichment, it will have produced the most energy and therefore the most fission products. The Humboldt Bay 6x6, Dresden 1 6x6, and LaCrosse assembly classes were not considered in the determination of the bounding fuel assembly. However, these assemblies were analyzed explicitly as discussed below.

Table 5.2.26 presents the characteristics of the fuel assemblies analyzed to determine the design basis zircaloy clad BWR fuel assembly. *The corresponding fuel assembly array class from Section 2.1.9 is also listed in the table.* The fuel assembly listed for each array type is the assembly that has the highest UO_2 mass. All fuel assemblies in Table 5.2.26 were analyzed at the same burnup and cooling time. The initial enrichment used in these analyses is consistent with Table 5.2.24. The results of the comparison are provided in Table 5.2.28. These results indicate that the 7x7 fuel assembly has the highest radiation source term of the zircaloy clad fuel assembly classes considered in Table 2.1.2. This fuel assembly also has the highest UO_2 mass which confirms that, for a given initial enrichment, burnup, and cooling time, the assembly with the highest UO_2 mass produces the highest radiation source term. According to Reference [5.2.6], the last discharge of a 7x7 assembly was in 1985 and the maximum average burnup for a 7x7 during their operation was 29,000 MWD/MTU. This clearly indicates that the existing 7x7 assemblies have an average burnup and minimum cooling time that is well within the burnup and cooling time limits in ~~Appendix B to the CoC~~*Section 2.1.9*. Therefore, the 7x7 assembly has never reached the burnup level analyzed in this chapter. However, in the interest of conservatism the 7x7 was chosen as the bounding fuel assembly array type. *The power/assembly values used in Table 5.2.26 were calculated by dividing 120% of the thermal power for commercial BWR reactors by the number of assemblies in the core. The higher thermal power, 120%, was used to account for potential power uprates. The power level used for the 7x7 is an additional 4% higher for consistency with previous revisions of the FSAR which also used this assembly as the design basis assembly.*

Since the LaCrosse fuel assembly type is a stainless steel clad 10x10 assembly it was analyzed separately. The maximum burnup and minimum cooling time for this assembly are limited to 22,500 MWD/MTU and 10-year cooling as specified in ~~Appendix B to the CoC~~*Section 2.1.9*. This assembly type is discussed further in Section 5.2.3.

The Humboldt Bay 6x6 and Dresden 1 6x6 fuel are older and shorter fuel than the other array types analyzed and therefore are considered separately. The Dresden 1 6x6 was chosen as the design basis fuel assembly for the Humboldt Bay 6x6 and Dresden 1 6x6 fuel assembly classes

because it has the higher UO_2 mass. Dresden 1 also contains a few 6x6 MOX fuel assemblies, which were explicitly analyzed as well.

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

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

As discussed below in Section 5.2.5.3, the allowable burnup limits in Section 2.1.9 were calculated for different array classes rather than using the design basis assembly to calculate the allowable burnups for all array classes. As mentioned above, the design basis assembly has the highest neutron and gamma source term of the various array classes for the same burnup and cooling time. In order to account for the fact that different array classes have different allowable burnups for the same cooling time, burnups which bound the 9x9G array class were used with the design basis assembly for the analysis in this chapter because those burnups bound the burnups from all other BWR array classes. This approach assures that the calculated source terms and dose rates will be conservative.

5.2.5.3 Decay Heat Loads *and Allowable Burnup and Cooling Times*

Section 2.1.6 describes the calculation of the *MPC maximum decay heat limits per assembly. These limits, which differ for uniform and regionalized loading, are presented in Section 2.1.9.* ~~burnup versus cooling time limits in the CoC that are based on a maximum permissible decay heat per assembly.~~ *The allowable burnup and cooling time limits are derived based on the allowable decay heat limits. Since the decay heat of an assembly will vary slightly with enrichment for a fixed burnup and cooling time, an equation is used to represent burnup as a function of decay heat and enrichment. This equation is of the form:*

$$B_u = A*q + B*q^2 + C*q^3 + D*E_{235}^{\ 2} + E*E_{235}*q + F*E_{235}*q^2 + G$$

where:
B_u = *Burnup in MWD/MTU*
q = *assembly decay heat (kW)*
E_{235} = *wt.%* ^{235}U

The coefficients for this equation were developed by fitting ORIGEN-S calculated data for a specific cooling time. ORIGEN-S calculations were performed for enrichments ranging from 0.7 to 5.0 wt.% ^{235}U *and burnups from 10,000 to 65,000 MWD/MTU for BWRs and 10,000 to 70,000 MWD/MTU for PWRs. The burnups were increased in 2,500 MWD/MTU increments. Using the ORIGEN-S data, the coefficients A through G were determined and then the constant, G, was*

adjusted so that all data points were bounded (i.e. calculated burnup less than or equal to ORIGEN-S value) by the fit. The coefficients were calculated using ORIGEN-S data for cooling times from 3 years to 20 years. As a result, Section 2.1.9 provides different equation coefficients for each cooling time from 3 to 20 years. Since the decay heat increases as the enrichment decreases, the allowable burnup will decrease as the enrichment decreases. Therefore, the enrichment used to calculated the allowable burnups becomes a minimum enrichment value and assemblies with an enrichment higher than the value used in the equation are acceptable for storage assuming they also meet the corresponding burnup and decay heat requirements.

~~The decay heat values per assembly were calculated using the methodology described in Section 5.2.~~ *Different array classes or combinations of classes were analyzed separately to determine the allowable burnup as a function of cooling time for the specified allowable decay heat limits. Calculating allowable burnups for individual array classes is appropriate because even two assemblies with the same MTU may have a different allowable burnup for the same allowable cooling time and permissible decay heat. The heavy metal mass specified in Table 5.2.25 and 5.2.26 and Section 2.1.9 for the various array classes is the value that was used in the determination of the coefficients as a function of cooling time and is the maximum for the respective assembly class. Equation coefficients for each array class listed in Tables 5.2.25 and 5.2.26 were developed. In the end, the equation for the 17x17B and 17x17C array classes resulted in almost identical burnups. Therefore, in Section 2.1.9 these array classes were combined and the coefficients for the 17x17C array class were used since these coefficients produce slightly lower allowable burnups.*

There is some uncertainty associated with the ORIGEN-S calculations due to uncertainty in the physics data (e.g. cross sections, decay constants, etc.) and the modeling techniques. To estimate this uncertainty, an approach similar to the one in Reference [5.2.14] was used. As a result, the potential error in the ORIGEN-S decay heat calculations was estimated to be in the range of 3.5 to 5.5% at 3 year cooling time and 1.5 to 3.5% at 20 year cooling. The difference is due to the change in isotopes important to decay heat as a function of cooling time. In order to be conservative in the derivation of the coefficients for the burnup equation, a 5% decay heat penalty was applied for the BWR array classes. A penalty was not applied to the PWR array classes since the thermal analysis in Chapter 4 has more than a 5% margin in the calculated allowable decay heat.
~~The design basis fuel assemblies, as described in Table 5.2.1, were used in the calculation of the burnup versus cooling time limits in the CoC. The enrichments used in the calculation of the decay heats were consistent with Table 5.2.24. As demonstrated in Tables 5.2.27 and 5.2.28, the design basis fuel assembly produces a higher decay heat value than the other assembly types considered. This is due to the higher heavy metal mass in the design basis fuel assemblies. Conservatively, Appendix B to the CoC limits the heavy metal mass to a value less than the design basis value utilized in this chapter. This provides additional assurance that the decay heat values are bounding values.~~

As ~~further~~ *a* demonstration that the decay heat values *used to determine the allowable burnups* ~~(calculated using the design basis fuel assemblies)~~ are conservative, a comparison between these

calculated decay heats and the decay heats reported in Reference [5.2.7] are presented in Table 5.2.29. This comparison is made for a burnup of 30,000 MWD/MTU and a cooling time of 5 years. The burnup was chosen based on the limited burnup data available in Reference [5.2.7].

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

As mentioned above, the fuel assembly burnup and cooling times in ~~Appendix B to the CoC~~*Section 2.1.9* were calculated using the decay heat limits which are also stipulated in ~~Appendix B to the CoC~~*Section 2.1.9*. The burnup and cooling times for the non-fuel hardware, in ~~Appendix B to the CoC~~*Section 2.1.9*, were chosen based on the radiation source term calculations discussed previously. The fuel assembly burnup, *decay heat, and enrichment equations* ~~and cooling times~~ were ~~calculated~~ *derived* without consideration for the decay heat from BPRAs, TPDs, CRAs, or APSRs. This is acceptable since the user of the HI-STORM 100 system is required to demonstrate compliance with the assembly decay heat limits in ~~Appendix B to the CoC~~*Section 2.1.9* regardless of the heat source (assembly or non-fuel hardware) and the actual decay heat from the non-fuel hardware is expected to be minimal. In addition, the shielding analysis presented in this chapter conservatively calculates the dose rates using both the burnup and cooling times for the fuel assemblies and non-fuel hardware. Therefore, the safety of the HI-STORM 100 system is guaranteed through the bounding analysis in this chapter, represented by the burnup and cooling time limits in the CoC, and the bounding thermal analysis in Chapter 4, represented by the decay heat limits in the CoC.

5.2.6 Thoria Rod Canister

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

A radiation source term was calculated for the 18 thoria rods using SAS2H and ORIGEN-S for a burnup of 16,000 MWD/MTU and a cooling time of 18 years. Table 5.2.36 describes the 8x8 fuel assembly that contains the thoria rods. Table 5.2.37 and 5.2.38 show the gamma and neutron source terms, respectively, that were calculated for the 18 thoria rods in the thoria rod canister.

Comparing these source terms to the design basis 6x6 source terms for Dresden Unit 1 fuel in Tables 5.2.7 and 5.2.18 clearly indicates that the design basis source terms bound the thoria rods source terms in all neutron groups and in all gamma groups except the 2.5-3.0 MeV group. As mentioned above, the thoria rods have a significant source in this energy range due to the decay of Tl-208.

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

5.2.7 Fuel Assembly Neutron Sources

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

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

5.2.8 Stainless Steel Channels

The LaCrosse nuclear plant used two types of channels for their BWR assemblies: stainless steel and zircaloy. Since the irradiation of zircaloy does not produce significant activation, there are no restrictions on the storage of these channels and they are not explicitly analyzed in this chapter. The stainless steel channels, however, can produce a significant amount of activation, predominantly from Co-60. LaCrosse has thirty-two stainless steel channels, a few of which, have been in the reactor core for, approximately, the lifetime of the plant. Therefore, the activation of the stainless steel channels was conservatively calculated to demonstrate that they are acceptable for storage in the HI-STORM 100 system. For conservatism, the number of stainless steel channels in an MPC-68 is being limited to sixteen and ~~Appendix B to the CoC~~*Section 2.1.9* requires that these channels be stored in the inner sixteen locations.

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

LaCrosse was commercially operated from November 1969 until it was shutdown in April 1987. Therefore, the shortest cooling time for the assemblies and the channels is 13 years. Assuming

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

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

Table 5.2.1

DESCRIPTION OF DESIGN BASIS ZIRCALOY CLAD FUEL

	PWR	BWR
Assembly type/class	B&W 15×15	GE 7×7
Active fuel length (in.)	144	144
No. of fuel rods	208	49
Rod pitch (in.)	0.568	0.738
Cladding material	Zircaloy-4	Zircaloy-2
Rod diameter (in.)	0.428	0.570
Cladding thickness (in.)	0.0230	0.0355
Pellet diameter (in.)	0.3742	0.488
Pellet material	UO_2	UO_2
Pellet density (gm/cc)	10.412 (95% of theoretical)	10.412 (95% of theoretical)
Enrichment (w/o ^{235}U)	3.6	3.2
~~Burnup (MWD/MTU)~~‡	~~52,500 (MPC-24)~~ ~~45,000 (MPC-32)~~	~~47,500 (MPC-68)~~
~~Cooling Time (years)~~‡	~~5 (MPC-24 and 32)~~	~~5 (MPC-68)~~
Specific power (MW/MTU)	40	30
Weight of UO_2 (kg)††	562.029	225.177
Weight of U (kg)††	495.485	198.516

Notes:

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

---

† Burnup and cooling time combinations conservatively bound the acceptable burnup and cooling times ~~listed~~ in ~~Appendix B to the CoC~~*Section 2.1.9*.

†† Derived from parameters in this table.

Table 5.2.1 (continued)

DESCRIPTION OF DESIGN BASIS FUEL

	PWR	BWR
No. of Water Rods	17	0
Water Rod O.D. (in.)	0.53	N/A
Water Rod Thickness (in.)	0.016	N/A
Lower End Fitting (kg)	8.16 (steel) 1.3 (inconel)	4.8 (steel)
Gas Plenum Springs (kg)	0.48428 (inconel) 0.23748 (steel)	1.1 (steel)
Gas Plenum Spacer (kg)	0.82824	N/A
Expansion Springs (kg)	N/A	0.4 (steel)
Upper End Fitting (kg)	9.28 (steel)	2.0 (steel)
Handle (kg)	N/A	0.5 (steel)
Incore Grid Spacers (kg)	4.9 (inconel)	0.33 (inconel springs)

Table 5.2.2

DESCRIPTION OF DESIGN BASIS GE 6x6 ZIRCALOY CLAD FUEL

	BWR
Fuel type	GE 6x6
Active fuel length (in.)	110
No. of fuel rods	36
Rod pitch (in.)	0.694
Cladding material	Zircaloy-2
Rod diameter (in.)	0.5645
Cladding thickness (in.)	0.035
Pellet diameter (in.)	0.494
Pellet material	UO_2
Pellet density (gm/cc)	10.412 (95% of theoretical)
Enrichment (w/o ^{235}U)	2.24
Burnup (MWD/MTU)	30,000
Cooling Time (years)	18
Specific power (MW/MTU)	16.5
Weight of UO_2 (kg)†	129.5
Weight of U (kg)†	114.2

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

† Derived from parameters in this table.

Table 5.2.3

DESCRIPTION OF DESIGN BASIS STAINLESS STEEL CLAD FUEL

	PWR	BWR
Fuel type	WE 15x15	LaCrosse 10x10
Active fuel length (in.)	144	144
No. of fuel rods	204	100
Rod pitch (in.)	0.563	0.565
Cladding material	304 SS	348H SS
Rod diameter (in.)	0.422	0.396
Cladding thickness (in.)	0.0165	0.02
Pellet diameter (in.)	0.3825	0.35
Pellet material	UO_2	UO_2
Pellet density (gm/cc)	10.412 (95% of theoretical)	10.412 (95% of theoretical)
Enrichment (w/o ^{235}U)	3.5	3.5
Burnup (MWD/MTU)†	40,000 (MPC-24 and 32)	22,500 (MPC-68)
Cooling Time (years)†	8 (MPC-24), 9 (MPC-32)	10 (MPC-68)
Specific power (MW/MTU)	37.96	29.17
No. of Water Rods	21	0
Water Rod O.D. (in.)	0.546	N/A
Water Rod Thickness (in.)	0.017	N/A

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

† Burnup and cooling time combinations are equivalent to or conservatively bound the limits in ~~Appendix B to the CoC~~*Section 2.1.9*.

Table 5.2.4

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

Lower Energy	**Upper Energy**	**~~45,000~~ *35,000* MWD/MTU ~~5~~ *3* Year Cooling**		**~~45~~*75,000* MWD/MTU ~~10~~ *8* Year Cooling**	
(MeV)	(MeV)	(MeV/s)	(Photons/s)	(MeV/s)	(Photons/s)
0.45	0.7	*2.30E+15*	*4.00E+15*	*2.52E+15*	*4.39E+15*
0.7	1.0	*9.62E+14*	*1.13E+15*	*5.41E+14*	*6.36E+14*
1.0	1.5	*2.18E+14*	*1.75E+14*	*1.66E+14*	*1.33E+14*
1.5	2.0	*2.45E+13*	*1.40E+13*	*7.51E+12*	*4.29E+12*
2.0	2.5	*3.57E+13*	*1.59E+13*	*6.94E+11*	*3.08E+11*
2.5	3.0	*9.59E+11*	*3.49E+11*	*4.99E+10*	*1.81E+10*
Total		*3.54E+15*	*5.34E+15*	*3.24E+15*	*5.16E+15*

Table 5.2.5

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

Lower Energy	Upper Energy	~~42,500~~*46,000* MWD/MTU ~~5~~*3* Year Cooling		~~52~~*47*,500 MWD/MTU ~~5~~*3* Year Cooling		~~57,500~~*75,000* MWD/MTU ~~12~~*5* Year Cooling	
(MeV)	(MeV)	(MeV/s)	(Photons/s)	(MeV/s)	(Photons/s)	(MeV/s)	(Photons/s)
0.45	0.7	*3.14E+15*	*5.45E+15*	*3.25E+15*	*5.65E+15*	*3.55E+15*	*6.17E+15*
0.7	1.0	*1.43E+15*	*1.68E+15*	*1.49E+15*	*1.75E+15*	*1.36E+15*	*1.60E+15*
1.0	1.5	*3.07E+14*	*2.46E+14*	*3.17E+14*	*2.53E+14*	*2.94E+14*	*2.35E+14*
1.5	2.0	*2.97E+13*	*1.70E+13*	*3.03E+13*	*1.73E+13*	*1.50E+13*	*8.59E+12*
2.0	2.5	*3.80E+13*	*1.69E+13*	*3.83E+13*	*1.70E+13*	*7.63E+12*	*3.39E+12*
2.5	3.0	*1.16E+12*	*4.22E+11*	*1.19E+12*	*4.33E+11*	*3.72E+11*	*1.35E+11*
Total		*4.94E+15*	*7.42E+15*	*5.12E+15*	*7.69E+15*	*5.23E+15*	*8.02E+15*

Table 5.2.6

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

Lower Energy	**Upper Energy**	***39,000 MWD/MTU 3 Year Cooling***		**40,000 MWD/MTU ~~5~~ *3* Year Cooling**		***~~47,500~~70,000* MWD/MTU ~~5~~ *6* Year Cooling**	
(MeV)	(MeV)	*(MeV/s)*	*(Photons/s)*	(MeV/s)	(Photons/s)	(MeV/s)	(Photons/s)
0.45	0.7	*1.00E+15*	*1.74E+15*	*1.02E+15*	*1.78E+15*	*1.10E+15*	*1.91E+15*
0.7	1.0	*4.25E+14*	*4.99E+14*	*4.37E+14*	*5.14E+14*	*3.21E+14*	*3.78E+14*
1.0	1.5	*9.18E+13*	*7.35E+13*	*9.40E+13*	*7.52E+13*	*7.67E+13*	*6.13E+13*
1.5	2.0	*9.19E+12*	*5.25E+12*	*9.27E+12*	*5.30E+12*	*3.55E+12*	*2.03E+12*
2.0	2.5	*1.17E+13*	*5.18E+12*	*1.17E+13*	*5.21E+12*	*1.03E+12*	*4.57E+11*
2.5	3.0	*3.69E+11*	*1.34E+11*	*3.70E+11*	*1.35E+11*	*5.83E+10*	*2.12E+10*
Total		*1.54E+15*	*2.32E+15*	*1.58E+15*	*2.38E+15*	*1.50E+15*	*2.35E+15*

Table 5.2.7

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

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

Table 5.2.8

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

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

Note: These source terms were calculated for a 144-inch fuel length. The limits in ~~Appendix B to the CoC~~*Section 2.1.9* are based on the actual 83-inch active fuel length.

Table 5.2.9

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

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

Note: These source terms were calculated for a 144-inch fuel length. The limits in ~~Appendix B to the CoC~~*Section 2.1.9* are based on the actual 122-inch active fuel length.

Table 5.2.10

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

Region	PWR	BWR
Handle	N/A	0.05
Upper End Fitting	0.1	0.1
Gas Plenum Spacer	0.1	N/A
Expansion Springs	N/A	0.1
Gas Plenum Springs	0.2	0.2
Incore Grid Spacer	1.0	1.0
Lower End Fitting	0.2	0.15

Table 5.2.11

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

Location	~~45,000~~35,000 MWD/MTU and ~~5~~3-Year Cooling (curies)	45~~75~~,000 MWD/MTU and ~~10~~8-Year Cooling (curies)
Lower End Fitting	*184.28*	*147.77*
Gas Plenum Springs	*14.06*	*11.27*
Gas Plenum Spacer	*8.07*	*6.47*
Expansion Springs	N/A	N/A
Incore Grid Spacers	*477.26*	*382.69*
Upper End Fitting	*90.39*	*72.48*
Handle	N/A	N/A

Table 5.2.12

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

Location	~~*42,500*~~*46,000* **MWD/MTU and ~~*53*~~-Year Cooling (curies)**	~~*52*~~*47*,500 **MWD/MTU and ~~*53*~~-Year Cooling (curies)**	~~*57,500*~~ *75,000* **MWD/MTU and - ~~*12*~~ *5* Year Cooling (curies)**
Lower End Fitting	*221.36*	*227.04*	*219.47*
Gas Plenum Springs	*16.89*	*17.32*	*16.74*
Gas Plenum Spacer	*9.69*	*9.94*	*9.61*
Expansion Springs	N/A	N/A	N/A
Incore Grid Spacers	*573.30*	*588.00*	*568.40*
Upper End Fitting	*108.58*	*111.36*	*107.65*
Handle	N/A	N/A	N/A

Table 5.2.13

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

Location	*39,000 MWD/MTU and 3-Year Cooling (curies)*	40,000 MWD/MTU and ~~5~~3-Year Cooling (curies)	~~47,500~~ *70,000* MWD/MTU and ~~5~~*6*-Year Cooling (curies)
Lower End Fitting	*82.47*	*82.69*	*68.73*
Gas Plenum Springs	*25.20*	*25.27*	*21.00*
Gas Plenum Spacer	*N/A*	N/A	N/A
Expansion Springs	*4.58*	*4.59*	*3.82*
Grid Spacer Springs	*37.80*	*37.90*	*31.50*
Upper End Fitting	*22.91*	*22.97*	*19.09*
Handle	*2.86*	*2.87*	*2.39*

Table 5.2.14

THIS TABLE INTENTIONALLY DELETED

Table 5.2.15

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

Lower Energy (MeV)	Upper Energy (MeV)	~~45,000~~*35,000* MWD/MTU ~~5~~*3*-Year Cooling (Neutrons/s)	~~45~~*75*,000 MWD/MTU ~~10~~*8*-Year Cooling (Neutrons/s)
1.0e-01	4.0e-01	*7.80E+06*	*5.97E+07*
4.0e-01	9.0e-01	*3.99E+07*	*3.05E+08*
9.0e-01	1.4	*3.65E+07*	*2.79E+08*
1.4	1.85	*2.70E+07*	*2.05E+08*
1.85	3.0	*4.79E+07*	*3.61E+08*
3.0	6.43	*4.33E+07*	*3.29E+08*
6.43	20.0	*3.82E+06*	*2.92E+07*
Totals		*2.06E+08*	*1.57E+09*

Table 5.2.16

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

Lower Energy (MeV)	**Upper Energy (MeV)**	***~~42,500~~46,000* MWD/MTU ~~5~~3-Year Cooling (Neutrons/s)**	**~~52~~47,500 MWD/MTU ~~5~~3-Year Cooling (Neutrons/s)**	***~~57,500~~75,000* MWD/MTU ~~12~~5-Year Cooling (Neutrons/s)**
1.0e-01	4.0e-01	*1.96E+07*	*2.19E+07*	*6.82E+07*
4.0e-01	9.0e-01	*1.00E+08*	*1.12E+08*	*3.48E+08*
9.0e-01	1.4	*9.16E+07*	*1.02E+08*	*3.18E+08*
1.4	1.85	*6.75E+07*	*7.54E+07*	*2.34E+08*
1.85	3.0	*1.19E+08*	*1.33E+08*	*4.11E+08*
3.0	6.43	*1.08E+08*	*1.21E+08*	*3.75E+08*
6.43	20.0	*9.60E+06*	*1.07E+07*	*3.34E+07*
Totals		*5.16E+08*	*5.76E+08*	*1.79E+09*

Table 5.2.17

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

Lower Energy (MeV)	Upper Energy (MeV)	*39,000 MWD/MTU 3-Year Cooling (Neutrons/s)*	40,000 MWD/MTU ~~5~~3-Year Cooling (Neutrons/s)	~~47,500~~*70,000* MWD/MTU ~~5~~*6*-Year Cooling (Neutrons/s)
1.0e-01	4.0e-01	*5.22E+06*	*5.45E+06*	*1.98E+07*
4.0e-01	9.0e-01	*2.67E+07*	*2.78E+07*	*1.01E+08*
9.0e-01	1.4	*2.44E+07*	*2.55E+07*	*9.26E+07*
1.4	1.85	*1.80E+07*	*1.88E+07*	*6.81E+07*
1.85	3.0	*3.18E+07*	*3.32E+07*	*1.20E+08*
3.0	6.43	*2.89E+07*	*3.02E+07*	*1.09E+08*
6.43	20.0	*2.56E+06*	*2.67E+06*	*9.71E+06*
Totals		*1.37E+08*	*1.44E+08*	*5.20E+08*

Table 5.2.18

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

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

Table 5.2.19

CALCULATED BWR NEUTRON SOURCE PER ASSEMBLY
FOR STAINLESS STEEL CLAD FUEL

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

Note: These source terms were calculated for a 144-inch fuel length. The limits in ~~Appendix B to the CoC~~*Section 2.1.9* are based on the actual 83-inch active fuel length.

Table 5.2.20

CALCULATED PWR NEUTRON SOURCE PER ASSEMBLY
FOR STAINLESS STEEL CLAD FUEL

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

Note: These source terms were calculated for a 144-inch fuel length. The limits in ~~Appendix B to the CoC~~*Section 2.1.9* are based on the actual 122-inch active fuel length.

Table 5.2.21

DESCRIPTION OF DESIGN BASIS ZIRCALOY CLAD MIXED OXIDE FUEL

	BWR
Fuel type	GE 6x6
Active fuel length (in.)	110
No. of fuel rods	36
Rod pitch (in.)	0.696
Cladding material	Zircaloy-2
Rod diameter (in.)	0.5645
Cladding thickness (in.)	0.036
Pellet diameter (in.)	0.482
Pellet material	UO_2 and $PuUO_2$
No. of UO_2 Rods	27
No. of $PuUO_2$ rods	9
Pellet density (gm/cc)	10.412 (95% of theoretical)
Enrichment (w/o ^{235}U)†	2.24 (UO_2 rods) 0.711 ($PuUO_2$ rods)
Burnup (MWD/MTU)	30,000
Cooling Time (years)	18
Specific power (MW/MTU)	16.5
Weight of UO_2,$PuUO_2$ (kg)††	123.3
Weight of U,Pu (kg)††	108.7

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

†† Derived from parameters in this table.

Table 5.2.22

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

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

Table 5.2.23

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

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

Table 5.2.24

INITIAL ENRICHMENTS USED IN THE SOURCE TERM CALCULATIONS

Burnup Range (MWD/MTU)	Initial Enrichment (wt.% ^{235}U)
BWR Fuel	
20,000-25,000	2.1
25,000-30,000	2.4
30,000-35,000	2.6
35,000-40,000	2.9
40,000-45,000	3.0
45,000-50,000	3.2
50,000-55,000	3.6
55,000-60,000	4.0
60,000-65,000	4.4
65,000-70,000	*4.8*
PWR Fuel	
20,000-25,000	2.3
25,000-30,000	2.6
30,000-35,000	2.9
35,000-40,000	3.2
40,000-45,000	3.4
45,000-50,000	3.6
50,000-55,000	3.9
55,000-60,000	4.2
60,000-65,000	4.5
65,000-70,000	4.8
70,000-75,000	*5.0*

Note: The burnup ranges do not overlap. Therefore, 20,000-25,000 MWD/MTU means 20,000-24,999.9 MWD/MTU, etc. *This note does not apply to the maximum burnups of 70,000 and 75,000 MWD/MTU.*

Table 5.2.25 *(page 1 of 2)*

DESCRIPTION OF EVALUATED ZIRCALOY CLAD PWR FUEL

Assembly	WE 14×14	*WE 14x14*	WE 15×15	WE 17×17	*WE 17x17*
Fuel assembly array class	*14x14B*	*14x14A*	*15x15ABC*	*17x17B*	*17x17A*
Active fuel length (in.)	144	*144*	144	144	*144*
No. of fuel rods	179	*179*	204	264	*264*
Rod pitch (in.)	0.556	*0.556*	0.563	0.496	*0.496*
Cladding material	Zr-4	*Zr-4*	Zr-4	Zr-4	*Zr-4*
Rod diameter (in.)	0.422	*0.4*	0.422	0.374	*0.36*
Cladding thickness (in.)	0.0243	*0.0243*	0.0245	0.0225	*0.0225*
Pellet diameter (in.)	0.3659	*0.3444*	0.3671	0.3232	*0.3088*
Pellet material	UO_2	*UO_2*	UO_2	UO_2	*UO_2*
Pellet density (gm/cc) (% of theoretical)	*10.522 (96%)*	*10.522 (96%)*	*10.522 (96%)*	*10.522 (96%)*	*10.522 (96%)*
Enrichment (wt.% ^{235}U)	3.4	*3.4*	3.4	3.4	*3.4*
Burnup (MWD/MTU)	40,000	*40,000*	40,000	40,000	*40,000*
Cooling time (years)	5	*5*	5	5	*5*
Power/assembly (MW)	*15.0*	*15.0*	*18.6*	*20.4*	*20.4*
Specific power (MW/MTU)	*36.409*	*41.097*	*39.356*	*43.031*	*47.137*
Weight of UO_2 (kg)†	*467.319*	*414.014*	*536.086*	*537.752*	*490.901*
Weight of U (kg)†	*411.988*	*364.994*	*472.613*	*474.082*	*432.778*
No. of Guide Tubes	17	*17*	21	25	*25*
Guide Tube O.D. (in.)	0.539	*0.539*	0.546	0.474	*0.474*
Guide Tube Thickness (in.)	0.0170	*0.0170*	0.0170	0.0160	*0.0160*

† Derived from parameters in this table.

Table 5.2.25 *(page 2 of 2)*

DESCRIPTION OF EVALUATED ZIRCALOY CLAD PWR FUEL

Assembly	CE 14×14	CE 16×16	B&W 15×15	B&W 17×17
Fuel assembly array class	*14x14C*	*16x16A*	*15x15DEFH*	*17x17C*
Active fuel length (in.)	144	150	144	144
No. of fuel rods	176	236	208	264
Rod pitch (in.)	0.580	0.5063	0.568	0.502
Cladding material	Zr-4	Zr-4	Zr-4	Zr-4
Rod diameter (in.)	0.440	0.382	0.428	0.377
Cladding thickness (in.)	0.0280	0.0250	0.0230	0.0220
Pellet diameter (in.)	*0.3805*	0.3255	0.3742	0.3252
Pellet material	UO_2	UO_2	UO_2	UO_2
Pellet density (gm/cc) (95% of theoretical)	*10.522 (96%)*	*10.522 (96%)*	10.412 *(95%)*	*10.522 (96%)*
Enrichment (wt.% ^{235}U)	3.4	3.4	3.4	3.4
Burnup (MWD/MTU)	40,000	40,000	40,000	40,000
Cooling time (years)	5	5	5	5
Power/assembly (MW)	*13.7*	*17.5*	*19.819*	*20.4*
Specific power (MW/MTU)	*31.275*	*39.083*	40	*42.503*
Weight of UO_2 (kg)†	*496.887*	*507.9*	562.029	*544.428*
Weight of U (kg)†	*438.055*	*447.764*	495.485	*479.968*
No. of Guide Tubes	5	5	17	25
Guide Tube O.D. (in.)	1.115	0.98	0.53	0.564
Guide Tube Thickness (in.)	0.0400	0.0400	0.0160	0.0175

† Derived from parameters in this table.

Table 5.2.26 *(page 1 of 2)*

DESCRIPTION OF EVALUATED ZIRCALOY CLAD BWR FUEL

Array Type	7×7	8×8	*8x8*	*9x9*	*9x9*
Fuel assembly array class	*7x7B*	*8x8B*	*8x8CDE*	*9x9A*	*9x9B*
Active fuel length (in.)	144	144	*150*	*144*	*150*
No. of fuel rods	49	64	*62*	*74*	*72*
Rod pitch (in.)	0.738	0.642	*0.64*	*0.566*	*0.572*
Cladding material	Zr-2	Zr-2	*Zr-2*	*Zr-2*	*Zr-2*
Rod diameter (in.)	0.570	0.484	*0.493*	*0.44*	*0.433*
Cladding thickness (in.)	0.0355	0.02725	*0.034*	*0.028*	*0.026*
Pellet diameter (in.)	0.488	0.4195	*0.416*	*0.376*	*0.374*
Pellet material	UO_2	UO_2	*UO_2*	*UO_2*	*UO_2*
Pellet density (gm/cc) (% of theoretical)	10.412 (95%)	10.412 (95%)	*10.412 (95%)*	*10.522 (96%)*	*10.522 (96%)*
Enrichment (wt.% ^{235}U)	3.0	3.0	*3.0*	*3.0*	*3.0*
Burnup (MWD/MTU)	40,000	40,000	*40,000*	*40,000*	*40,000*
Cooling time (years)	5	5	*5*	*5*	*5*
Power/assembly (MW)	*5.96*	*5.75*	*5.75*	*5.75*	*5.75*
Specific power (MW/MTU)	30	30	*30.24*	*31.97*	*31.88*
Weight of UO_2 (kg)†	225.177	217.336	*215.673*	*204.006*	*204.569*
Weight of U (kg)†	198.516	191.603	*190.137*	*179.852*	*180.348*
No. of Water Rods	0	0	*2*	*2*	*1*
Water Rod O.D. (in.)	n/a	n/a	*0.493*	*0.98*	*1.516*
Water Rod Thickness (in.)	n/a	n/a	*0.034*	*0.03*	*0.0285*

† Derived from parameters in this table.

Table 5.2.26 *(page 1 of 2)*

DESCRIPTION OF EVALUATED ZIRCALOY CLAD BWR FUEL

Array Type	*9x9*	9x9	*9x9*	10x10	*10x10*
Fuel assembly array class	*9x9CD*	*9x9EF*	*9x9G*	*10x10AB*	*10x10C*
Active fuel length (in.)	*150*	144	*150*	144	*150*
No. of fuel rods	*80*	*76*	*72*	92	*96*
Rod pitch (in.)	*0.572*	0.572	*0.572*	0.510	*0.488*
Cladding material	*Zr-2*	Zr-2	*Zr-2*	Zr-2	*Zr-2*
Rod diameter (in.)	*0.423*	0.443	*0.424*	0.404	*0.378*
Cladding thickness (in.)	*0.0295*	0.0285	*0.03*	0.0260	*0.0243*
Pellet diameter (in.)	*0.3565*	0.3745	*0.3565*	0.345	*0.3224*
Pellet material	*UO_2*	UO_2	*UO_2*	UO_2	*UO_2*
Pellet density (gm/cc) (% of theoretical)	*10.522 (96%)*	10.522 (96%)	*10.522 (96%)*	10.522 (96%)	*10.522 (96%)*
Enrichment (wt.% ^{235}U)	*3.0*	3.0	*3.0*	3.0	*3.0*
Burnup (MWD/MTU)	*40,000*	40,000	*40,000*	40,000	*40,000*
Cooling time (years)	*5*	5	*5*	5	*5*
Power/assembly (MW)	*5.75*	*5.75*	*5.75*	*5.75*	*5.75*
Specific power (MW/MTU)	*31.58*	*31.38*	*35.09*	*30.54*	*32.18*
Weight of UO_2 (kg)†	*206.525*	*207.851*	*185.873*	213.531	*202.687*
Weight of U (kg)†	*182.073*	*183.242*	*163.865*	188.249	*178.689*
No. of Water Rods	*1*	*5*	*1*	2	*1*
Water Rod O.D. (in.)	*0.512*	0.546	*1.668*	0.980	*Note 1*
Water Rod Thickness (in.)	*0.02*	0.0120	*0.032*	0.0300	*Note 1*

Note 1: 10x10C has a diamond shaped water rod with 4 additional segments dividing the fuel rods into four quadrants.

† Derived from parameters in this table.

Table 5.2.27

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

Assembly	*WE 14x14*	WE 14x14	WE 15x15	*WE 17x17*	WE 17x17	CE 14x14	CE 16x16	B&W 15x15	B&W 17x17
Array class	*14x14A*	*14x14B*	*15x15 ABC*	*17x17A*	*17x17B*	*14x14C*	*16x16A*	*15x15 DEFH*	*17x17C*
Neutrons/sec	*1.76E+8* *1.78E+8*	2.32E+8 2.35E+8	2.70E+8 2.73E+8	*2.18E+8*	2.68E+8	2.32E+8	2.38E+8	2.94E+8	2.68E+8
Photons/sec (0.45-3.0 MeV)	*2.88E+15* *2.93E+15*	3.28E+15 3.32E+15	3.80E+15 3.86E+15	*3.49E+15*	3.85E+15	3.37E+15	3.57E+15	4.01E+15	3.89E+15
Thermal power (watts)	*809.5* *820.7*	*923.5* *933.7*	*1073* *1086*	*985.6*	*1090*	*946.6*	*1005*	1137	*1098*

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

Table 5.2.28

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

Assembly	7×7	*8x8*	*8x8*	*9x9*	*9x9*	*9x9*	*9x9*	*9x9*	*10x10*	*10x10*
Array Class	*7x7B*	*8x8B*	*8x8CDE*	*9x9A*	*9x9B*	*9x9CD*	*9x9EF*	*9x9G*	*10x10AB*	*10x10C*
Neutrons/sec	1.33E+8	*1.22E+8*	*1.22E+8*	*1.13E+8*	*1.06E+8*	*1.09E+8*	*1.24E+8*	*9.15E+7*	*1.24E+8*	*1.07E+8*
Photons/sec (0.45-3.0 MeV)	1.55E+15	*1.49E+15*	*1.48E+15*	*1.41E+15*	*1.40E+15*	*1.42E+15*	*1.45E+15*	*1.28E+15*	*1.48E+15*	*1.40E+15*
Thermal power (watts)	435.5	*417.3*	*414.2*	*394.2*	*389.8*	*395*	*405.8*	*356.9*	*413.5*	*389.2*

Table 5.2.29

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

Fuel Assembly Class	Decay Heat from the DOE Database (watts/assembly)	Decay Heat from ~~Design Basis Fuel~~ *Source Term Calculations* (watts/assembly)
PWR Fuel		
B&W 15x15	752.0	827.5
B&W 17x17	732.9	~~827.5~~*802.7*
CE 16x16	653.7	~~827.5~~*734.3*
CE 14x14	601.3	~~827.5~~*694.9*
WE 17x17	742.5	~~827.5~~*795.4*
WE 15x15	762.2	~~827.5~~*796.2*
WE 14x14	649.6	~~827.5~~*682.9*
BWR Fuel		
7x7	310.9	315.7
8x8	296.6	~~315.7~~*302.8*
9x9	275.0	~~315.7~~*286.8*

Notes:

1. The ~~PWR and BWR design basis fuels are the B&W 15x15 and the GE 7x7, respectively~~*decay heat from the source term calculations is the maximum value calculated for that fuel assembly class.*
2. The decay heat values from the database include contributions from in-core material (e.g. spacer grids).
3. Information on the 10x10 was not available in the DOE database. However, based on the results in Table 5.2.28, the actual decay heat values from the 10x10 would be very similar to the values shown above for the 8x8.
4. *The enrichments used for the column labeled "Decay Heat from Source Term Calculations" were consistent with Table 5.2.24.*

† Reference [5.2.7].

Table 5.2.30

DESCRIPTION OF DESIGN BASIS BURNABLE POISON ROD ASSEMBLY AND THIMBLE PLUG DEVICE

Region	BPRA	TPD
Upper End Fitting (kg of steel)	2.62	2.3
Upper End Fitting (kg of inconel)	0.42	0.42
Gas Plenum Spacer (kg of steel)	0.77488	1.71008
Gas Plenum Springs (kg of steel)	0.67512	1.48992
In-core (kg of steel)	13.2	N/A

Table 5.2.31

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

Region	BPRA	TPD
Upper End Fitting (curies Co-60)	~~30.4~~32.7	25.21
Gas Plenum Spacer (curies Co-60)	~~4.6~~5.0	9.04
Gas Plenum Springs (curies Co-60)	~~8.2~~8.9	15.75
In-core (curies Co-60)	~~787.8~~848.4	N/A

Table 5.2.32

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

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

Table 5.2.33

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

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

Table 5.2.34

DESIGN BASIS SOURCE TERMS FOR CONTROL ROD ASSEMBLY CONFIGURATIONS

Axial Dimensions Relative to Bottom of Active Fuel			Photons/sec from AgInCd			Curies Co-60 from Inconel
Start (in)	Finish (in)	Length (in)	0.3-0.45 MeV	0.45-0.7 MeV	0.7-1.0 MeV	
Configuration 1 - 10% Inserted - *80.8 watts decay heat*						
0.0	15.0	15.0	1.91e+14	1.78e+14	1.42e+14	1111.38
15.0	18.8125	3.8125	9.71e+12	9.05e+12	7.20e+12	56.50
18.8125	28.25	9.4375	1.20e+13	1.12e+13	8.92e+12	69.92
Configuration 2 - Fully Removed - *8.25 watts decay heat*						
0.0	3.8125	3.8125	9.71e+12	9.05e+12	7.20e+12	56.50
3.8125	13.25	9.4375	1.20e+13	1.12e+13	8.92e+12	69.92

Table 5.2.35

DESIGN BASIS SOURCE TERMS FROM AXIAL POWER SHAPING ROD CONFIGURATIONS

Axial Dimensions Relative to Bottom of Active Fuel			
Start (in)	**Finish (in)**	**Length (in)**	**Curies of Co-60**
Configuration 1 - 10% Inserted - *46.2 watts decay heat*			
0.0	15.0	15.0	2682.57
15.0	18.8125	3.8125	136.36
18.8125	28.25	9.4375	168.78
Configuration 2 - Fully Removed - *4.72 watts decay heat*			
0.0	3.8125	3.8125	136.36
3.8125	13.25	9.4375	168.78
Configuration 3 - Fully Inserted - *178.9 watts decay heat*			
0.0	63.0	63.0	11266.80
63.0	66.8125	3.8125	136.36
66.8125	76.25	9.4375	168.78

Table 5.2.36

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

	BWR
Fuel type	8x8
Active fuel length (in.)	110.5
No. of UO_2 fuel rods	55
No. of UO_2/ThO_2 fuel rods	9
Rod pitch (in.)	0.523
Cladding material	zircaloy
Rod diameter (in.)	0.412
Cladding thickness (in.)	0.025
Pellet diameter (in.)	0.358
Pellet material	98.2% ThO_2 and 1.8% UO_2 for UO_2/ThO_2 rods
Pellet density (gm/cc)	10.412
Enrichment (w/o ^{235}U)	93.5 in UO_2 for UO_2/ThO_2 rods and 1.8 for UO_2 rods
Burnup (MWD/MTIHM)	16,000
Cooling Time (years)	18
Specific power (MW/MTIHM)	16.5
Weight of THO_2 and UO_2 (kg)†	121.46
Weight of U (kg)†	92.29
Weight of Th (kg)†	14.74

† Derived from parameters in this table.

Table 5.2.37

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

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

Table 5.2.38

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

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

5.3 MODEL SPECIFICATIONS

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

As discussed in Section 5.1.1, off-normal conditions do not have any implications for the shielding analysis. Therefore, the MCNP models and results developed for the normal conditions also represent the off-normal conditions. Section 5.1.2 discussed the accident conditions and stated that the only accident that would impact the shielding analysis would be a loss of the neutron shield (water) in the HI-TRAC. Therefore, the MCNP model of the normal HI-TRAC condition has the neutron shield in place while the accident condition replaces the neutron shield with void. Section 5.1.2 also mentioned that there is no credible accident scenario that would impact the HI-STORM shielding analysis. Therefore, models and results for the normal and accident conditions are identical for the HI-STORM overpack.

5.3.1 Description of the Radial and Axial Shielding Configuration

Chapter 1 provides the drawings that describe the HI-STORM 100 System, including the HI-TRAC transfer casks. These drawings, using nominal dimensions, were used to create the MCNP models used in the radiation transport calculations. Modeling deviations from these drawings are discussed below. Figures 5.3.1 through 5.3.6 show cross sectional views of the HI-STORM 100 overpack and MPC as it was modeled in MCNP for each of the MPCs. Figures 5.3.1 through 5.3.3 were created with the MCNP two-dimensional plotter and are drawn to scale. The inlet and outlet vents were modeled explicitly, therefore, streaming through these components is accounted for in the calculations of the dose adjacent to the overpack and at 1 meter. Figure 5.3.7 shows a cross sectional view of the 100-ton HI-TRAC with the MPC-24 inside as it was modeled in MCNP. Since the fins and pocket trunnions were modeled explicitly, neutron streaming through these components is accounted for in the calculations of the dose adjacent to the overpack and 1 meter dose. In Section 5.4.1, the dose effect of localized streaming through these compartments is analyzed.

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

MPC Modeling Discrepancies

1. In the MPCs, there is a sump in the baseplate to enhance draining of the MPC. This localized reduction in the thickness of the baseplate was not modeled. Since there is significant shielding and distance in both the HI-TRAC and the HI-STORM outside the MPC baseplate, this localized reduction in shielding will not affect the calculated dose rates outside the HI-TRAC or the HI-STORM.

2. The design configuration of the MPC-24 has been enhanced for criticality purposes. The general location of the 24 assemblies remains basically the same, therefore the shielding analysis continues to use the superseded configuration. Since the new MPC-24 configuration and the configuration of the MPC-24E are almost identical, the analysis of the earlier MPC-24 configuration is valid for the MPC-24E as well. Figure 5.3.21 shows the superseded and current configuration for the MPC-24 for comparison.

3. The sheathing thickness on the new MPC-24 configuration was reduced from 0.06 inches to 0.0235 inches. However, the model still uses 0.06 inches. This discrepancy is compensated for by the use of 9/32 inch cell walls and 5 inch boral on the periphery as described above. MCNP calculations were performed with the new MPC-24 configuration in the 100-ton HI-TRAC for comparison to the superceded configuration. These results indicate that on the side of the overpack, the dose rates decrease by approximately 12% on the surface. These results demonstrate that using the superceded MPC-24 design is conservative.

HI-TRAC Modeling Discrepancies

1. The pocket trunnion on the HI-TRAC 125 was modeled as penetrating the lead. This is conservative for gamma dose rates as it reduces effective shielding thickness. The HI-TRAC 125D does not use pocket trunnions.

2. The lifting blocks in the top lid of the 125-ton HI-TRACs were not modeled. Holtite-A was modeled instead. This is a small, localized item and will not impact the dose rates.

3. The door side plates that are in the middle of the transfer lid of the HI-TRAC 125 are not modeled. This is acceptable because the dose location calculated on the bottom of the transfer lid is in the center.

4. The outside diameter of the Holtite-A portion of the top lid of the 125-ton HI-TRACs was modeled as 4 inches larger than it is due to a design enhancement. This is acceptable because the peak dose rates on the top lid occur on the inner portions of the lid.

HI-STORM Modeling Discrepancies

1. The steel channels in the cavity between the MPC and overpack were not modeled. This is conservative since it removes steel that would provide a small amount of additional shielding.

2. The bolt anchor blocks were not explicitly modeled. Concrete was used instead. These are small, localized items and will not impact the dose rates.

3. In the HI-STORM 100S model, the exit vents were modeled as being inline with the inlet vents. In practice, they are rotated 45 degrees and positioned above the short radial plates. Therefore, this modeling change has the exit vents positioned above the full length radial plates. This modeling change has minimal impact on the dose rates at the exit vents.

4. The short radial plates in the HI-STORM 100S overpack were modeled in MCNP even though they are optional.

5. The pedestal baseplate, which is steel with holes for pouring concrete, in the HI-STORM overpacks was modeled as concrete rather than steel. This is acceptable because this piece of steel is positioned at the bottom of the pedestal below 5 inches of steel and a minimum of 11.5 inches of concrete and therefore will have no impact on the dose rates at the bottom vent.

6. Minor penetrations in the body of the overpack (e.g. holes for grounding straps) are not modeled as these are small localized effects which will not affect the off-site dose rates.

7. In June 2001, the inner shield shell of the HI-STORM 100 overpack was removed and the concrete density in the body of the overpack (not the pedestal of lid) was increased to compensate. Appendix 5.E presents a comparison of the dose rates calculated for a HI-STORM 100 overpack with and without the inner shield shell. The MPC-24 was used in this comparison. The results indicate that there is very little difference in the calculated dose rates when the inner shield shell is removed and the concrete density is increased. Therefore, all HI-STORM 100 analysis presented in the main portion of this chapter includes the inner shield shell.

8. The drawings in Section 1.5 indicate that the HI-STORM 100S has a variable height. This is achieved by adjusting the height of the body of the overpack. The pedestal height is not adjusted. Conservatively, all calculations in this chapter used the shorter height for the HI-STORM 100S.

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

5.3.1.1 Fuel Configuration

As described earlier, the active fuel region is modeled as a homogenous zone. The end fittings and the plenum regions are also modeled as homogenous regions of steel. The masses of steel used in these regions are shown in Table 5.2.1. The axial description of the design basis fuel assemblies is provided in Table 5.3.1. Figures 5.3.8 and 5.3.9 graphically depict the location of the PWR and BWR fuel assemblies within the HI-STORM 100 System. The axial locations of the Boral, basket, inlet vents, and outlet vents are shown in these figures.

5.3.1.2 Streaming Considerations

The MCNP model of the HI-STORM overpack completely describes the inlet and outlet vents, thereby properly accounting for their streaming effect. The gamma shield cross plates located in the inlet and outlet vents, which effectively reduce the gamma dose in these locations, are modeled explicitly.

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

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

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

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

5.3.2 Regional Densities

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

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

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

The MPCs in the HI-STORM 100 System 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 thickness and minimum ^{10}B areal density are the same for Boral and Metamic. 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.

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

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

Table 5.3.1

DESCRIPTION OF THE AXIAL MCNP MODEL OF THE FUEL ASSEMBLIES†

Region	Start (in.)	Finish (in.)	Length (in.)	Actual Material	Modeled Material
PWR					
Lower End Fitting	0.0	7.375	7.375	SS304	SS304
Space	7.375	8.375	1.0	zircaloy	void
Fuel	8.375	152.375	144	fuel & zircaloy	fuel
Gas Plenum Springs	152.375	156.1875	3.8125	SS304 & zircaloy	SS304
Gas Plenum Spacer	156.1875	160.5625	4.375	SS304 & zircaloy	SS304
Upper End Fitting	160.5625	165.625	5.0625	SS304	SS304
BWR					
Lower End Fitting	0.0	7.385	7.385	SS304	SS304
Fuel	7.385	151.385	144	fuel & zircaloy	fuel
Space	151.385	157.385	6	zircaloy	void
Gas Plenum Springs	157.385	166.865	9.48	SS304 & zircaloy	SS304
Expansion Springs	166.865	168.215	1.35	SS304	SS304
Upper End Fitting	168.215	171.555	3.34	SS304	SS304
Handle	171.555	176	4.445	SS304	SS304

† All dimensions start at the bottom of the fuel assembly. The length of the lower fuel spacer must be added to the distances to determine the distance from the top of the MPC baseplate.

Table 5.3.2

COMPOSITION OF THE MATERIALS IN THE HI-STORM 100 SYSTEM

Component	Density (g/cm³)	Elements	Mass Fraction (%)
Uranium Oxide	10.412	^{235}U	2.9971(BWR) 3.2615(PWR)
		^{238}U	85.1529(BWR) 84.8885(PWR)
		O	11.85
Boral†	2.644	^{10}B	4.4226 (MPC-68 and MPC-32 in HI-STORM & HI-TRAC; MPC-24 in HI-STORM)4.367 (MPC-24 in HI-TRAC)
		^{11}B	20.1474 (MPC-68 and MPC-32 in HI-STORM & HI-TRAC; MPC-24 in HI-STORM) 19.893 (MPC-24 in HI-TRAC)
		Al	68.61 (MPC-68 and MPC-32 in HI-STORM & HI-TRAC; MPC-24 in HI-STORM) 69.01 (MPC-24 in HI-TRAC)
		C	6.82 (MPC-68 and MPC-32 in HI-STORM & HI-TRAC; MPC-24 in HI-STORM) 6.73 (MPC-24 in HI-TRAC)
SS304	7.92	Cr	19
		Mn	2
		Fe	69.5
		Ni	9.5
Carbon Steel	7.82	C	0.5
		Fe	99.5
Zircaloy	6.55	Zr	100

† All B-10 loadings in the Boral compositions are conservatively lower than the values defined in the Bill of Materials.

Table 5.3.2 (continued)

COMPOSITION OF THE MATERIALS IN THE HI-STORM 100 SYSTEM

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

Table 5.3.2 (continued)

COMPOSITION OF THE MATERIALS IN THE HI-STORM 100 SYSTEM

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

Table 5.3.2 (continued)

COMPOSITION OF THE MATERIALS IN THE HI-STORM 100 SYSTEM

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

Table 5.3.3

COMPOSITION OF THE FUEL PELLETS IN THE MIXED OXIDE FUEL ASSEMBLIES

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

5.4 SHIELDING EVALUATION

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

The energy distribution of the source term, as described earlier, is used explicitly in the MCNP model. A different MCNP calculation is performed for each of the three source terms (neutron, decay gamma, and ^{60}Co). The axial distribution of the fuel source term is described in Table 2.1.11 and Figures 2.1.3 and 2.1.4. The PWR and BWR axial burnup distributions were obtained from References [5.4.5] and [5.4.6], respectively. These axial distributions were obtained from operating plants and are representative of PWR and BWR fuel with burnups greater than 30,000 MWD/MTU. The ^{60}Co source in the hardware was assumed to be uniformly distributed over the appropriate regions.

It has been shown that the neutron source strength varies as the burnup level raised by the power of 4.2. Since this relationship is non-linear and since the burnup in the axial center of a fuel assembly is greater than the average burnup, the neutron source strength in the axial center of the assembly is greater than the relative burnup times the average neutron source strength. In order to account for this effect, the neutron source strength in each of the 10 axial nodes listed in Table 2.1.11 was determined by multiplying the average source strength by the relative burnup level raised to the power of 4.2. The peak relative burnups listed in Table 2.1.11 for the PWR and BWR fuels are 1.105 and 1.195 respectively. Using the power of 4.2 relationship results in a 37.6% ($1.105^{4.2}/1.105$) and 76.8% ($1.195^{4.2}/1.195$) increase in the neutron source strength in the peak nodes for the PWR and BWR fuel respectively. The total neutron source strength increases by 15.6% for the PWR fuel assemblies and 36.9% for the BWR fuel assemblies.

MCNP was used to calculate doses at the various desired locations. MCNP calculates neutron or photon flux and these values can be converted into dose by the use of dose response functions. This is done internally in MCNP and the dose response functions are listed in the input file in Appendix 5.C. The response functions used in these calculations are listed in Table 5.4.1 and were taken from ANSI/ANS 6.1.1, 1977 [5.4.1].

The dose rate at the various locations were calculated with MCNP using a two step process. The first step was to calculate the dose rate for each dose location per starting particle for each neutron and gamma group in the fuel and each axial location in the end fittings. The second and last step was to multiply the dose rate per starting particle for each group or starting location by the source strength (i.e. particles/sec) in that group or location and sum the resulting dose

rates for all groups in each dose location. The standard deviations of the various results were statistically combined to determine the standard deviation of the total dose in each dose location.

The HI-STORM shielding analysis was performed for conservative burnup and cooling time combinations which bound the uniform and regionalized loading specifications for zircaloy clad fuel specified in ~~Appendix B to the CoC~~*Section 2.1.9*. Therefore, the HI-STORM shielding analysis presented in this chapter is conservatively bounding for the MPC-24, MPC-32, and MPC-68.

Tables 5.1.1 through 5.1.3 provide the maximum dose rates adjacent to the HI-STORM overpack during normal conditions for each of the MPCs. Tables 5.1.4 through 5.1.6 provide the maximum dose rates at one meter from the overpack. A detailed discussion of the normal, off-normal, and accident condition dose rates is provided in Sections 5.1.1 and 5.1.2.

Tables 5.1.7 and 5.1.8 provide dose rates for the 100-ton and 125-ton HI-TRAC transfer casks, respectively, with the MPC-24 loaded with design basis fuel in the normal condition, in which the MPC is dry and the HI-TRAC water jacket is filled with water. Table 5.4.2 shows the corresponding dose rates adjacent to and one meter away from the 100-ton HI-TRAC for the fully flooded MPC condition with an empty water-jacket (condition in which the HI-TRAC is removed from the spent fuel pool). Table 5.4.3 shows the dose rates adjacent to and one meter away from the 100-ton HI-TRAC for the fully flooded MPC condition with the water jacket filled with water (condition in which welding operations are performed). Dose locations 4 and 5, which are on the top and bottom of the HI-TRAC were not calculated at the one-meter distance for these configurations. For the conditions involving a fully flooded MPC, the internal water level was 10 inches below the MPC lid. These dose rates represent the various conditions of the HI-TRAC during operations. Comparing these results to Table 5.1.7 indicates that the dose rates in the upper and lower portions of the HI-TRAC are reduced by about 50% with the water in the MPC. The dose at the center of the HI-TRAC is reduced by approximately 50% when there is also water in the water jacket and is essentially unchanged when there is no water in the water jacket as compared to the normal condition results shown in Table 5.1.7.

The burnup and cooling time combination of ~~42,500~~*46,000* MWD/MTU and ~~5~~ *3* years was selected for the 100-ton MPC-24 HI-TRAC analysis because this combination of burnup and cooling time results in the highest dose rates, and therefore, bounds all other requested combinations in the 100-ton HI-TRAC. For comparison, dose rates corresponding to a burnup of ~~52,500~~*75,000* MWD/MTU and ~~10~~ *5* year cooling time for the MPC-24 are provided in Table 5.4.4. The dose rate at 1 meter from the pool lid was not calculated because a concrete floor was placed 6 inches below the pool lid to account for potential ground scattering. These results clearly indicate that as the burnup and cooling time increase, the reduction in the gamma dose rate due to the increased cooling time results in a net decrease in the total dose rate. This result is due to the fact that the dose rates surrounding the 100-ton HI-TRAC transfer cask are gamma dominated.

In contrast, the dose rates surrounding the HI-TRAC 125 and 125D transfer casks have significantly higher neutron component. Therefore, the dose rates at ~~57,500~~*75,000* MWD/MTU burnup and ~~12~~*5* year cooling are ~~slightly~~ higher than the dose rates at ~~42,500~~*46,000* MWD/MTU burnup and ~~5~~*3* year cooling. The dose rates for the 125-ton HI-TRACs with the MPC-24 at ~~57,500~~*75,000* MWD/MTU and ~~12~~*5* year cooling are listed in Table 5.1.8 of Section 5.1. For comparison, dose rates corresponding to a burnup of ~~42,500~~*46,000* MWD/MTU and ~~5~~*3* year cooling time for the MPC-24 are provided in Table 5.4.5.

Tables 5.4.9 and 5.4.10 provide dose rates adjacent to and one meter away from the 100-ton HI-TRAC with the MPC-68 at burnup and cooling time combinations of ~~40~~*39*,000 MWD/MTU and ~~5~~*3* years and ~~50~~*70*,000 MWD/MTU and ~~10~~*6* years, respectively. The dose rate at 1 meter from the pool lid was not calculated because a concrete floor was placed 6 inches below the pool lid to account for potential ground scattering. These results demonstrate that the dose rates on contact at the top and bottom of the 100-ton HI-TRAC are somewhat higher in the MPC-68 case than in the MPC-24 case. However, the MPC-24 produces higher dose rates than the MPC-68 at the center of the HI-TRAC, on-contact, and at locations ~~1 to 2 feet~~*1 meter* away from the HI-TRAC. Therefore, the MPC-24 is still used for the exposure calculations in Chapter 10 of the FSAR.

Tables 5.4.11 and 5.4.12 provide dose rates adjacent to and one meter away from the 100-ton HI-TRAC with the MPC-32 at burnup and cooling time combinations of ~~32,500~~*35,000* MWD/MTU and ~~5~~*3* years and ~~45~~*75*,000 MWD/MTU and ~~10~~*8* years, respectively. The dose rate at 1 meter from the pool lid was not calculated because a concrete floor was placed 6 inches below the pool lid to account for potential ground scattering. These results demonstrate that the dose rates on contact at the top ~~and bottom~~ of the 100-ton HI-TRAC are somewhat higher in the MPC-32 case than in the MPC-24 case. However, the MPC-24 produces ~~comparable or~~ higher dose rates than the MPC-32 at the center of the HI-TRAC, on-contact, and at locations ~~1 to 2 feet~~*1 meter* away from the HI-TRAC. Therefore, the MPC-24 is still used for the exposure calculations in Chapter 10 of the FSAR.

As mentioned in Section 5.0, all MPCs offer a regionalized loading pattern as described in ~~Appendix B to the CoC~~*Section 2.1.9*. This loading pattern authorizes fuel of higher decay heat than uniform loading (i.e. higher burnups and shorter cooling times) to be stored in the center region, region 1, of the MPC. The outer region, region 2, of the MPC in regionalized loading is authorized to store fuel of lower decay heat than uniform loading (i.e. lower burnups and longer cooling times). From a shielding perspective, the older fuel on the outside provides shielding for the inner fuel in the radial direction. Regionalized patterns were specifically analyzed in each MPC in the 100-ton HI-TRAC. Based on analysis using the same burnup and cooling times in region 1 and 2 the following percentages were calculated for dose location 2 on the 100-ton HI-TRAC.

- Approximately 21%, 27%, and 8% of the neutron dose at the edge of the water jacket comes from region 1 fuel assemblies in the MPC-32, MPC-68, and MPC-24

respectively. Region 1 contains 12 (38% of total), 32 (47% of total), and 4 (17% of total) assemblies in the MPC-32, MPC-68, and MPC-24 respectively.

- Approximately 1%, 2%, and 0.2% of the photon dose at the edge of the water jacket comes from region 1 fuel assemblies in the MPC-32, MPC-68, and MPC-24 respectively.

These results clearly indicate that the outer fuel assemblies shield almost all of the gamma source from the inner assemblies in the radial direction and a significant percentage of the neutron source. The conclusion from this analysis is that the total dose rate on the external radial surfaces of the cask can be greatly reduced by placing longer cooled and lower burnup fuels on the outside of the basket. In the axial direction, regionalized loading results in higher dose rates in the center portion of the cask since the region 2 assemblies are not shielding the region 1 assemblies for axial dose locations.

~~All~~ *Bounding* burnup and cooling time combinations for regionalized loading were analyzed and compared to the dose rates from uniform loading patterns. It was concluded that, in general, the radial dose rates from regionalized loading are bounded by the radial dose rates from uniform loading patterns. Therefore, dose rates for specific regionalized loading patterns are not presented in this chapter. In the axial direction, the reverse may be true since the inner fuel assemblies in a regionalized loading pattern have a higher burnup than the assemblies in the uniform loading patterns. However, as depicted in the graphical data in Section 5.1.1, the dose rate along the pool or transfer lids decrease substantially moving radially outward from the center of the lid. Therefore, this increase in the dose rate in the center of the lids due to regionalized loading does not significantly impact the occupational exposure. Section 5.4.9 provides additional discussion on regionalized loading dose rates compared to uniform loading dose rates.

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

Since MCNP is a statistical code, there is an uncertainty associated with the calculated values. In MCNP the uncertainty is expressed as the relative error which is defined as the standard deviation of the mean divided by the mean. Therefore, the standard deviation is represented as a percentage of the mean. The relative error for the total dose rates presented in this chapter were typically less than 5% and the relative error for the individual dose components was typically less than 10%.

5.4.1 Streaming Through Radial Steel Fins and Pocket Trunnions and Azimuthal Variations

The HI-STORM 100 overpack and the HI-TRAC utilize radial steel fins for structural support and cooling. The attenuation of neutrons through steel is substantially less than the attenuation of neutrons through concrete and water. Therefore, it is possible to have neutron streaming through the fins that could result in a localized dose peak. The reverse is true for photons, which would

result in a localized reduction in the photon dose. In addition to the fins, the pocket trunnions in the HI-TRAC 100 and 125 are essentially blocks of steel that are approximately 12 inches wide and 12 inches high. The effect of the pocket trunnion on neutron streaming and photon transmission will be more substantial than the effect of a single fin.

Analysis of the pocket trunnions in the HI-TRAC 100 and 125 and the steel fins in the HI-TRAC 100, 125, and 125D indicate that neutron streaming is noticeable at the surface of the transfer cask. The neutron dose rate on the surface of the pocket trunnion is approximately 5 times higher than the circumferential average dose rate at that location. The gamma dose rate is approximately 10 times lower than the circumferential average dose rate at that location. The streaming at the rib location is the largest in the HI-TRAC 125D because the ribs are thicker than in the HI-TRAC 100 or 125. The neutron dose rate on the surface of the rib in the 125D is approximately 3 times higher than the circumferential average dose rate at that location. The gamma dose rate on the surface of the rib in the 125D is approximately 3 times lower than the circumferential average dose rate at that location. At one meter from the cask surface there is little difference between the dose rates calculated over the fins and the pocket trunnions compared to the other areas of the water jackets.

These conclusions indicate that localized neutron streaming is noticeable on the surface of the transfer casks. However, at one meter from the surface the streaming has dissipated. Since most HI-TRAC operations will involve personnel moving around the transfer cask at some distance from the cask only surface average dose rates are reported in this chapter.

Below each lifting trunnion, there is a localized area where the water jacket has been reduced in height by 4.125 inches to accommodate the lift yoke (see Figures 5.3.12 and 5.3.13). This area experiences a significantly higher than average dose rate on contact of the HI-TRAC. The peak dose in this location is ~~1.5~~*2.6* Rem/hr for the MPC-32, ~~1.4~~*1.9* Rem/hr for the MPC-68 and ~~1.3~~*2.4* Rem/hr for the MPC-24 in the 100-ton HI-TRAC and ~~649~~ *1.7* ~~mrem~~*Rem*/hr for the MPC-24 in the HI-TRAC 125D. At a distance of 1 to 2 feet from the edge of the HI-TRAC the localized effect is greatly reduced. This dose rate is acceptable because during lifting operations the lift yoke will be in place, which, due to the additional lift yoke steel (~3 inches), will greatly reduce the dose rate. However, more importantly, people will be prohibited from being in the vicinity of the lifting trunnions during lifting operations as a standard rigging practice. In addition the lift yoke is remote in its attachment and detachment, further minimizing personnel exposure. Immediately following the detachment of the lift yoke, in preparation for closure operations, temporary shielding may be placed in this area. Any temporary shielding (e.g., lead bricks, water tanks, lead blankets, steel plates, etc.) is sufficient to attenuate the localized hot spot. The operating procedure in Chapter 8 discusses the placement of temporary shielding in this area. For the 100-ton HI-TRAC, the optional temporary shield ring will replace the water that was lost from the axial reduction in the water jacket thereby eliminating the localized hot spot. When the HI-TRAC is in the horizontal position, during transport operations, it will (at a minimum) be positioned a few feet off the ground by the transport vehicle and therefore this location below the lifting trunnions will be positioned above people which will minimize the effect on personnel

exposure. In addition, good operating practice will dictate that personnel remain at least a few feet away from the transport vehicle. During vertical transport of a loaded HI-TRAC, the localized hot spot will be even further from the operating personnel. Based on these considerations, the conclusion is that this localized hot spot does not significantly impact the personnel exposure.

5.4.2 Damaged Fuel Post-Accident Shielding Evaluation

5.4.2.1 Dresden 1 and Humboldt Bay Damaged Fuel

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 of Chapter 1, 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.

Dividing the total fuel gamma source for a 6x6 fuel assembly in Table 5.2.7 by the 80 inch rubble height provides a gamma source per inch of 3.41E+12 photon/s. Dividing the total neutron source for a 6x6 fuel assembly in Table 5.2.18 by 80 inches provides a neutron source per inch of 2.75E+05 neutron/s. These values are both bounded by the BWR design basis fuel gamma source per inch and neutron source per inch values of 1.08E+13 photon/s and 9.17E+05 neutron/s, respectively, *for a burnup and cooling time of 40,000 MWD/MTU and 5 years*. These BWR design basis values were calculated by dividing the total source strengths for 40,000 MWD/MTU and 5 year cooling ~~in Tables 5.2.6 and 5.2.17~~ by the active fuel length of 144 inches. Therefore, damaged Dresden 1 and Humboldt Bay fuel assemblies are bounded by the design basis intact BWR fuel assembly for accident conditions. No explicit analysis of the damaged fuel dose rates from Dresden 1 or Humboldt Bay fuel assemblies are provided as they are bounded by the intact fuel analysis.

5.4.2.2 Generic PWR and BWR Damaged Fuel

The Holtec Generic PWR and BWR DFCs are designed to accommodate any PWR or BWR fuel assembly that can physically fit inside the DFC. Damaged fuel assemblies under normal conditions, for the most part, resemble intact fuel assemblies from a shielding perspective. Under accident conditions, it can not be guaranteed that the damaged fuel assembly will remain intact.

As a result, the damaged fuel assembly may begin to resemble fuel debris in its possible configuration after an accident.

Since damaged fuel is identical to intact fuel from a shielding perspective no specific analysis is required for damaged fuel under normal conditions. However, a generic shielding evaluation was performed to demonstrate that fuel debris under normal or accident conditions, or damaged fuel in a post-accident configuration, will not result in a significant increase in the dose rates around the 100-ton HI-TRAC. Only the 100-ton HI-TRAC was analyzed because it can be concluded that if the dose rate change is not significant for the 100-ton HI-TRAC then the change will not be significant for the 125-ton HI-TRACs or the HI-STORM overpacks.

Fuel debris or a damaged fuel assembly which has collapsed can have an average fuel density which is higher than the fuel density for an intact fuel assembly. If the damaged fuel assembly were to fully or partially collapse, the fuel density in one portion of the assembly would increase and the density in the other portion of the assembly would decrease. This scenario was analyzed with MCNP-4A in a conservative bounding fashion to determine the potential change in dose rate as a result of fuel debris or a damaged fuel assembly collapse. The analysis consisted of modeling the fuel assemblies in the damaged fuel locations in the MPC-24 (4 peripheral locations in the MPC-24E or MPC-24EF) and the MPC-68 (16 peripheral locations) with a fuel density that was twice the normal fuel density and correspondingly increasing the source rate for these locations by a factor of two. A flat axial power distribution was used which is approximately representative of the source distribution if the top half of an assembly collapsed into the bottom half of the assembly. Increasing the fuel density over the entire fuel length, rather than in the top half or bottom half of the fuel assembly, is conservative and provides the dose rate change in both the top and bottom portion of the cask.

Tables 5.4.13 and 5.4.14 provide the results for the MPC-24 and MPC-68, respectively. Only the radial dose rates are provided since the axial dose rates will not be significantly affected because the damaged fuel assemblies are located on the periphery of the baskets. A comparison of these results to the results in Tables 5.1.7 and 5.4.9 indicate that the dose rates in the top and bottom portion of the 100-ton HI-TRAC increase by less than 20% while the dose rate in the center of the HI-TRAC actually decreases a little bit. The increase in the bottom and top is due to the assumed flat power distribution. The dose rates shown in Tables 5.4.13 and 5.4.14 were averaged over the circumference of the cask. Since almost all of the peripheral cells in the MPC-68 are filled with DFCs, an azimuthal variation would not be expected for the MPC-68. However, since there are only 4 DFCs in the MPC-24E, an azimuthal variation in dose due to the damaged fuel/fuel debris might be expected. Therefore, the dose rates were evaluated in four smaller regions, one outside each DFC, that encompass about 44% of the circumference. There was no significant change in the dose rate as a result of the localized dose calculation. These results indicate that the potential effect on the dose rate is not very significant for the storage of damaged fuel and/or fuel debris. This conclusion is further reinforced by the fact that the majority of the significantly damaged fuel assemblies in the spent fuel inventories are older assemblies from the earlier days of nuclear plant operations. Therefore, these assemblies will

have a considerably lower burnup and longer cooling times than the assemblies analyzed in this chapter.

The MPC-32 was not explicitly analyzed for damaged fuel or fuel debris in this chapter. However, based on the analysis described above for the MPC-24 and the MPC-68, it can be concluded that the shielding performance of the MPC-32 will not be significantly affected by the storage of damaged fuel.

5.4.3 Site Boundary Evaluation

NUREG-1536 [5.2.1] states that detailed calculations need not be presented since SAR 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 characteristics would be factored into the evaluation performed by the licensee.

As an example of the methodology, the dose from a single HI-STORM overpack loaded with an MPC-24 and various arrays of loaded HI-STORMs at distances equal to and greater than 100 meters were evaluated with MCNP. In the model, the casks were placed on an infinite slab of dirt to account for earth-shine effects. The atmosphere was represented by dry air at a uniform density corresponding to 20 degrees C. The height of air modeled was 700 meters. This is more than sufficient to properly account for skyshine effects. The models included either 500 or 1050 meters of air around the cask. Based on the behavior of the dose rate as a function of distance, 50 meters of air, beyond the detector locations, is sufficient to account for back-scattering. Therefore, the HI-STORM MCNP off-site dose models account for back scattering by including more than 50 meters of air beyond the detector locations for all cited dose rates. Since gamma back-scattering has an effect on the off-site dose, it is recommended that the site-specific evaluation under 10CFR72.212 include at least 50 to 100 meters of air, beyond the detector locations, in the calculational models.

The MCNP calculations of the off-site dose used a two-stage process. In the first stage a binary surface source file (MCNP terminology) containing particle track information was written for particles crossing the outer radial and top surfaces of the HI-STORM overpack. In the second stage of the calculation, this surface source file was used with the particle tracks originating on the outer edge of the overpack and the dose rate was calculated at the desired location (hundreds of meters away from the overpack). The results from this two-stage process are statistically the same as the results from a single calculation. However, the advantage of the two-stage process is that each stage can be optimized independently.

The annual dose, assuming 100% occupancy (8760 hours), at ~~200~~ *250* meters from one cask is presented in Table 5.4.6 for the design basis burnup and cooling time analyzed. This table indicates that the dose due to neutrons is ~~7~~ *2.5* % 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 transmissions and could lead to low estimates of the site boundary dose.

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

1. The annual dose from the radiation leaving the side of the HI-STORM 100 overpack was calculated at the distance desired. Dose value = A.

2. The annual dose from the radiation leaving the top of the HI-STORM 100 overpack was calculated at the distance desired. Dose value = B.

3. The annual dose from the radiation leaving the side of a HI-STORM 100 overpack, when it is behind another cask, was calculated at the distance desired. The casks have an assumed 15-foot pitch. Dose value = C.

The doses calculated in the steps above are listed in Table 5.4.7 for the bounding burnup and cooling time of ~~52~~*47*,500 MWD/MTU and ~~5~~*3*-year cooling. Using these values, the annual dose (at the center of the long side) from an arbitrary 2 by Z array of HI-STORM 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 ~~300~~*400* meters is presented.

1. The annual dose from the side of a single cask: Dose A = *4.38*
2. The annual dose from the top of a single cask: Dose B = *1.65E-2*
3. The annual dose from the side of a cask positioned behind another cask: Dose C = *0.88*

Using the formula shown above (Z=3), the total dose at ~~300~~*400* meters from a 2x3 array of HI-STORM overpacks is ~~19.11~~*15.88* mrem/year, assuming a 8760 hour occupancy.

An important point to notice here is that the dose from the side of the back row of casks is *approximately* 16 % 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 typical arrays of HI-STORM overpacks can be found in Section 5.1. While the off-site dose analyses were performed for typical arrays of casks containing design basis fuel, compliance with the requirements of 10CFR72.104(a) can only be demonstrated on a site-specific basis. Therefore, a site-specific evaluation of dose at the controlled area boundary

must be performed for each ISFSI in accordance with 10CFR72.212. The site-specific evaluation will consider the site-specific characteristics (such as exposure duration and the number of casks deployed), dose from other portions of the facility and the specifics of the fuel being stored (burnup and cooling time).

5.4.4 Stainless Steel Clad Fuel Evaluation

Table 5.4.8 presents the dose rates at the center of the HI-STORM 100 overpack, adjacent and at one meter distance, from the stainless steel clad fuel. These dose rates, when compared to Tables 5.1.1 through 5.1.6, are similar to the dose rates from the design basis zircaloy clad fuel, indicating that these fuel assemblies are acceptable for storage.

As described in Section 5.2.3, it would be incorrect to compare the total source strength from the stainless steel clad fuel assemblies to the source strength from the design basis zircaloy clad fuel assemblies since these assemblies do not have the same active fuel length and since there is a significant gamma source from Cobalt-60 activation in the stainless steel. Therefore it is necessary to calculate the dose rates from the stainless steel clad fuel and compare them to the dose rates from the zircaloy clad fuel. In calculating the dose rates, the source term for the stainless steel fuel was calculated with an artificial active fuel length of 144 inches to permit a simple comparison of dose rates from stainless steel clad fuel and zircaloy clad fuel at the center of the HI-STORM 100 overpack. Since the true active fuel length is shorter than 144 inches and since the end fitting masses of the stainless steel clad fuel are assumed to be identical to the end fitting masses of the zircaloy clad fuel, the dose rates at the other locations on the overpack are bounded by the dose rates from the design basis zircaloy clad fuel, and therefore, no additional dose rates are presented.

5.4.5 Mixed Oxide Fuel Evaluation

The source terms calculated for the Dresden 1 GE 6x6 MOX fuel assemblies can be compared to the source terms for the BWR design basis zircaloy clad fuel assembly (GE 7x7) 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 5.2.22 by the 110 inch active fuel height provides a gamma source per inch of 2.36E+12 photons/s. Dividing the total neutron source for the MOX fuel assemblies in Table 5.2.23 by 110 inches 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 1.08E+13 photons/s and 9.17E+5 neutrons/s *for 40,000 MWD/MTU and 5 year cooling.* These BWR design basis values were calculated by dividing the total source strengths for 40,000 MWD/MTU and 5 year cooling ~~in Tables 5.2.6 and 5.2.17~~ by the active fuel length of 144 inches. This comparison shows that the MOX fuel source

terms are bound by the design basis source terms. Therefore, no explicit analysis of dose rates is provided for MOX fuel.

Since the MOX fuel assemblies are Dresden Unit 1 6x6 assemblies, they can also be considered as damaged fuel. Using the same methodology as described in Section 5.4.2.1, the source term for the MOX fuel is calculated on a per inch basis assuming a post accident rubble height of 80 inches. The resulting gamma and neutron source strengths are 3.25E+12 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.

5.4.6 Non-Fuel Hardware

As discussed in Section 5.2.4, non-fuel hardware in the form of BPRAs, TPDs, CRAs, and APSRs are permitted for storage, integral with a PWR fuel assembly, in the HI-STORM 100 System. Since each device occupies the same location within an assembly, only one device will be present in a given assembly. BPRAs and TPDs are authorized for unrestricted storage in an MPC while the CRAs and APSRs are restricted to the center four locations in the MPC-24, MPC-24E, MPC-24EF and MPC-32. The calculation of the source term and a description of the bounding fuel devices was provided in Section 5.2.4. The dose rate due to BPRAs and TPDs being stored in a fuel assembly was explicitly calculated. Table 5.4.15 provides the dose rates at various locations on the surface and one meter from the 100-ton HI-TRAC due to the BPRAs and TPDs for the MPC-24 and MPC-32. These results were added to the totals in the other table to provide the total dose rate with BPRAs. Table 5.4.15 indicates that the dose rates from BPRAs bound the dose rates from TPDs.

As discussed in Section 5.2.4, two different configurations were analyzed for CRAs and three different configurations were analyzed for APSRs. The dose rate due to CRAs and APSRs being stored in the inner four fuel locations was explicitly calculated for dose locations around the 100-ton HI-TRAC. Tables 5.4.16 and 5.4.17 provide the results for the different configurations of CRAs and APSRs, respectively, in the MPC-24 and MPC-32. These results indicate the dose rate on the radial surfaces of the overpack due to the storage of these devices is minimal and the dose rate out the top of the overpack is essentially 0. The latter is due to the fact that CRAs and APSRs do not achieve significant activation in the upper portion of the devices due to the manner in which they are utilized during normal reactor operations. In contrast, the dose rate out the bottom of the overpack is substantial due to these devices. However, as noted in Tables 5.4.16 and 5.4.17, the dose rate at the edge of the transfer lid is almost negligible due to APSRs and CRAs. Therefore, even though the dose rates calculated (using a very conservative source term evaluation) are daunting, they do not pose a risk from an operations perspective because they are localized in nature. Section 5.1.1 provides additional discussion on the acceptability of the relatively high localized doses on the bottom of the HI-TRACs.

5.4.7 Dresden Unit 1 Antimony-Beryllium Neutron Sources

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

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

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

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

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

Since this is a localized source (77.25 inches in length) it is appropriate to compare the neutron source per inch from the design basis Dresden Unit 1 fuel assembly, 6x6, containing an Sb-Be neutron source to the design basis fuel neutron source per inch. This comparison, presented in Table 5.4.18, 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 5.2.37 and 5.2.7 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 5.2.18, bounds the thoria rod neutron spectra, Table 5.2.38, 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 100-ton HI-TRAC and the HI-STORM 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 for the 100-ton HI-TRAC and only 15% lower for the HI-STORM overpack 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 Regionalized Loading Dose Rate Evaluation

Dose rates were calculated for regionalized loading patterns for the MPC-24, MPC-32, and MPC-68 using MCNP-4A. *Burnup and cooling time combinations bounding the 14x14A and 9x9G array classes were used in the analysis since for uniform loading these array classes have the highest permissible burnup for a given cooling time. Section 2.1.9 permits the user to calculate the burnup and cooling times for regions 1 and 2 after determining the region 1 and 2 allowable heat loads. Since the region 1 and 2 burnups are variable, it is impossible to analyze all regionalized loading burnup and cooling time combinations. Therefore, regionalized burnup and cooling time combinations corresponding to X=3 and X=6 (see Section 2.1.9.1.2) were analyzed.* ~~All burnup and cooling time combinations in Appendix B to the CoC were analyzed for both uniform and regionalized loading.~~ The dose rates for all dose locations reported in this chapter were compared for the uniform loading patterns and the regionalized loading patterns.

It was determined that for the MPC-32, all radial ~~surface and~~ 1 meter dose rates for regionalized loading were bounded by the uniform loading dose rates reported in this chapter. The maximum calculated ~~surface~~ dose rates in the axial locations for regionalized loading were less than ~~15~~*10%* higher than the uniform dose rates reported in this chapter ~~for the surface~~*at 1 meter from* ~~of~~ the overpack. ~~At one meter from the overpack, dose location 4 (in the center) was the only dose location which produced a slightly higher (5%) dose rate for regionalized loading compared to uniform loading.~~

For the MPC-24 *and MPC-68* it was determined that *all 1 meter dose rates for regionalized loading were bounded by the uniform loading dose rates reported in this chapter.* ~~the maximum calculated dose rates in the axial direction for regionalized loading were less than 21% higher than the maximum calculated dose rates for uniform loading reported in this chapter. At one meter distance, the uniform loading dose rates reported in this chapter bound the regionalized loading dose rates. In the radial direction, the uniform loading dose rates reported in this chapter bound the regionalized loading dose rates for both surface and one meter locations.~~

~~For the MPC-68 it was determined that all radial surface and 1 meter dose rates for regionalized loading were bounded by the uniform loading dose rates reported in this chapter. The maximum calculated surface dose rates in the axial locations for regionalized loading were less than 21% higher than the uniform dose rates reported in this chapter for the surface of the overpack. At one meter from the overpack, dose locations 4 (in the center) and 5 (transfer lid center) were the only dose locations which produced a slightly higher (5% and 1.5% respectively) dose rate for regionalized loading compared to uniform loading.~~

Based on these results it can be stated that regionalized loading patterns will reduce the dose rate in the radial direction by shielding the hotter fuel on the inside of the cask with colder fuel on the outside of the cask. However, in the axial direction the localized dose rates in the center of the cask may increase as a result of the regionalized loading pattern. This is a localized effect, which has dissipated at the edge of the cask, and therefore will not result in a significant increase to the

occupational exposure rates. In addition, it should be mentioned that the localized increase on the bottom center of the overpack is an area where workers will normally not be present and the increase in the top center of the overpack is an area where workers minimize their stay.

Table 5.4.1

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

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

Table 5.4.1 (continued)

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

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

Table 5.4.1 (continued)

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

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

† Includes the Quality Factor.

Table 5.4.2

DOSE RATES FOR THE 100-TON HI-TRAC FOR THE FULLY FLOODED MPC CONDITION WITH AN EMPTY NEUTRON SHIELD MPC-24 DESIGN BASIS ZIRCALOY CLAD FUEL AT ~~42,500~~ *46,000* MWD/MTU AND ~~5~~*3*-YEAR COOLING

Dose Point† Location	Fuel Gammas†† (mrem/hr)	^{60}Co Gammas (mrem/hr)	Neutrons (mrem/hr)	Totals (mrem/hr)	Totals with BPRAs (mrem/hr)
ADJACENT TO THE 100-TON HI-TRAC					
1	*34.70*	*282.04*	*24.34*	*341.07*	*343.69*
2	*2212.18*	*0.66*	*423.66*	*2636.50*	*2839.98*
3	*8.60*	*429.18*	*5.92*	*443.70*	*575.29*
4	*31.22*	*326.11*	*0.98*	*358.31*	*460.57*
5 (pool lid)	*111.45*	*1835.89*	*3.33*	*1950.68*†††	*1960.87*
ONE METER FROM THE 100-TON HI-TRAC					
1	*294.71*	*63.68*	*60.22*	*418.60*	*445.03*
2	*979.53*	*6.23*	*139.30*	*1125.06*	*1215.29*
3	*117.27*	*104.40*	*24.91*	*246.57*	*290.89*

Note: MPC internal water level is 10 inches below the MPC lid.

† Refer to Figures 5.1.2 and 5.1.4.

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

††† Cited dose rates correspond to the cask center. Figures 5.1.6, 5.1.7, and 5.1.11 illustrate the substantial reduction in dose rates moving radially outward from the axial center of the HI-TRAC.

Table 5.4.3

DOSE RATES FOR THE 100-TON HI-TRAC FOR THE FULLY FLOODED MPC CONDITION WITH A FULL NEUTRON SHIELD MPC-24 DESIGN BASIS ZIRCALOY CLAD FUEL AT ~~42,500~~*46,000* MWD/MTU AND ~~5~~*3*-YEAR COOLING

Dose Point† Location	Fuel Gammas†† (mrem/hr)	^{60}Co Gammas (mrem/hr)	Neutrons (mrem/hr)	Totals (mrem/hr)	Totals with BPRAs (mrem/hr)
ADJACENT TO THE 100-TON HI-TRAC					
1	*29.73*	*282.19*	*3.17*	*315.10*	*317.20*
2	*1313.86*	*0.44*	*27.67*	*1341.97*	*1457.53*
3	*5.61*	*428.14*	*0.56*	*434.31*	*565.24*
4	*31.19*	*326.10*	*1.00*	*358.30*	*460.55*
5 (pool lid)	*111.11*	*1836.07*	*2.82*	*1950.01*†††	*1960.18*
ONE METER FROM THE 100-TON HI-TRAC					
1	*170.07*	*43.84*	*3.70*	*217.61*	*232.40*
2	*573.05*	*3.49*	*10.41*	*586.95*	*637.50*
3	*67.61*	*72.02*	*1.26*	*140.89*	*169.77*

Note: MPC internal water level is 10 inches below the MPC lid.

† Refer to Figures 5.1.2 and 5.1.4.

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

††† Cited dose rates correspond to the cask center. Figures 5.1.6, 5.1.7, and 5.1.11 illustrate the substantial reduction in dose rates moving radially outward from the axial center of the HI-TRAC.

Table 5.4.4

DOSE RATES FROM THE 100-TON HI-TRAC FOR NORMAL CONDITIONS
MPC-24 DESIGN BASIS ZIRCALOY CLAD FUEL AT
~~52,500~~75,000 MWD/MTU AND ~~10~~5-YEAR COOLING

Dose Point Location	Fuel Gammas (mrem/hr)	(n,γ) Gammas (mrem/hr)	^{60}Co Gammas (mrem/hr)	Neutrons (mrem/hr)	Totals (mrem/hr)	Totals with BPRAs (mrem/hr)
ADJACENT TO THE 100-TON HI-TRAC						
1	*61.51*	*59.98*	*841.88*	*848.87*	*1812.25*	*1820.79*
2	*1720.78*	*244.11*	*0.84*	*450.27*	*2416.00*	*2663.24*
3	*16.84*	*11.76*	*464.19*	*710.32*	*1203.11*	*1351.64*
3 (temp)	*7.62*	*20.93*	*215.15*	*11.42*	*255.11*	*323.26*
4	*41.62*	*4.64*	*373.59*	*874.50*	*1294.35*	*1418.85*
4 (outer)	*11.60*	*2.95*	*93.02*	*590.24*	*697.81*	*729.14*
5 (pool lid)	*298.84*	*85.64*	*4241.72*	*5701.99*	*10328.19*	*10392.96*
5 (transfer)	*732.62*	*4.69*	*6320.81*	*3264.99*	*10323.11*	*10419.95*
5(t-outer)	*178.17*	*1.60*	*611.80*	*1290.11*	*2081.69*	*2103.16*
ONE METER FROM THE 100-TON HI-TRAC						
1	*226.79*	*32.24*	*125.15*	*137.98*	*522.16*	*554.64*
2	*754.47*	*74.62*	*9.90*	*168.82*	*1007.81*	*1117.29*
3	*94.60*	*17.96*	*103.96*	*66.26*	*282.78*	*332.22*
3 (temp)	*94.09*	*19.29*	*88.55*	*25.04*	*226.97*	*271.54*
4	*14.19*	*0.81*	*115.34*	*217.83*	*348.17*	*386.74*
5 (transfer)	*315.47*	*0.86*	*2582.07*	*911.26*	*3809.66*	*3848.79*
5(t-outer)	*42.95*	*2.78*	*232.74*	*261.61*	*540.08*	*543.98*

Notes:

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

Table 5.4.5

DOSE RATES FROM THE 125-TON HI-TRAC FOR NORMAL CONDITIONS
MPC-24 DESIGN BASIS ZIRCALOY CLAD FUEL AT
~~42,500~~46,000 MWD/MTU AND ~~5~~3-YEAR COOLING

Dose Point Location	Fuel Gammas (mrem/hr)	(n,γ) Gammas (mrem/hr)	^{60}Co Gammas (mrem/hr)	Neutrons (mrem/hr)	Totals (mrem/hr)	Totals with BPRAs (mrem/hr)
ADJACENT TO THE 125-TON HI-TRAC						
1	*12.43*	*17.84*	*101.50*	*119.96*	*251.73*	*252.44*
2	*211.88*	*52.83*	*0.01*	*83.09*	*347.81*	*363.68*
3	*2.66*	*1.89*	*62.80*	*191.39*	*258.73*	*278.44*
4	*71.16*	*2.42*	*343.61*	*221.50*	*638.70*	*754.12*
4 (outer)	*9.28*	*1.73*	*42.67*	*4.65*	*58.33*	*72.52*
5 (pool)	*108.22*	*0.90*	*529.32*	*766.89*	*1405.34*	*1413.04*
5 (transfer)	*112.43*	*1.38*	*606.59*	*127.01*	*847.40*	*852.89*
ONE METER FROM THE 125-TON HI-TRAC						
1	*28.53*	*7.12*	*13.01*	*19.74*	*68.40*	*70.43*
2	*95.52*	*17.13*	*0.53*	*28.34*	*141.51*	*148.58*
3	*10.94*	*4.02*	*12.69*	*17.61*	*45.26*	*50.18*
4	*20.01*	*0.58*	*82.73*	*22.81*	*126.13*	*153.78*
5 (transfer)	*41.40*	*0.27*	*293.26*	*22.00*	*356.92*	*359.85*

Notes:

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

Table 5.4.6

ANNUAL DOSE AT ~~200~~*250* METERS FROM A SINGLE HI-STORM OVERPACK WITH AN MPC-24 WITH DESIGN BASIS ZIRCALOY CLAD FUEL†

Dose Component	~~52~~*47*,500 MWD/MTU ~~5~~*3*-Year Cooling (mrem/yr)
Fuel gammas††	~~16.52~~*21.02*
^{60}Co Gammas	~~2.17~~*1.24*
Neutrons	~~1.50~~*0.57*
Total	~~20.19~~*22.83*

† 8760 hour annual occupancy is assumed.

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

Table 5.4.7

DOSE VALUES USED IN CALCULATING ANNUAL DOSE FROM VARIOUS ISFSI CONFIGURATIONS ~~52~~47,500 MWD/MTU AND ~~5~~3-YEAR COOLING ZIRCALOY CLAD FUEL†

Distance	A Side of Overpack (mrem/yr)	B Top of Overpack (mrem/yr)	C Side of Shielded Overpack (mrem/yr)
100 meters	*306.6*	*1.44*	*61.32*
150 meters	*109.2*	*0.56*	*21.84*
200 meters	*48.4*	*0.25*	*9.68*
250 meters	*24.0*	*0.11*	*4.80*
300 meters	*12.9*	*6.09E-2*	*2.58*
350 meters	*7.48*	*3.12E-2*	*1.50*
400 meters	*4.38*	*1.65E-2*	*0.88*

† 8760 hour annual occupancy is assumed.

Table 5.4.8

DOSE RATES AT THE CENTERLINE OF THE OVERPACK FOR DESIGN BASIS STAINLESS STEEL CLAD FUEL WITHOUT BPRAs

Dose Point† Location	Fuel Gammas†† (mrem/hr)	^{60}Co Gammas (mrem/hr)	Neutrons (mrem/hr)	Totals (mrem/hr)
MPC-24 (40,000 MWD/MTU AND 8-YEAR COOLING)				
2 (Adjacent)	36.97	0.02	1.11	38.10
2 (One Meter)	18.76	0.17	0.50	19.43
MPC-32 (40,000 MWD/MTU AND 9-YEAR COOLING)				
2 (Adjacent)	37.58	0.00	1.49	39.08
2 (One Meter)	18.74	0.25	0.58	19.57
MPC-68 (22,500 MWD/MTU AND 10-YEAR COOLING)				
2 (Adjacent)	17.79	0.01	0.10	17.90
2 (One Meter)	8.98	0.13	0.04	9.15

† Refer to Figure 5.1.1.

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

Table 5.4.9

DOSE RATES FROM THE 100-TON HI-TRAC FOR NORMAL CONDITIONS
MPC-68 DESIGN BASIS ZIRCALOY CLAD FUEL AT
4~~0~~39,000 MWD/MTU AND ~~5~~3-YEAR COOLING

Dose Point Location	Fuel Gammas (mrem/hr)	(n,γ) Gammas (mrem/hr)	^{60}Co Gammas (mrem/hr)	Neutrons (mrem/hr)	Totals (mrem/hr)
ADJACENT TO THE 100-TON HI-TRAC					
1	*101.95*	*13.60*	*1148.49*	*181.72*	*1445.78*
2	*2235.38*	*67.10*	*0.73*	*121.56*	*2424.77*
3	*6.47*	*1.42*	*695.08*	*77.00*	*779.97*
3 (temp)	*3.74*	*2.27*	*330.06*	*1.42*	*337.49*
4	*14.60*	*0.60*	*273.56*	*100.40*	*389.15*
4 (outer)	*4.04*	*0.40*	*72.46*	*60.14*	*137.03*
5 (pool lid)	*302.21*	*16.66*	*5142.93*	*1089.34*	*6551.14*
5(transfer lid)	*424.40*	*0.78*	*7748.87*	*686.26*	*8860.30*
5 (t-outer)	*157.68*	*0.33*	*683.21*	*256.32*	*1097.53*
ONE METER FROM THE 100-TON HI-TRAC					
1	*304.87*	*8.25*	*107.26*	*32.11*	*452.50*
2	*962.27*	*18.99*	*7.91*	*41.90*	*1031.07*
3	*75.11*	*3.26*	*157.28*	*8.89*	*244.54*
3 (temp)	*75.04*	*3.41*	*127.40*	*4.27*	*210.11*
4	*5.39*	*0.11*	*91.29*	*20.93*	*117.72*
5(transfer lid)	*218.21*	*0.33*	*3437.93*	*183.88*	*3840.35*
5 (t-outer)	*27.48*	*0.59*	*290.36*	*51.99*	*370.41*

Notes:

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

Table 5.4.10

DOSE RATES FROM THE 100-TON HI-TRAC FOR NORMAL CONDITIONS
MPC-68 DESIGN BASIS ZIRCALOY CLAD FUEL AT
~~50~~70,000 MWD/MTU AND ~~10~~6-YEAR COOLING

Dose Point Location	Fuel Gammas (mrem/hr)	(n,γ) Gammas (mrem/hr)	^{60}Co Gammas (mrem/hr)	Neutrons (mrem/hr)	Totals (mrem/hr)
ADJACENT TO THE 100-TON HI-TRAC					
1	*46.14*	*51.49*	*957.08*	*687.54*	*1742.24*
2	*1122.53*	*253.96*	*0.61*	*459.77*	*1836.86*
3	*2.37*	*5.36*	*579.24*	*291.33*	*878.29*
3 (temp)	*1.52*	*8.60*	*275.05*	*5.39*	*290.56*
4	*5.50*	*2.26*	*227.96*	*379.71*	*615.44*
4 (outer)	*1.59*	*1.52*	*60.38*	*227.53*	*291.02*
5 (pool lid)	*138.45*	*63.04*	*4285.78*	*4121.87*	*8609.13*
5(transfer lid)	*239.62*	*2.94*	*6457.39*	*2597.01*	*9296.96*
5 (t-outer)	*81.20*	*1.24*	*569.34*	*969.87*	*1621.64*
ONE METER FROM THE 100-TON HI-TRAC					
1	*151.92*	*31.24*	*89.38*	*121.49*	*394.03*
2	*480.13*	*71.90*	*6.59*	*158.45*	*717.07*
3	*37.29*	*12.34*	*131.07*	*33.63*	*214.32*
3 (temp)	*37.24*	*12.90*	*106.17*	*16.14*	*172.44*
4	*2.35*	*0.43*	*76.08*	*79.17*	*158.02*
5(transfer lid)	*116.13*	*1.25*	*2864.94*	*695.86*	*3678.18*
5 (t-outer)	*14.24*	*2.22*	*241.96*	*196.72*	*455.15*

Notes:

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

Table 5.4.11

DOSE RATES FROM THE 100-TON HI-TRAC FOR NORMAL CONDITIONS
MPC-32 DESIGN BASIS ZIRCALOY CLAD FUEL
~~32,500~~*35,000* MWD/MTU AND ~~5~~*3*-YEAR COOLING

Dose Point Location	Fuel Gammas (mrem/hr)	(n,γ) Gammas (mrem/hr)	^{60}Co Gammas (mrem/hr)	Neutrons (mrem/hr)	Totals (mrem/hr)	Totals with BPRAs (mrem/hr)
ADJACENT TO THE 100-TON HI-TRAC						
1	*101.70*	*7.68*	*943.95*	*111.78*	*1165.11*	*1175.39*
2	*2499.72*	*34.07*	*1.35*	*63.94*	*2599.08*	*2890.62*
3	*35.51*	*1.50*	*595.93*	*88.06*	*721.00*	*946.40*
4	*84.64*	*0.99*	*446.65*	*110.31*	*642.58*	*821.55*
4 (outer)	*23.54*	*0.39*	*112.05*	*75.06*	*211.03*	*256.03*
5 (pool)	*604.32*	*10.75*	*5208.45*	*727.74*	*6551.26*	*6633.59*
5 (transfer)	*1056.27*	*0.42*	*7860.47*	*408.03*	*9325.19*	*9424.04*
5(t-outer)	*199.21*	*0.23*	*662.48*	*162.60*	*1024.51*	*1044.60*
ONE METER FROM THE 100-TON HI-TRAC						
1	*334.53*	*4.51*	*140.68*	*18.47*	*498.20*	*536.18*
2	*1107.24*	*10.70*	*10.72*	*23.33*	*1151.99*	*1281.43*
3	*144.93*	*2.54*	*122.47*	*8.79*	*278.73*	*345.55*
4	*26.50*	*0.14*	*133.19*	*27.24*	*187.07*	*240.23*
5 (transfer)	*447.93*	*0.14*	*3124.65*	*113.90*	*3686.60*	*3730.23*
5(t-outer)	*46.85*	*0.43*	*277.06*	*33.17*	*357.51*	*361.97*

Notes:

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

Table 5.4.12

DOSE RATES FROM THE 100-TON HI-TRAC FOR NORMAL CONDITIONS
MPC-32 DESIGN BASIS ZIRCALOY CLAD FUEL
~~45~~75,000 MWD/MTU AND ~~10~~8-YEAR COOLING

Dose Point Location	Fuel Gammas (mrem/hr)	(n,γ) Gammas (mrem/hr)	^{60}Co Gammas (mrem/hr)	Neutrons (mrem/hr)	Totals (mrem/hr)	Totals with BPRAs (mrem/hr)
ADJACENT TO THE 100-TON HI-TRAC						
1	*35.36*	*58.40*	*756.90*	*849.12*	*1699.78*	*1710.06*
2	*1008.77*	*258.96*	*1.09*	*485.41*	*1754.23*	*2045.77*
3	*10.68*	*11.39*	*477.85*	*669.04*	*1168.96*	*1394.36*
4	*29.28*	*7.51*	*358.14*	*837.70*	*1232.64*	*1411.61*
4 (outer)	*7.45*	*2.97*	*89.84*	*570.22*	*670.48*	*715.48*
5 (pool)	*239.46*	*81.70*	*4176.38*	*5528.60*	*10026.14*	*10108.46*
5 (transfer)	*464.10*	*3.16*	*6302.90*	*3101.39*	*9871.56*	*9970.41*
5(t-outer)	*82.86*	*1.72*	*531.20*	*1235.27*	*1851.06*	*1871.15*
ONE METER FROM THE 100-TON HI-TRAC						
1	*130.79*	*34.31*	*112.81*	*140.25*	*418.16*	*456.14*
2	*441.14*	*81.34*	*8.60*	*177.10*	*708.18*	*837.61*
3	*56.21*	*19.30*	*98.20*	*66.75*	*240.47*	*307.29*
4	*8.34*	*1.07*	*106.80*	*206.96*	*323.17*	*376.33*
5 (transfer)	*186.92*	*1.03*	*2505.49*	*865.27*	*3558.70*	*3602.33*
5(t-outer)	*18.96*	*3.25*	*222.16*	*251.98*	*496.36*	*500.83*

Notes:

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

Table 5.4.13

DOSE RATES FROM THE 100-TON HI-TRAC FOR ACCIDENT CONDITIONS
WITH FOUR DAMAGED FUEL CONTAINERS
MPC-24 DESIGN BASIS ZIRCALOY CLAD FUEL
~~42,500~~*46,000* MWD/MTU AND ~~5~~*3*-YEAR COOLING
WITHOUT BPRAs

Dose Point† Location	Fuel Gammas (mrem/hr)	(n,γ) Gammas (mrem/hr)	^{60}Co Gammas (mrem/hr)	Neutrons (mrem/hr)	Totals (mrem/hr)
ADJACENT TO THE 100-TON HI-TRAC					
1	*139.14*	*20.90*	*849.14*	*317.12*	*1326.29*
2	*2635.40*	*75.69*	*1.01*	*137.79*	*2849.89*
3	*40.59*	*4.69*	*468.20*	*304.41*	*817.88*
ONE METER FROM THE 100-TON HI-TRAC					
1	*376.67*	*10.74*	*126.22*	*48.19*	*561.82*
2	*1175.03*	*23.42*	*9.99*	*51.90*	*1260.33*
3	*169.41*	*6.13*	*104.86*	*28.32*	*308.72*

† Refer to Figures 5.1.2 and 5.1.4.

Table 5.4.14

DOSE RATES FROM THE 100-TON HI-TRAC FOR ACCIDENT CONDITIONS
WITH SIXTEEN DAMAGED FUEL CONTAINERS
MPC-68 DESIGN BASIS ZIRCALOY CLAD FUEL
~~40~~39,000 MWD/MTU AND ~~5~~3-YEAR COOLING

Dose Point† Location	Fuel Gammas (mrem/hr)	(n,γ) Gammas (mrem/hr)	^{60}Co Gammas (mrem/hr)	Neutrons (mrem/hr)	Totals (mrem/hr)
ADJACENT TO THE 100-TON HI-TRAC					
1	*237.63*	*19.51*	*1148.49*	*336.58*	*1742.21*
2	*2107.52*	*67.58*	*0.73*	*114.34*	*2290.16*
3	*8.12*	*2.86*	*695.08*	*170.31*	*876.38*
ONE METER FROM THE 100-TON HI-TRAC					
1	*353.82*	*10.21*	*107.26*	*48.47*	*519.76*
2	*923.33*	*20.45*	*9.30*	*46.37*	*999.45*
3	*103.30*	*4.54*	*157.28*	*17.67*	*282.78*

† Refer to Figures 5.1.2 and 5.1.4.

Table 5.4.15

DOSE RATES DUE TO BPRAs AND TPDs FROM THE 100-TON HI-TRAC FOR NORMAL CONDITIONS

	MPC-24		MPC-32	
Dose Point Location	**BPRAs (mrem/hr)**	**TPDs (mrem/hr)**	**BPRAs (mrem/hr)**	**TPDs (mrem/hr)**
ADJACENT TO THE 100-TON HI-TRAC				
1	*8.54*	0.00	*10.28*	0.01
2	*247.24*	0.03	*291.53*	0.04
3	*148.53*	125.75	*225.40*	188.04
3 (temp)	*68.15*	56.21	*93.60*	76.97
4	*124.50*	106.71	*178.97*	156.15
4 (outer)	*31.33*	27.12	*45.00*	39.32
5 (pool lid)	*64.77*	0.00	*82.32*	0.00
5(transfer lid)	*96.84*	0.00	*98.85*	0.00
5(t-outer)	*21.47*	0.00	*20.09*	0.00
ONE METER FROM THE 100-TON HI-TRAC				
1	*32.48*	0.18	*37.98*	0.23
2	*109.47*	1.20	*129.44*	1.62
3	*49.43*	38.93	*66.82*	54.93
3 (temp)	*44.57*	35.01	*59.10*	48.77
4	*38.57*	33.37	*53.16*	47.19
5(transfer lid)	*39.13*	0.00	*43.62*	0.00
5(t-outer)	*3.90*	0.00	*4.47*	0.00

Notes:

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

Table 5.4.16

DOSE RATES DUE TO CRAs FROM THE 100-TON HI-TRAC
FOR NORMAL CONDITIONS

	MPC-24		MPC-32	
Dose Point Location	Config. 1 (mrem/hr)	Config. 2 (mrem/hr)	Config. 1 (mrem/hr)	Config. 2 (mrem/hr)
ADJACENT TO THE 100-TON HI-TRAC				
1	5.39	1.02	3.28	0.68
2	0.09	0.00	0.01	0.00
3	0.00	0.00	0.00	0.00
4	0.00	0.00	0.00	0.00
5 (pool lid)	919.59	170.85	1141.10	213.24
5(transfer lid)	1519.98	287.72	2012.93	380.57
5(t-outer)	1.54	0.25	1.01	0.19
ONE METER FROM THE 100-TON HI-TRAC				
1	1.20	0.20	0.69	0.14
2	0.26	0.03	0.05	0.01
3	0.01	0.00	0.00	0.00
4	0.00	0.00	0.00	0.00
5(transfer lid)	223.62	41.60	257.95	49.19
5(t-outer)	8.26	1.54	8.87	1.70

Notes:

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

Table 5.4.17

DOSE RATES DUE TO APSRs FROM THE 100-TON HI-TRAC
FOR NORMAL CONDITIONS

	MPC-24			MPC-32		
Dose Point Location	Config. 1 (mrem/hr)	Config. 2 (mrem/hr)	Config. 3 (mrem/hr)	Config. 1 (mrem/hr)	Config. 2 (mrem/hr)	Config. 3 (mrem/hr)
ADJACENT TO THE 100-TON HI-TRAC						
1	12.42	2.35	12.25	7.57	1.56	7.51
2	0.21	0.01	9.12	0.03	0.00	0.19
3	0.00	0.00	0.00	0.00	0.00	0.00
4	0.00	0.00	0.00	0.00	0.00	0.00
5 (pool lid)	1996.57	371.98	1941.51	2414.84	453.88	2687.17
5(transfer)	3021.08	572.85	2994.54	3980.02	750.17	3860.83
5(t-outer)	3.41	0.54	3.57	2.23	0.42	1.94
ONE METER FROM THE 100-TON HI-TRAC						
1	2.73	0.46	3.49	1.57	0.32	1.58
2	0.61	0.07	3.31	0.12	0.02	0.18
3	0.02	0.00	0.04	0.01	0.00	0.01
4	0.00	0.00	0.00	0.00	0.00	0.00
5(transfer)	458.06	84.81	444.44	521.02	99.10	510.78
5(t-outer)	17.11	3.19	17.36	18.34	3.48	18.20

Notes:

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

Table 5.4.18

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

Assembly	Active fuel length (inch)	Neutrons per sec per inch	Neutrons per sec per inch with Sb-Be source	Reference for neutrons per sec per inch
7x7 design basis	144	9.17E+5	N/A	~~Table 5.2.17~~ 40 GWD/MTU and 5 year cooling
6x6 design basis	110	2.0E+5	2.6E+5	Table 5.2.18
6x6 design basis MOX	110	3.06E+5	3.66E+5	Table 5.2.23

5.6 REFERENCES

[5.1.1] J.F. Briesmeister, Ed., "MCNP - A General Monte Carlo N-Particle Transport Code, Version 4A." Los Alamos National Laboratory, LA-12625-M (1993).

[5.1.2] O.W. Hermann, C.V. Parks, "SAS2H: A Coupled One-Dimensional Depletion and Shielding Analysis Module," NUREG/CR-0200, Revision 5, (ORNL/NUREG/CSD-2/V2/R5), Oak Ridge National Laboratory, September 1995.

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

[5.2.1] NUREG-1536, SRP for Dry Cask Storage Systems, USNRC, Washington, DC, January 1997.

[5.2.2] A.G. Croff, M.A. Bjerke, G.W. Morrison, L.M. Petrie, "Revised Uranium-Plutonium Cycle PWR and BWR Models for the ORIGEN Computer Code," ORNL/TM-6051, Oak Ridge National Laboratory, September 1978.

[5.2.3] A. Luksic, "Spent Fuel Assembly Hardware: Characterization and 10CFR 61 Classification for Waste Disposal," PNL-6906-vol. 1, Pacific Northwest Laboratory, June 1989.

[5.2.4] J.W. Roddy et al., "Physical and Decay Characteristics of Commercial LWR Spent Fuel," ORNL/TM-9591/V1&R1, Oak Ridge National Laboratory, January 1996.

[5.2.5] "Characteristics of Spent Fuel, High Level Waste, and Other Radioactive Wastes Which May Require Long-Term Isolation," DOE/RW-0184, U.S. Department of Energy, December 1987.

[5.2.6] "Spent Nuclear Fuel Discharges from U.S. Reactors 1994," SR/CNEAF/96-01, Energy Information Administration, U.S. Department of Energy, February 1996.

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

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

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

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

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

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

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

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

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

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

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

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

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

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

[5.4.6] Commonwealth Edison Company, Letter No. NFS-BND-95-083, Chicago, Illinois.

CHAPTER 6†: CRITICALITY EVALUATION

This chapter documents the criticality evaluation of the HI-STORM 100 System for the storage of spent nuclear fuel in accordance with 10CFR72.124. The results of this evaluation demonstrate that the HI-STORM 100 System is consistent with the Standard Review Plan for Dry Cask Storage Systems, NUREG-1536, and thus, fulfills the following acceptance criteria:

1. The multiplication factor (k_{eff}), including all biases and uncertainties at a 95-percent confidence level, should not exceed 0.95 under all credible normal, off-normal, and accident conditions.

2. At least two unlikely, independent, and concurrent or sequential changes to the conditions essential to criticality safety, under normal, off-normal, and accident conditions, should occur before an accidental criticality is deemed to be possible.

3. When practicable, criticality safety of the design should be established on the basis of favorable geometry, permanent fixed neutron-absorbing materials (poisons), or both.

4. Criticality safety of the cask system should not rely on use of the following credits:

    a. burnup of the fuel
    b. fuel-related burnable neutron absorbers
    c. more than 75 percent for fixed neutron absorbers when subject to standard acceptance test††.

In addition to demonstrating that the criticality safety acceptance criteria are satisfied, this chapter describes the HI-STORM 100 System design structures and components important to criticality safety and defines the limiting fuel characteristics in sufficient detail to identify the package accurately and provide a sufficient basis for the evaluation of the package. Analyses for the HI-STAR 100 System, which are applicable to the HI-STORM 100 System, have been previously submitted to the USNRC under Docket Numbers 72-1008 and 71-9261.

---

† This chapter has been prepared in the format and section organization set forth in Regulatory Guide 3.61. However, the material content of this chapter also fulfills the requirements of NUREG-1536. Pagination and numbering of sections, figures, and tables are consistent with the convention set down in *Chapter 1*, Section 1.0, herein. Finally, all terms-of-art used in this chapter are consistent with the terminology of the glossary (Table 1.0.1) and component nomenclature of the Bill-of-Materials (Section 1.5).

†† *For greater credit allowance, fabrication tests capable of verifying the presence and uniformity of the neutron absorber are needed.*

6.1 DISCUSSION AND RESULTS

In conformance with the principles established in NUREG-1536 [6.1.1], 10CFR72.124 [6.1.2], and NUREG-0800 Section 9.1.2 [6.1.3], the results in this chapter demonstrate that the effective multiplication factor (k_{eff}) of the HI-STORM 100 System, including all biases and uncertainties evaluated with a 95% probability at the 95% confidence level, does not exceed 0.95 under all credible normal, off-normal, and accident conditions. Moreover, these results demonstrate that the HI-STORM 100 System is designed and maintained such that at least two unlikely, independent, and concurrent or sequential changes must occur to the conditions essential to criticality safety before a nuclear criticality accident is possible. These criteria provide a large subcritical margin, sufficient to assure the criticality safety of the HI-STORM 100 System when fully loaded with fuel of the highest permissible reactivity.

Criticality safety of the HI-STORM 100 System depends on the following four principal design parameters:

1. The inherent geometry of the fuel basket designs within the MPC (and the flux-trap water gaps in the MPC-24, MPC-24E and MPC-24EF);

2. The incorporation of permanent fixed neutron-absorbing panels ~~(Boral)~~ in the fuel basket structure;

3. An administrative limit on the maximum enrichment for PWR fuel and maximum planar-average enrichment for BWR fuel; and

4. An administrative limit on the minimum soluble boron concentration in the water for loading/unloading fuel with higher enrichments in the MPC-24, MPC-24E and MPC-24EF, and for loading/unloading fuel in the MPC-32 *and MPC-32F.*

The off-normal and accident conditions defined in Chapter 2 and considered in Chapter 11 have no adverse effect on the design parameters important to criticality safety, and thus, the off-normal and accident conditions are identical to those for normal conditions.

The HI-STORM 100 System is designed such that the fixed neutron absorber ~~(Boral)~~ will remain effective for a storage period greater than 20 years, and there are no credible means to lose it. Therefore, in accordance with 10CFR72.124(b), there is no need to provide a surveillance or monitoring program to verify the continued efficacy of the neutron absorber.

Criticality safety of the HI-STORM 100 System does not rely on the use of any of the following credits:

- burnup of fuel
- fuel-related burnable neutron absorbers
- more than 75 percent of the B-10 content for the *Boral* fixed neutron absorber ~~(Boral)~~
- *more than 90 percent of the B-10 content for the Metamic fixed neutron absorber, with comprehensive fabrication tests as described in Section 9.1.5.3.2.*

The following four interchangeable basket designs are available for use in the HI-STORM 100 System:

- a 24-cell basket (MPC-24), designed for intact PWR fuel assemblies with a specified maximum enrichment and, for higher enrichments, a minimum soluble boron concentration in the pool water for loading/unloading operations,
- a 24-cell basket (MPC-24E) for intact and damaged PWR fuel assemblies. This is a variation of the MPC-24, with an optimized cell arrangement, increased ^{10}B content in the ~~Boral~~*fixed neutron absorber* and with four cells capable of accommodating either intact fuel or a damaged fuel container (DFC). Additionally, a variation in the MPC-24E, designated MPC-24EF, is designed for intact and damaged PWR fuel assemblies and PWR fuel debris. The MPC-24E and MPC-24EF are designed for fuel assemblies with a specified maximum enrichment and, for higher enrichments, a minimum soluble boron concentration in the pool water for loading/unloading operations,
- a 32-cell basket (MPC-32), designed for intact *and damaged* PWR fuel assemblies of a specified maximum enrichment and minimum soluble boron concentration for loading/unloading. *Additionally, a variation in the MPC-32, designated MPC-32F, is designed for intact and damaged PWR fuel assemblies and PWR fuel debris.*~~, a~~*A*nd
- a 68-cell basket (MPC-68), designed for both intact and damaged BWR fuel assemblies with a specified maximum planar-average enrichment. Additionally, variations in the MPC-68, designated MPC-68F and MPC-68FF, are designed for intact and damaged BWR fuel assemblies and BWR fuel debris with a specified maximum planar-average enrichment.

Two interchangeable neutron absorber materials are used in these baskets, Boral and Metamic. For Boral, 75 percent of the minimum B-10 content is credited in the criticality analysis, while for Metamic, 90 percent of the minimum B-10 content is credited, based on the neutron absorber

tests specified in Section 9.1.5.3. However, the B-10 content in Metamic is chosen to be lower than the B-10 content in Boral, and is chosen so that the absolute B-10 content credited in the criticality analysis is the same for the two materials. This makes the two materials identical from a criticality perspective. This is confirmed by comparing results for a selected number of cases that were performed with both materials (see Section 6.4.11). Calculations in this chapter are therefore only performed for the Boral neutron absorber, with results directly applicable to Metamic.

The HI-STORM 100 System includes the HI-TRAC transfer cask and the HI-STORM storage cask. The HI-TRAC transfer cask is required for loading and unloading fuel into the MPC and for transfer of the MPC into the HI-STORM storage cask. HI-TRAC uses a lead shield for gamma radiation and a water-filled jacket for neutron shielding. The HI-STORM storage cask uses concrete as a shield for both gamma and neutron radiation. Both the HI-TRAC transfer cask and the HI-STORM storage cask, as well as the HI-STAR System[†], accommodate the interchangeable MPC designs. The three cask designs (HI-STAR, HI-STORM, and HI-TRAC) differ only in the overpack reflector materials (steel for HI-STAR, concrete for HI-STORM, and lead for HI-TRAC), which do not significantly affect the reactivity. Consequently, analyses for the HI-STAR System are directly applicable to the HI-STORM 100 system and vice versa. Therefore, the majority of criticality calculations to support both the HI-STAR and the HI-STORM System have been performed for only one of the two systems, namely the HI-STAR System. Only a selected number of analyses has been performed for both systems to demonstrate that this approach is valid. Therefore, unless specifically noted otherwise, all analyses documented throughout this chapter have been performed for the HI-STAR System. For the cases where analyses were performed for both the HI-STORM and HI-STAR System, this is clearly indicated.

The HI-STORM 100 System for storage (concrete overpack) is dry (no moderator), and thus, the reactivity is very low (k_{eff} <0.52). However, the HI-STORM 100 System for cask transfer (HI-TRAC, lead overpack) is flooded for loading and unloading operations, and thus, represents the limiting case in terms of reactivity.

The MPC-24EF, *MPC-32F and MPC-68FF* contains the same basket as the MPC-24E, *MPC-32 and MPC-68, respectively*. More specifically, all dimensions relevant to the criticality analyses are identical between the MPC-24E and MPC-24EF, *the MPC-32 and MPC-32F, and the MPC-68 and MPC-68FF*. Therefore, all criticality results obtained for the MPC-24E, *MPC-32 and MPC-68* are valid for the MPC-24EF, *MPC-32F and MPC-68FF, respectively,* and no separate analyses for the MPC-24EF, *MPC-32F and MPC-68FF* are necessary. *Therefore, throughout this chapter and unless otherwise noted, 'MPC-68' refers to 'MPC-68 and/or MPC-68FF', 'MPC-24E' or 'MPC-24E/EF' refers to 'MPC-24E and/or MPC-24EF', and 'MPC-32' or 'MPC-32/32F' refers to 'MPC-32 and/or MPC-32F'.*

---

† Analyses for the HI-STAR System have previously been submitted to the USNRC under Docket Numbers 72-1008 and 71-9261.

~~The MPC-68FF contains the same basket as the MPC-68. More specifically, all dimensions relevant to the criticality analyses are identical between the MPC-68 and MPC-68FF. Therefore, all criticality results obtained for the MPC-68 are valid for the MPC-68FF and no separate analyses for the MPC-68FF are necessary.~~

Confirmation of the criticality safety of the HI-STORM 100 System was accomplished with the three-dimensional Monte Carlo code MCNP4a [6.1.4]. Independent confirmatory calculations were made with NITAWL-KENO5a from the SCALE-4.3 package [6.4.1]. KENO5a [6.1.5] calculations used the 238-group SCALE cross-section library in association with the NITAWL-II program [6.1.6], which adjusts the uranium-238 cross sections to compensate for resonance self-shielding effects. The Dancoff factors required by NITAWL-II were calculated with the CELLDAN code [6.1.13], which includes the SUPERDAN code [6.1.7] as a subroutine. K-factors for one-sided statistical tolerance limits with 95% probability at the 95% confidence level were obtained from the National Bureau of Standards (now NIST) Handbook 91 [6.1.8].

To assess the incremental reactivity effects due to manufacturing tolerances, CASMO-3, a two-dimensional transport theory code [6.1.9-6.1.12] for fuel assemblies, and MCNP4a [6.1.4] were used. The CASMO-3 and MCNP4a calculations identify those tolerances that cause a positive reactivity effect, enabling the subsequent Monte Carlo code input to define the worst case (most conservative) conditions. CASMO-3 was not used for quantitative information, but only to qualitatively indicate the direction and approximate magnitude of the reactivity effects of the manufacturing tolerances.

Benchmark calculations were made to compare the primary code packages (MCNP4a and KENO5a) with experimental data, using critical experiments selected to encompass, insofar as practical, the design parameters of the HI-STORM 100 System. The most important parameters are (1) the enrichment, (2) the water-gap size (MPC-24, MPC-24E and MPC-24EF) or cell spacing (MPC-32, *MPC-32F, MPC-68, MPC-68F* and MPC-68*FF*), (3) the ^{10}B loading of the neutron absorber panels, and (4) the soluble boron concentration in the water. The critical experiment benchmarking is presented in Appendix 6.A.

Applicable codes, standards, and regulations, or pertinent sections thereof, include the following:

- NUREG-1536, Standard Review Plan for Dry Cask Storage Systems, USNRC, Washington D.C., January 1997.
- 10CFR72.124, Criteria For Nuclear Criticality Safety.
- Code of Federal Regulations, Title 10, Part 50, Appendix A, General Design Criterion 62, Prevention of Criticality in Fuel Storage and Handling.

- USNRC Standard Review Plan, NUREG-0800, Section 9.1.2, Spent Fuel Storage, Rev. 3, July 1981.

To assure the true reactivity will always be less than the calculated reactivity, the following conservative design criteria and assumptions were made:

- The MPCs are assumed to contain the most reactive fresh fuel authorized to be loaded into a specific basket design.

- Consistent with NUREG-1536, no credit for fuel burnup is assumed, either in depleting the quantity of fissile nuclides or in producing fission product poisons.

- Consistent with NUREG-1536, the criticality analyses assume 75% of the manufacturer's minimum Boron-10 content for the Boral neutron absorber *and 90% of the manufacturer's minimum Boron-10 content for the Metamic neutron absorber.*

- The fuel stack density is conservatively assumed to be *at least* 96% of theoretical (10.522 g/cm^3) for all criticality analyses. Fuel stack density is approximately equal to 98% of the pellet density. Therefore, while the pellet density of some fuels may be slightly greater than 96% of theoretical, the actual stack density will be less.

- No credit is taken for the ^{234}U and ^{236}U in the fuel.

- When flooded, the moderator is assumed to be water, with or without soluble boron, at a temperature and density corresponding to the highest reactivity within the expected operating range.

- When credit is taken for soluble boron, a ^{10}B content of 18.0 wt% in boron is assumed.

- Neutron absorption in minor structural members and optional heat conduction elements is neglected, i.e., spacer grids, basket supports, and optional aluminum heat conduction elements are replaced by water.

- Consistent with NUREG-1536, the worst hypothetical combination of tolerances (most conservative values within the range of acceptable values), as identified in Section 6.3, is assumed.

- When flooded, the fuel rod pellet-to-clad gap regions are assumed to be flooded with pure unborated water.

- Planar-averaged enrichments are assumed for BWR fuel. (Consistent with NUREG-1536, analysis is presented in Appendix 6.B to demonstrate that the use of planar-average enrichments produces conservative results.)

- Consistent with NUREG-1536, fuel-related burnable neutron absorbers, such as the Gadolinia normally used in BWR fuel and IFBA normally used in PWR fuel, are neglected.

- For evaluation of the bias, all benchmark calculations that result in a k_{eff} greater than 1.0 are conservatively truncated to 1.0000, consistent with NUREG-1536.

- The water reflector above and below the fuel is assumed to be unborated water, even if borated water is used in the fuel region.

- For fuel assemblies that contain low-enriched axial blankets, the governing enrichment is that of the highest planar average, and the blankets are not included in determining the average enrichment.

- *Regarding the position of assemblies in the basket, configurations with centered and eccentric positioning of assemblies in the fuel storage locations are considered. For further discussions see Section 6.3.3.*

- For intact fuel assemblies, as defined in ~~the Certificate of Compliance~~ *Table 1.0.1*, missing fuel rods must be replaced with dummy rods that displace a volume of water that is equal to, or larger than, that displaced by the original rods.

Results of the design basis criticality safety calculations for single internally flooded HI-TRAC transfer casks with full water reflection on all sides (limiting cases for the HI-STORM 100 System), *and for single unreflected, internally flooded HI-STAR casks (limiting cases for the HI-STAR 100 System),* loaded with intact fuel assemblies are listed in Tables 6.1.1 through 6.1.8, conservatively evaluated for the worst combination of manufacturing tolerances (as identified in Section 6.3), and including the calculational bias, uncertainties, and calculational statistics. ~~To~~ *Comparing corresponding results for the HI-TRAC and HI-STAR* demonstrates that the overpack material does not significantly affect the reactivity. *Consequently, analyses for the HI-STAR System are directly applicable to the HI-STORM 100 System and vice versa.*~~, results of the design basis criticality safety calculations for single unreflected, internally flooded HI-STAR casks (limiting cases for the HI-STAR 100 System) are listed in Tables 6.1.1 through 6.1.8 for comparison.~~ In addition, a few results for single internally dry (no moderator) HI-STORM storage casks with full water reflection on all external surfaces of the overpack, including the annulus region between the MPC and overpack, are listed to confirm the low reactivity of the HI-STORM 100 System in storage.

For each of the MPC designs, minimum soluble boron concentration (if applicable) and fuel assembly classes[††], Tables 6.1.1 through 6.1.8 list the bounding maximum k_{eff} value, and the associated maximum allowable enrichment. The maximum allowed enrichments and the minimum soluble boron concentrations are ~~defined in Appendix B to the Certificate of Compliance~~ *also listed in Section 2.1.9*. The candidate fuel assemblies, that are bounded by those listed in Tables 6.1.1 through 6.1.8, are given in Section 6.2.

Results of the design basis criticality safety calculations for single unreflected, internally flooded casks (limiting cases) loaded with damaged fuel assemblies or a combination of intact and damaged fuel assemblies are listed in Tables 6.1.9 through 6.1.1~~1~~*2*. The results include the calculational bias, uncertainties, and calculational statistics. For each of the MPC designs qualified for damaged fuel and/or fuel debris (MPC-24E, MPC-24EF, MPC-68, MPC-68F, ~~and~~ MPC-68FF, *MPC-32 and MPC-32F*), Tables 6.1.9 through 6.1.1~~1~~*2* indicate the maximum number of DFCs and list the fuel assembly classes, the bounding maximum k_{eff} value, ~~and~~ the associated maximum allowable enrichment, *and if applicable the minimum soluble boron concentration*. For the permissible location of DFCs see Subsection 6.4.4.2. The maximum allowed enrichments are ~~defined in Appendix B to the Certificate of Compliance~~ *also listed in Section 2.1.9*.

A table listing the maximum k_{eff} (including bias, uncertainties, and calculational statistics), calculated k_{eff}, standard deviation, and energy of the average lethargy causing fission (EALF) for each of the candidate fuel assemblies and basket configurations is provided in Appendix 6.C. These results confirm that the maximum k_{eff} values for the HI-STORM 100 System are below the limiting design criteria ($k_{eff} < 0.95$) when fully flooded and loaded with any of the candidate fuel assemblies and basket configurations. Analyses for the various conditions of flooding that support the conclusion that the fully flooded condition corresponds to the highest reactivity, and thus is most limiting, are presented in Section 6.4. The capability of the HI-STORM 100 System to safely accommodate damaged fuel and fuel debris is demonstrated in Subsection 6.4.4.

Accident conditions have also been considered and no credible accident has been identified that would result in exceeding the design criteria limit on reactivity. After the MPC is loaded with spent fuel, it is seal-welded and cannot be internally flooded. The HI-STORM 100 System for storage is dry (no moderator) and the reactivity is very low. For arrays of HI-STORM storage casks, the radiation shielding and the physical separation between overpacks due to the large diameter and cask pitch preclude any significant neutronic coupling between the casks.

---

†† For each array size (e.g., 6x6, 7x7, 14x14, etc.), the fuel assemblies have been subdivided into a number of assembly classes, where an assembly class is defined in terms of the (1) number of fuel rods; (2) pitch; (3) number and location of guide tubes (PWR) or water rods (BWR); and (4) cladding material. The assembly classes for BWR and PWR fuel are defined in Section 6.2.

Table 6.1.1

BOUNDING MAXIMUM k_{eff} VALUES FOR EACH ASSEMBLY CLASS IN THE MPC-24
(no soluble boron)

Fuel Assembly Class	Maximum Allowable Enrichment (wt% ^{235}U)	Maximum† k_{eff}		
		HI-STORM	HI-TRAC	HI-STAR
14x14A	4.6	0.3080	0.9283	0.9296
14x14B	4.6	---	0.9237	0.9228
14x14C	4.6	---	0.9274	0.9287
14x14D	4.0	---	0.8531	0.8507
14x14E	5.0	---	0.7627	0.7627
15x15A	4.1	---	0.9205	0.9204
15x15B	4.1	---	0.9387	0.9388
15x15C	4.1	---	0.9362	0.9361
15x15D	4.1	---	0.9354	0.9367
15x15E	4.1	---	0.9392	0.9368
15x15F	4.1	0.3648	0.9393††	0.9395†††
15x15G	4.0	---	0.8878	0.8876
15x15H	3.8	---	0.9333	0.9337
16x16A	4.6	0.3447	0.9273	0.9287
17x17A	4.0	0.3243	0.9378	0.9368
17x17B	4.0	---	0.9318	0.9324
17x17C	4.0	---	0.9319	0.9336

Note: The HI-STORM results are for internally dry (no moderator) HI-STORM storage casks with full water reflection on all sides, the HI-TRAC results are for internally fully flooded HI-TRAC transfer casks (which are part of the HI-STORM 100 System) with full water reflection on all sides, and the HI-STAR results are for unreflected, internally fully flooded HI-STAR casks.

† The term "maximum k_{eff}" as used here, and elsewhere in this document, means the highest possible k-effective, including bias, uncertainties, and calculational statistics, evaluated for the worst case combination of manufacturing tolerances.

†† KENO5a verification calculation resulted in a maximum k_{eff} of 0.9383.

††† KENO5a verification calculation resulted in a maximum k_{eff} of 0.9378.

Table 6.1.2

BOUNDING MAXIMUM k_{eff} VALUES FOR EACH ASSEMBLY CLASS IN THE MPC-24 WITH 400 PPM SOLUBLE BORON

Fuel Assembly Class	Maximum Allowable Enrichment (wt% ^{235}U)	Maximum† k_{eff}		
		HI-STORM	HI-TRAC	HI-STAR
14x14A	5.0	---	---	0.8884
14x14B	5.0	---	---	0.8900
14x14C	5.0	---	---	0.8950
14x14D	5.0	---	---	0.8518
14x14E	5.0	---	---	0.7132
15x15A	5.0	---	---	0.9119
15x15B	5.0	---	---	0.9284
15x15C	5.0	---	---	0.9236
15x15D	5.0	---	---	0.9261
15x15E	5.0	---	---	0.9265
15x15F	5.0	0.4013	0.9301	0.9314
15x15G	5.0	---	---	0.8939
15x15H	5.0	---	0.9345	0.9366
16x16A	5.0	---	---	0.8955
17x17A	5.0	---	---	0.9264
17x17B	5.0	---	---	0.9284
17x17C	5.0	---	0.9296	0.9294

Note: The HI-STORM results are for internally dry (no moderator) HI-STORM storage casks with full water reflection on all sides, the HI-TRAC results are for internally fully flooded HI-TRAC transfer casks (which are part of the HI-STORM 100 System) with full water reflection on all sides, and the HI-STAR results are for unreflected, internally fully flooded HI-STAR casks.

† The term "maximum k_{eff}" as used here, and elsewhere in this document, means the highest possible k-effective, including bias, uncertainties, and calculational statistics, evaluated for the worst case combination of manufacturing tolerances.

Table 6.1.3

BOUNDING MAXIMUM k_{eff} VALUES FOR EACH ASSEMBLY CLASS IN THE MPC-24E AND MPC-24EF (no soluble boron)

Fuel Assembly Class	Maximum Allowable Enrichment (wt% ^{235}U)	Maximum† k_{eff}		
		HI-STORM	HI-TRAC	HI-STAR
14x14A	5.0	---	---	0.9380
14x14B	5.0	---	---	0.9312
14x14C	5.0	---	---	0.9356
14x14D	5.0	---	---	0.8875
14x14E	5.0	---	---	0.7651
15x15A	4.5	---	---	0.9336
15x15B	4.5	---	---	0.9465
15x15C	4.5	---	---	0.9462
15x15D	4.5	---	---	0.9440
15x15E	4.5	---	---	0.9455
15x15F	4.5	0.3699	0.9465	0.9468
15x15G	4.5	---	---	0.9054
15x15H	4.2	---	---	0.9423
16x16A	5.0	---	---	0.9341
17x17A	4.4	---	0.9467	0.9447
17x17B	4.4	---	---	0.9421
17x17C	4.4	---	---	0.9433

Note: The HI-STORM results are for internally dry (no moderator) HI-STORM storage casks with full water reflection on all sides, the HI-TRAC results are for internally fully flooded HI-TRAC transfer casks (which are part of the HI-STORM 100 System) with full water reflection on all sides, and the HI-STAR results are for unreflected, internally fully flooded HI-STAR casks.

† The term "maximum k_{eff} " as used here, and elsewhere in this document, means the highest possible k-effective, including bias, uncertainties, and calculational statistics, evaluated for the worst case combination of manufacturing tolerances.

Table 6.1.4

BOUNDING MAXIMUM k_{eff} VALUES FOR EACH ASSEMBLY CLASS IN THE MPC-24E AND MPC-24EF WITH 300 PPM SOLUBLE BORON

Fuel Assembly Class	Maximum Allowable Enrichment (wt% ^{235}U)	Maximum† k_{eff}		
		HI-STORM	HI-TRAC	HI-STAR
14x14A	5.0	---	---	0.8963
14x14B	5.0	---	---	0.8974
14x14C	5.0	---	---	0.9031
14x14D	5.0	---	---	0.8588
14x14E	5.0	---	---	0.7249
15x15A	5.0	---	---	0.9161
15x15B	5.0	---	---	0.9321
15x15C	5.0	---	---	0.9271
15x15D	5.0	---	---	0.9290
15x15E	5.0	---	---	0.9309
15x15F	5.0	0.3897	0.9333	0.9332
15x15G	5.0	---	---	0.8972
15x15H	5.0	---	0.9399	0.9399
16x16A	5.0	---	---	0.9021
17x17A	5.0	---	0.9320	0.9332
17x17B	5.0	---	---	0.9316
17x17C	5.0	---	---	0.9312

Note: The HI-STORM results are for internally dry (no moderator) HI-STORM storage casks with full water reflection on all sides, the HI-TRAC results are for internally fully flooded HI-TRAC transfer casks (which are part of the HI-STORM 100 System) with full water reflection on all sides, and the HI-STAR results are for unreflected, internally fully flooded HI-STAR casks.

† The term "maximum k_{eff} " as used here, and elsewhere in this document, means the highest possible k-effective, including bias, uncertainties, and calculational statistics, evaluated for the worst case combination of manufacturing tolerances.

Table 6.1.5

BOUNDING MAXIMUM k_{eff} VALUES FOR EACH ASSEMBLY CLASS IN THE MPC-32 *AND MPC-32F FOR 4.1% ENRICHMENT* ~~WITH 1900 PPM SOLUBLE BORON~~

Fuel Assembly Class	Maximum Allowable Enrichment (wt% ^{235}U)	*Minimum Soluble Boron Concentration (ppm)*	Maximum† k_{eff}		
			HI-STORM	HI-TRAC	HI-STAR
14x14A	4.1	*1300*	---	---	*0.9041*
14x14B	4.1	*1300*	---	---	*0.9257*
14x14C	4.1	*1300*	---	---	*0.9423*
14x14D	4.1	*1300*	---	---	*0.8970*
14x14E	4.1	*1300*	---	---	*0.7340*
15x15A	4.1	*1800*	---	---	*0.9206*
15x15B	4.1	*1800*	---	---	*0.9397*
15x15C	4.1	*1800*	---	---	*0.9266*
15x15D	4.1	*1900*	---	---	*0.9384*
15x15E	4.1	*1900*	---	---	*0.9365*
15x15F	4.1	*1900*	0.4691	0.9403	0.9411
15x15G	4.1	*1800*	---	---	*0.9147*
15x15H	4.1	*1900*	---	---	*0.9276*
16x16A	4.1	*1300*	---	---	*0.9468*
17x17A	4.1	*1900*	---	---	*0.9111*
17x17B	4.1	*1900*	---	---	0.9309
17x17C	4.1	*1900*	---	0.9365	0.9355

Note: The HI-STORM results are for internally dry (no moderator) HI-STORM storage casks with full water reflection on all sides, the HI-TRAC results are for internally fully flooded HI-TRAC transfer casks (which are part of the HI-STORM 100 System) with full water reflection on all sides, and the HI-STAR results are for unreflected, internally fully flooded HI-STAR casks.

† The term "maximum k_{eff} " as used here, and elsewhere in this document, means the highest possible k-effective, including bias, uncertainties, and calculational statistics, evaluated for the worst case combination of manufacturing tolerances.

Table 6.1.6

BOUNDING MAXIMUM k_{eff} VALUES FOR EACH ASSEMBLY CLASS IN THE MPC-32 *AND MPC-32F FOR 5.0% ENRICHMENT*~~WITH 2600 PPM SOLUBLE BORON~~

Fuel Assembly Class	Maximum Allowable Enrichment (wt% ^{235}U)	*Minimum Soluble Boron Concentration (ppm)*	Maximum† k_{eff}		
			HI-STORM	HI-TRAC	HI-STAR
14x14A	5.0	*1900*	---	---	*0.9000*
14x14B	5.0	*1900*	---	---	*0.9214*
14x14C	5.0	*1900*	---	---	*0.9480*
14x14D	5.0	*1900*	---	---	*0.9050*
14x14E	5.0	*1900*	---	---	*0.7415*
15x15A	5.0	*2500*	---	---	*0.9230*
15x15B	5.0	*2500*	---	---	*0.9429*
15x15C	5.0	*2500*	---	---	*0.9307*
15x15D	5.0	*2600*	---	---	0.9466
15x15E	5.0	*2600*	---	---	0.9434
15x15F	5.0	*2600*	0.5142	0.9470	0.9483
15x15G	5.0	*2500*	---	---	*0.9251*
15x15H	5.0	*2600*	---	---	*0.9333*
16x16A	5.0	*1900*	---	---	*0.9474*
17x17A	5.0	*2600*	---	---	*0.9161*
17x17B	5.0	*2600*	---	---	0.9371
17x17C	5.0	*2600*	---	0.9436	0.9437

Note: The HI-STORM results are for internally dry (no moderator) HI-STORM storage casks with full water reflection on all sides, the HI-TRAC results are for internally fully flooded HI-TRAC transfer casks (which are part of the HI-STORM 100 System) with full water reflection on all sides, and the HI-STAR results are for unreflected, internally fully flooded HI-STAR casks.

† The term "maximum k_{eff} " as used here, and elsewhere in this document, means the highest possible k-effective, including bias, uncertainties, and calculational statistics, evaluated for the worst case combination of manufacturing tolerances.

Table 6.1.7

BOUNDING MAXIMUM k_{eff} VALUES FOR EACH ASSEMBLY CLASS IN THE MPC-68 AND MPC-68FF

Fuel Assembly Class	Maximum Allowable Planar-Average Enrichment (wt% ^{235}U)	Maximum† k_{eff}		
		HI-STORM	HI-TRAC	HI-STAR
6x6A	2.7††	---	0.7886	0.7888†††
6x6B‡	2.7††	---	0.7833	0.7824†††
6x6C	2.7††	0.2759	0.8024	0.8021†††
7x7A	2.7††	---	0.7963	0.7974†††
7x7B	4.2	0.4061	0.9385	0.9386
8x8A	2.7††	---	0.7690	0.7697†††
8x8B	4.2	0.3934	0.9427	0.9416
8x8C	4.2	0.3714	0.9429	0.9425
8x8D	4.2	---	0.9408	0.9403
8x8E	4.2	---	0.9309	0.9312
8x8F	4.0	---	0.9396	0.9411

Note: The HI-STORM results are for internally dry (no moderator) HI-STORM storage casks with full water reflection on all sides, the HI-TRAC results are for internally fully flooded HI-TRAC transfer casks (which are part of the HI-STORM 100 System) with full water reflection on all sides, and the HI-STAR results are for unreflected, internally fully flooded HI-STAR casks.

---

† The term "maximum k_{eff} " as used here, and elsewhere in this document, means the highest possible k-effective, including bias, uncertainties, and calculational statistics, evaluated for the worst case combination of manufacturing tolerances.

†† This calculation was performed for 3.0% planar-average enrichment, however, the ~~actual fuel and Certificate of Compliance~~ *authorized contents* are limited to *a* maximum planar-average enrichment of 2.7%. Therefore, the listed maximum k_{eff} value is conservative.

††† This calculation was performed for a ^{10}B loading of 0.0067 g/cm^2, which is 75% of a minimum ^{10}B loading of 0.0089 g/cm^2. The minimum ^{10}B loading in the MPC-68 is *at least* 0.037~~2~~*10* g/cm^2. Therefore, the listed maximum k_{eff} value is conservative.

‡ Assemblies in this class contain both MOX and UO_2 pins. The composition of the MOX fuel pins is given in Table 6.3.4. The maximum allowable planar-average enrichment for the MOX pins is given in the ~~Certificate of Compliance~~ *specification of authorized contents in Section 2.1.9.*

Table 6.1.7 (continued)

BOUNDING MAXIMUM k_{eff} VALUES FOR EACH ASSEMBLY CLASS IN THE MPC-68 AND MPC-68FF

Fuel Assembly Class	Maximum Allowable Planar-Average Enrichment (wt% ^{235}U)	Maximum† k_{eff}		
		HI-STORM	HI-TRAC	HI-STAR
9x9A	4.2	0.3365	0.9434	0.9417
9x9B	4.2	---	0.9417	0.9436
9x9C	4.2	---	0.9377	0.9395
9x9D	4.2	---	0.9387	0.9394
9x9E	4.0		0.9402	0.9401
9x9F	4.0	---	0.9402	0.9401
9x9G	4.2	---	0.9307	0.9309
10x10A	4.2	0.3379	0.9448‡‡	0.9457*
10x10B	4.2	---	0.9443	0.9436
10x10C	4.2	---	0.9430	0.9433
10x10D	4.0	---	0.9383	0.9376
10x10E	4.0	---	0.9157	0.9185

Note: The HI-STORM results are for internally dry (no moderator) HI-STORM storage casks with full water reflection on all sides, the HI-TRAC results are for internally fully flooded HI-TRAC transfer casks (which are part of the HI-STORM 100 System) with full water reflection on all sides, and the HI-STAR results are for unreflected, internally fully flooded HI-STAR casks.

† The term "maximum k_{eff} " as used here, and elsewhere in this document, means the highest possible k-effective, including bias, uncertainties, and calculational statistics, evaluated for the worst case combination of manufacturing tolerances.

‡‡ KENO5a verification calculation resulted in a maximum k_{eff} of 0.9451.

* KENO5a verification calculation resulted in a maximum k_{eff} of 0.9453.

Table 6.1.8

BOUNDING MAXIMUM k_{eff} VALUES FOR EACH ASSEMBLY CLASS IN THE MPC-68F

Fuel Assembly Class	Maximum Allowable Planar-Average Enrichment (wt% ^{235}U)	Maximum† k_{eff}		
		HI-STORM	HI-TRAC	HI-STAR
6x6A	2.7††	---	0.7886	0.7888
6x6B†††	2.7	---	0.7833	0.7824
6x6C	2.7	0.2759	0.8024	0.8021
7x7A	2.7	---	0.7963	0.7974
8x8A	2.7	---	0.7690	0.7697

Notes:

1. The HI-STORM results are for internally dry (no moderator) HI-STORM storage casks with full water reflection on all sides, the HI-TRAC results are for internally fully flooded HI-TRAC transfer casks (which are part of the HI-STORM 100 System) with full water reflection on all sides, and the HI-STAR results are for unreflected, internally fully flooded HI-STAR casks.

2. These calculations were performed for a ^{10}B loading of 0.0067 g/cm^2, which is 75% of a minimum ^{10}B loading of 0.0089 g/cm^2. The minimum ^{10}B loading in the MPC-68F is 0.010 g/cm^2. Therefore, the listed maximum k_{eff} values are conservative.

---

† The term "maximum k_{eff} " as used here, and elsewhere in this document, means the highest possible k-effective, including bias, uncertainties, and calculational statistics, evaluated for the worst case combination of manufacturing tolerances.

†† These calculations were performed for 3.0% planar-average enrichment, however, the ~~actual fuel and Certificate of Compliance~~ *authorized contents* are limited to a maximum planar-average enrichment of 2.7%. Therefore, the listed maximum k_{eff} values are conservative.

††† Assemblies in this class contain both MOX and UO_2 pins. The composition of the MOX fuel pins is given in Table 6.3.4. The maximum allowable planar-average enrichment for the MOX pins is specified in the ~~Certificate of Compliance~~ *specification of authorized contents in Section 2.1.9.*

Table 6.1.9

BOUNDING MAXIMUM k_{eff} VALUES FOR THE MPC-24E AND MPC-24EF WITH UP TO 4 DFCs

Fuel Assembly Class	**Maximum Allowable Enrichment (wt% ^{235}U)**		***Minimum Soluble Boron Concentration (ppm)***	**Maximum k_{eff}**	
	Intact Fuel	**Damaged Fuel *and Fuel Debris***		**HI-TRAC**	**HI-STAR**
All PWR Classes	4.0	4.0	*0*	0.9486	0.9480
All PWR Classes	*5.0*	*5.0*	*600*	*0.9177*	*0.9185*

Table 6.1.10

BOUNDING MAXIMUM k_{eff} VALUES FOR THE MPC-68, MPC-68F AND MPC-68FF WITH UP TO 68 DFCs

Fuel Assembly Class	**Maximum Allowable Planar-Average Enrichment (wt% ^{235}U)**		**Maximum k_{eff}**	
	Intact Fuel	**Damaged Fuel *and Fuel Debris***	**HI-TRAC**	**HI-STAR**
6x6A, 6x6B, 6x6C, 7x7A, 8x8A	2.7	2.7	0.8024	0.8021

Table 6.1.11

BOUNDING MAXIMUM k_{eff} VALUES FOR THE MPC-68 AND MPC-68FF WITH UP TO 16 DFCs

Fuel Assembly Class	**Maximum Allowable Planar-Average Enrichment (wt% ^{235}U)**		**Maximum k_{eff}**	
	Intact Fuel	**Damaged Fuel *and Fuel Debris***	**HI-TRAC**	**HI-STAR**
All BWR Classes	3.7	4.0	0.9328	0.9328

Table 6.1.12

BOUNDING MAXIMUM k_{eff} VALUES FOR THE MPC-32 AND MPC-32F WITH UP TO 8 DFCs

Fuel Assembly Class of Intact Fuel	Maximum Allowable Enrichment for Intact Fuel and Damaged Fuel/Fuel Debris (wt% ^{235}U)	Minimum Soluble Boron Content (ppm)	Maximum k_{eff}	
			HI-TRAC	HI-STAR
14x14A, B, C, D, E	4.1	1500	---	0.9336
	5.0	2300	---	0.9269
15x15A, B, C, G	4.1	1900	0.9349	0.9350
	5.0	2700	---	0.9365
15x15D, E, F, H	4.1	2100	---	0.9340
	5.0	2900	0.9382	0.9397
16x16A	4.1	1500	---	0.9335
	5.0	2300	---	0.9289
17x17A, B, C	4.1	2100	---	0.9294
	5.0	2900	---	0.9367

6.2 SPENT FUEL LOADING

Specifications for the BWR and PWR fuel assemblies that were analyzed are given in Tables 6.2.1 and 6.2.2, respectively. For the BWR fuel characteristics, the number and dimensions for the water rods are the actual number and dimensions. For the PWR fuel characteristics, the actual number and dimensions of the control rod guide tubes and thimbles are used. Table 6.2.1 lists 72 unique BWR assemblies while Table 6.2.2 lists 46 unique PWR assemblies, all of which were explicitly analyzed for this evaluation. Examination of Tables 6.2.1 and 6.2.2 reveals that there are a large number of minor variations in fuel assembly dimensions.

Due to the large number of minor variations in the fuel assembly dimensions, the use of explicit dimensions ~~in the Certificate of Compliance~~ *defining the authorized contents* could limit the applicability of the HI-STORM 100 System. To resolve this limitation, bounding criticality analyses are presented in this section for a number of defined fuel assembly classes for both fuel types (PWR and BWR). The results of the bounding criticality analyses justify using bounding fuel dimensions~~, as defined in the Certificate of Compliance~~ *for defining the authorized contents.*

6.2.1 Definition of Assembly Classes

For each array size (e.g., 6x6, 7x7, 15x15, etc.), the fuel assemblies have been subdivided into a number of defined classes, where a class is defined in terms of (1) the number of fuel rods; (2) pitch; (3) number and locations of guide tubes (PWR) or water rods (BWR); and (4) cladding material. The assembly classes for BWR and PWR fuel are defined in Tables 6.2.1 and 6.2.2, respectively. It should be noted that these assembly classes are unique to this evaluation and are not known to be consistent with any class designations in the open literature.

For each assembly class, calculations have been performed for all of the dimensional variations for which data is available (i.e., all data in Tables 6.2.1 and 6.2.2). These calculations demonstrate that the maximum reactivity corresponds to:

- maximum active fuel length,
- maximum fuel pellet diameter,
- minimum cladding outside diameter (OD),
- maximum cladding inside diameter (ID),
- minimum guide tube/water rod thickness, and
- maximum channel thickness (for BWR assemblies only).

Therefore, for each assembly class, a bounding assembly was defined based on the above characteristics and a calculation for the bounding assembly was performed to demonstrate compliance with the regulatory requirement of $k_{eff} < 0.95$. In some assembly classes this bounding assembly corresponds directly to one of the actual (real) assemblies; while in most

assembly classes, the bounding assembly is artificial (i.e., based on bounding dimensions from more than one of the actual assemblies). In classes where the bounding assembly is artificial, the reactivity of the actual (real) assemblies is typically much less than that of the bounding assembly; thereby providing additional conservatism. As a result of these analyses, ~~the Certificate of Compliance will define acceptability~~ *the authorized contents in Section 2.1.9 are defined* in terms of the bounding assembly parameters for each class.

To demonstrate that the aforementioned characteristics are bounding, a parametric study was performed for a reference BWR assembly, designated herein as 8x8C04 (identified generally as a GE8x8R). Additionally, parametric studies were performed for a PWR assembly (the 15x15F assembly class) in the MPC-24 and MPC-32 with soluble boron in the water flooding the MPC. The results of these studies are shown in Table 6.2.3 through 6.2.5, and verify the positive reactivity effect associated with (1) increasing the pellet diameter, (2) maximizing the cladding ID (while maintaining a constant cladding OD), (3) minimizing the cladding OD (while maintaining a constant cladding ID), (4) decreasing the water rod/guide tube thickness, (5) artificially replacing the Zircaloy water rod tubes/guide tubes with water, ~~and~~ (6) maximizing the channel thickness (for BWR Assemblies), *and (7) increasing the active length*. These results, and the many that follow, justify the approach for using bounding dimensions ~~in the Certificate of Compliance~~ *for defining the authorized contents*. Where margins permit, the Zircaloy water rod tubes (BWR assemblies) are artificially replaced by water in the bounding cases to remove the requirement for water rod thickness from the ~~Certificate of Compliance~~ *specification of the authorized contents*. As these studies were performed with and without soluble boron, they also demonstrate that the bounding dimensions are valid independent of the soluble boron concentration.

As mentioned, the bounding approach used in these analyses often results in a maximum k_{eff} value for a given class of assemblies that is much greater than the reactivity of any of the actual (real) assemblies within the class, and yet, is still below the 0.95 regulatory limit.

6.2.2 Intact PWR Fuel Assemblies

6.2.2.1 Intact PWR Fuel Assemblies in the MPC-24 without Soluble Boron

For PWR fuel assemblies (specifications listed in Table 6.2.2) the 15x15F01 fuel assembly at 4.1% enrichment has the highest reactivity (maximum k_{eff} of 0.9395). The 17x17A01 assembly (otherwise known as a Westinghouse 17x17 OFA) has a similar reactivity (see Table 6.2.20) and was used throughout this criticality evaluation as a reference PWR assembly. The 17x17A01 assembly is a representative PWR fuel assembly in terms of design and reactivity and is useful for the reactivity studies presented in Sections 6.3 and 6.4. Calculations for the various PWR fuel assemblies in the MPC-24 are summarized in Tables 6.2.6 through 6.2.22 for the fully flooded condition without soluble boron in the water.

Tables 6.2.6 through 6.2.22 show the maximum k_{eff} values for the assembly classes that are acceptable for storage in the MPC-24. All maximum k_{eff} values include the bias, uncertainties, and calculational statistics, evaluated for the worst combination of manufacturing tolerances. All calculations for the MPC-24 were performed for a ^{10}B loading of 0.020 g/cm^2, which is 75% of the minimum loading~~,~~*of* 0.0267 g/cm^2 *for Boral, or 90% of the minimum loading of 0.0223 g/cm^2 for Metamic*~~, specified on BM-1478, Bill of Materials for 24-Assembly HI-STAR 100 PWR MPC, in Section 1.5~~. The maximum allowable enrichment in the MPC-24 varies from 3.8 to 5.0 wt% ^{235}U, depending on the assembly class, and is defined in Tables 6.2.6 through 6.2.22. It should be noted that the maximum allowable enrichment does not vary within an assembly class. Table 6.1.1 summarizes the maximum allowable enrichments for each of the assembly classes that are acceptable for storage in the MPC-24.

Tables 6.2.6 through 6.2.22 are formatted with the assembly class information in the top row, the unique assembly designations, dimensions, and k_{eff} values in the following rows above the bold double lines, and the bounding dimensions selected~~ for the Certificate of Compliance~~ *to define the authorized contents* and corresponding bounding k_{eff} values in the final rows. Where the bounding assembly corresponds directly to one of the actual assemblies, the fuel assembly designation is listed in the bottom row in parentheses (e.g., Table 6.2.6). Otherwise, the bounding assembly is given a unique designation. For an assembly class that contains only a single assembly (e.g., 14x14D, see Table 6.2.9), the *authorized contents* dimensions~~ listed in the Certificate of Compliance~~ are based on the assembly dimensions from that single assembly. All of the maximum k_{eff} values corresponding to the selected bounding dimensions are greater than or equal to those for the actual assembly dimensions and are below the 0.95 regulatory limit.

The results of the analyses for the MPC-24, which were performed for all assemblies in each class (see Tables 6.2.6 through 6.2.22), further confirm the validity of the bounding dimensions established in Section 6.2.1. Thus, for all following calculations, namely analyses of the MPC-24E, MPC-32, and MPC-24 with soluble boron present in the water, only the bounding assembly in each class is analyzed.

6.2.2.2 Intact PWR Fuel Assemblies in the MPC-24 with Soluble Boron

Additionally, the HI-STAR 100 system is designed to allow credit for the soluble boron typically present in the water of PWR spent fuel pools. For a minimum soluble boron concentration of 400ppm, the maximum allowable fuel enrichment is 5.0 wt% ^{235}U for all assembly classes identified in Tables 6.2.6 through 6.2.22. Table 6.1.2 shows the maximum k_{eff} for the bounding assembly in each assembly class. All maximum k_{eff} values are below the 0.95 regulatory limit. The 15x15H assembly class has the highest reactivity (maximum k_{eff} of 0.9366). The calculated k_{eff} and calculational uncertainty for each class is listed in Appendix 6.C.

6.2.2.3 Intact PWR Assemblies in the MPC-24E and MPC-24EF with and without Soluble Boron

The MPC-24E and MPC-24EF are variations of the MPC-24, which provide for storage of higher enriched fuel than the MPC-24 through optimization of the storage cell layout. The MPC-24E and MPC-24EF also allow for the loading of up to 4 PWR Damaged Fuel Containers (DFC) with damaged PWR fuel (MPC-24E and MPC–24EF) and PWR fuel debris (MPC-24EF only). The requirements for damaged fuel and fuel debris in the MPC-24E and MPC-24EF are discussed in Section 6.2.4.3.

Without credit for soluble boron, the maximum allowable fuel enrichment varies between 4.2 and 5.0 wt% ^{235}U, depending on the assembly classes as identified in Tables 6.2.6 through 6.2.22. The maximum allowable enrichment for each assembly class is listed in Table 6.1.3, together with the maximum k_{eff} for the bounding assembly in the assembly class. All maximum k_{eff} values are below the 0.95 regulatory limit The 15x15F assembly class at 4.5% enrichment has the highest reactivity (maximum k_{eff} of 0.9468). The calculated k_{eff} and calculational uncertainty for each class is listed in Appendix 6.C.

For a minimum soluble boron concentration of 300ppm, the maximum allowable fuel enrichment is 5.0 wt% ^{235}U for all assembly classes identified in Tables 6.2.6 through 6.2.22. Table 6.1.4 shows the maximum k_{eff} for the bounding assembly in each assembly class. All maximum k_{eff} values are below the 0.95 regulatory limit. The 15x15H assembly class has the highest reactivity (maximum k_{eff} of 0.9399). The calculated k_{eff} and calculational uncertainty for each class is listed in Appendix 6.C.

6.2.2.4 Intact PWR Assemblies in the MPC-32 *and MPC-32F*

When loading any PWR fuel assembly in the MPC-32 *or MPC-32F*, a minimum soluble boron concentration is required.

For a ~~minimum soluble boron concentration of 1900ppm, the~~ maximum allowable fuel enrichment ~~is~~*of* 4.1 wt% ^{235}U for all assembly classes identified in Tables 6.2.6 through 6.2.22, *a minimum soluble boron concentration between 1300ppm and 1900ppm is required, depending on the assembly class.* Table 6.1.5 shows the maximum k_{eff} for the bounding assembly in each assembly class. All maximum k_{eff} values are below the 0.95 regulatory limit. The ~~15x15F~~*16x16A* assembly class has the highest reactivity (maximum k_{eff} of 0.94~~11~~*68*). The calculated k_{eff} and calculational uncertainty for each class is listed in Appendix 6.C.

For a ~~minimum soluble boron concentration of 2600ppm, the~~ maximum allowable fuel enrichment ~~is~~*of* 5.0 wt% ^{235}U for all assembly classes identified in Tables 6.2.6 through 6.2.22, *a minimum soluble boron concentration between 1900ppm and 2600ppm is required, depending*

on the assembly class. Table 6.1.6 shows the maximum k_{eff} for the bounding assembly in each assembly class. All maximum k_{eff} values are below the 0.95 regulatory limit. The 15x15F assembly class has the highest reactivity (maximum k_{eff} of 0.9483). The calculated k_{eff} and calculational uncertainty for each class is listed in Appendix 6.C.

6.2.3 Intact BWR Fuel Assemblies in the MPC-68 and MPC-68FF

For BWR fuel assemblies (specifications listed in Table 6.2.1) the artificial bounding assembly for the 10x10A assembly class at 4.2% enrichment has the highest reactivity (maximum k_{eff} of 0.9457). Calculations for the various BWR fuel assemblies in the MPC-68 and MPC-68FF are summarized in Tables 6.2.23 through 6.2.40 for the fully flooded condition. In all cases, the gadolinia (Gd_2O_3) normally incorporated in BWR fuel was conservatively neglected.

For calculations involving BWR assemblies, the use of a uniform (planar-average) enrichment, as opposed to the distributed enrichments normally used in BWR fuel, produces conservative results. Calculations confirming this statement are presented in Appendix 6.B for several representative BWR fuel assembly designs. These calculations justify the specification of planar-average enrichments to define acceptability of BWR fuel for loading into the MPC-68.

Tables 6.2.23 through 6.2.40 show the maximum k_{eff} values for assembly classes that are acceptable for storage in the MPC-68 and MPC-68FF. All maximum k_{eff} values include the bias, uncertainties, and calculational statistics, evaluated for the worst combination of manufacturing tolerances. With the exception of assembly classes 6x6A, 6x6B, 6x6C, 7x7A, and 8x8A, which will be discussed in Section 6.2.4, all calculations for the MPC-68 and MPC-68FF were performed with a ^{10}B loading of 0.0279 g/cm^2, which is 75% of the minimum loading~~,~~ *of* 0.0372 g/cm^2 *for Boral, or 90% of the minimum loading of 0.031 g/cm^2 for Metamic*~~, specified on BM-1479, Bill of Materials for 68-Assembly HI-STAR 100 BWR MPC, in Section 1.5~~. Calculations for assembly classes 6x6A, 6x6B, 6x6C, 7x7A, and 8x8A were conservatively performed with a ^{10}B loading of 0.0067 g/cm^2. The maximum allowable enrichment in the MPC-68 and MPC-68FF varies from 2.7 to 4.2 wt% ^{235}U, depending on the assembly class. It should be noted that the maximum allowable enrichment does not vary within an assembly class. Table 6.1.7 summarizes the maximum allowable enrichments for all assembly classes that are acceptable for storage in the MPC-68 and MPC-68FF.

Tables 6.2.23 through 6.2.40 are formatted with the assembly class information in the top row, the unique assembly designations, dimensions, and k_{eff} values in the following rows above the bold double lines, and the bounding dimensions selected~~ for the Certificate of Compliance~~ *to define the authorized contents* and corresponding bounding k_{eff} values in the final rows. Where an assembly class contains only a single assembly (e.g., 8x8E, see Table 6.2.27), the *authorized contents* dimensions~~ listed in the Certificate of Compliance~~ are based on the assembly dimensions from that single assembly. For assembly classes that are suspected to contain

assemblies with thicker channels (e.g., 120 mils), bounding calculations are also performed to qualify the thicker channels (e.g. 7x7B, see Table 6.2.23). All of the maximum k_{eff} values corresponding to the selected bounding dimensions are shown to be greater than or equal to those for the actual assembly dimensions and are below the 0.95 regulatory limit.

For assembly classes that contain partial length rods (i.e., 9x9A, 10x10A, and 10x10B), calculations were performed for the actual (real) assembly configuration and for the axial segments (assumed to be full length) with and without the partial length rods. In all cases, the axial segment with only the full length rods present (where the partial length rods are absent) is bounding. Therefore, the bounding maximum k_{eff} values reported for assembly classes that contain partial length rods bound the reactivity regardless of the active fuel length of the partial length rods. As a result, the ~~Certificate of Compliance~~ *specification of the authorized contents* has no minimum requirement for the active fuel length of the partial length rods.

For BWR fuel assembly classes where margins permit, the Zircaloy water rod tubes are artificially replaced by water in the bounding cases to remove the requirement for water rod thickness from the ~~Certificate of Compliance~~ *specification of the authorized contents*. For these cases, the bounding water rod thickness is listed as zero.

As mentioned, the highest observed maximum k_{eff} value is 0.9457, corresponding to the artificial bounding assembly in the 10x10A assembly class. This assembly has the following bounding characteristics: (1) the partial length rods are assumed to be zero length (most reactive configuration); (2) the channel is assumed to be 120 mils thick; and (3) the active fuel length of the full length rods is 155 inches. Therefore, the maximum reactivity value is bounding compared to any of the real BWR assemblies listed.

6.2.4 BWR and PWR Damaged Fuel Assemblies and Fuel Debris

In addition to storing intact PWR and BWR fuel assemblies, the HI-STORM 100 System is designed to store BWR and PWR damaged fuel assemblies and fuel debris. Damaged fuel assemblies and fuel debris are defined in ~~Section 2.1.3 and Appendix B to the Certificate of Compliance~~ *Table 1.0.1*. Both damaged fuel assemblies and fuel debris are required to be loaded into Damaged Fuel Containers (DFCs) prior to being loaded into the MPC. ~~Four~~*Five* different DFC types with different cross sections are considered; three types for BWR fuel and ~~one~~*two* for PWR fuel. DFCs containing fuel debris must be stored in the MPC-68F, MPC-68FF, ~~or~~ MPC-24EF *or MPC-32F*. DFCs containing BWR damaged fuel assemblies may be stored in the MPC-68, MPC-68F or MPC-68FF. DFCs containing PWR damaged fuel may be stored in the MPC-24E, ~~and~~ MPC-24EF, *MPC-32 or MPC-32F*. The criticality evaluation of various possible damaged conditions of the fuel is presented in Subsection 6.4.4.

6.2.4.1 Damaged BWR Fuel Assemblies and BWR Fuel Debris in Assembly Classes 6x6A, 6x6B, 6x6C, 7x7A and 8x8A

Tables 6.2.41 through 6.2.45 show the maximum k_{eff} values for the five assembly classes 6x6A, 6x6B, 6x6C, 7x7A and 8x8A. All maximum k_{eff} values include the bias, uncertainties, and calculational statistics, evaluated for the worst combination of manufacturing tolerances. All calculations were performed for a ^{10}B loading of 0.0067 g/cm^2, which is 75% of a minimum loading, 0.0089 g/cm^2. However, because the practical manufacturing lower limit for minimum ^{10}B loading is 0.01 g/cm^2, the minimum ^{10}B loading of 0.01 g/cm^2 is specified on ~~BM-1479, Bill of Materials for 68-Assembly HI-STAR 100 BWR MPC,~~ *the drawing* in Section 1.5, for the MPC-68F. As an additional level of conservatism in the analyses, the calculations were performed for an enrichment of 3.0 wt% ^{235}U, while the maximum allowable enrichment for these assembly classes is limited to 2.7 wt% ^{235}U in the ~~Certificate of Compliance~~ *specification of the authorized contents*. Therefore, the maximum k_{eff} values for damaged BWR fuel assemblies and fuel debris are conservative. Calculations for the various BWR fuel assemblies in the MPC-68F are summarized in Tables 6.2.41 through 6.2.45 for the fully flooded condition.

For the assemblies that may be stored as damaged fuel or fuel debris, the 6x6C01 assembly at 3.0 wt% ^{235}U enrichment has the highest reactivity (maximum k_{eff} of 0.8021). Considering all of the conservatism built into this analysis (e.g., higher than allowed enrichment and lower than actual ^{10}B loading), the actual reactivity will be lower.

Because the analysis for the damaged BWR fuel assemblies and fuel debris was performed for a ^{10}B loading of 0.0089 g/cm^2, which conservatively bounds the analysis of damaged BWR fuel assemblies in an MPC-68 or MPC-68FF with a minimum ^{10}B loading of 0.0372 g/cm^2, damaged BWR fuel assemblies may also be stored in the MPC-68 or MPC-68FF. However, fuel debris is limited to the MPC-68F and MPC-68FF by ~~Appendix B to the Certificate of Compliance~~ *the specification of the authorized contents*.

Tables 6.2.41 through 6.2.45 are formatted with the assembly class information in the top row, the unique assembly designations, dimensions, and k_{eff} values in the following rows above the bold double lines, and the bounding dimensions selected ~~for the Certificate of Compliance~~ *to define the authorized contents* and corresponding bounding k_{eff} values in the final rows. Where an assembly class contains only a single assembly (e.g., 6x6C, see Table 6.2.43), the *authorized contents* dimensions ~~listed in the Certificate of Compliance~~ are based on the assembly dimensions from that single assembly. All of the maximum k_{eff} values corresponding to the selected bounding dimensions are greater than or equal to those for the actual assembly dimensions and are well below the 0.95 regulatory limit.

6.2.4.2 Damaged BWR Fuel Assemblies and Fuel Debris in the MPC-68 and MPC-68FF

Damaged BWR fuel assemblies and fuel debris from all BWR classes may be loaded into the MPC-68 and MPC-68FF by restricting the locations of the DFCs to 16 specific cells on the periphery of the fuel basket. The MPC-68 may be loaded with up to 16 DFCs containing damaged fuel assemblies. The MPC-68FF may also be loaded with up to 16 DFCs, with up to 8 DFCs containing fuel debris.

For all assembly classes, the enrichment of the damaged fuel or fuel debris is limited to a maximum of 4.0 wt% ^{235}U, while the enrichment of the intact assemblies stored together with the damaged fuel is limited to a maximum of 3.7 wt% ^{235}U. The maximum k_{eff} is 0.9328. The criticality evaluation of the damaged fuel assemblies and fuel debris in the MPC-68 and MPC-68FF is presented in Section 6.4.4.2.

6.2.4.3 Damaged PWR Fuel Assemblies and Fuel Debris ~~in the MPC-24E and MPC-24EF~~

In addition to storing intact PWR fuel assemblies, the HI-STORM 100 System is designed to store damaged PWR fuel assemblies (MPC-24E, ~~and~~ MPC-24EF, *MPC-32 and MPC-32F*) and fuel debris (MPC-24EF *and MPC-32F* only). Damaged fuel assemblies and fuel debris are defined in ~~Section 2.1.3 and Appendix B of the Certificate of Compliance~~ *Table 1.0.1*. Damaged PWR fuel assemblies and fuel debris are required to be loaded into PWR Damaged Fuel Containers (DFCs) ~~prior to being loaded into the MPC~~.

6.2.4.3.1 Damaged PWR Fuel Assemblies and Fuel Debris in the MPC-24E and MPC-24EF

Up to four DFCs may be stored in the MPC-24E or MPC-24EF. When loaded with damaged fuel and/or fuel debris, the maximum enrichment for intact and damaged fuel is 4.0 wt% ^{235}U for all assembly classes listed in Table 6.2.6 through 6.2.22 *without credit for soluble boron*. The maximum k_{eff} for these classes is 0.9486. *For a minimum soluble boron concentration of 600ppm, the maximum enrichment for intact and damaged fuel is 5.0 wt% ^{235}U for all assembly classes listed in Table 6.2.6 through 6.2.22.* The criticality evaluation of the damaged fuel is presented in Subsection 6.4.4.2.

6.2.4.3.2 Damaged PWR Fuel Assemblies and Fuel Debris in the MPC-32 and MPC-32F

Up to eight DFCs may be stored in the MPC-32 or MPC-32F. For a maximum allowable fuel enrichment of 4.1 wt% ^{235}U for intact fuel, damaged fuel and fuel debris for all assembly classes identified in Tables 6.2.6 through 6.2.22, a minimum soluble boron concentration between 1500ppm and 2100ppm is required, depending on the assembly class of the intact assembly. For a maximum allowable fuel enrichment of 5.0 wt% ^{235}U for intact fuel, damaged fuel and fuel debris, a minimum soluble boron concentration between 2300ppm and 2900ppm is required,

depending on the assembly class of the intact assembly. Table 6.1.12 shows the maximum k_{eff} by assembly class. All maximum k_{eff} values are below the 0.95 regulatory limit.

6.2.5 Thoria Rod Canister

Additionally, the HI-STORM 100 System is designed to store a Thoria Rod Canister in the MPC-68, MPC-68F or MPC-68FF. The canister is similar to a DFC and contains 18 intact Thoria Rods placed in a separator assembly. The reactivity of the canister in the MPC is very low compared to the approved fuel assemblies (The ^{235}U content of these rods correspond to UO_2 rods with an initial enrichment of approximately 1.7 wt% ^{235}U). It is therefore permissible to the Thoria Rod Canister together with any approved content in a MPC-68 or MPC-68F. Specifications of the canister and the Thoria Rods that are used in the criticality evaluation are given in Table 6.2.46. The criticality evaluation are presented in Subsection 6.4.6.

Table 6.2.1 (page 1 of 7)
BWR FUEL CHARACTERISTICS AND ASSEMBLY CLASS DEFINITIONS
(all dimensions are in inches)

Fuel Assembly Designation	Clad Material	Pitch	Number of Fuel Rods	Cladding OD	Cladding Thickness	Pellet Diameter	Active Fuel Length	Number of Water Rods	Water Rod OD	Water Rod ID	Channel Thickness	Channel ID
6x6A Assembly Class												
6x6A01	Zr	0.694	36	0.5645	0.0350	0.4940	110.0	0	n/a	n/a	0.060	4.290
6x6A02	Zr	0.694	36	0.5645	0.0360	0.4820	110.0	0	n/a	n/a	0.060	4.290
6x6A03	Zr	0.694	36	0.5645	0.0350	0.4820	110.0	0	n/a	n/a	0.060	4.290
6x6A04	Zr	0.694	36	0.5550	0.0350	0.4820	110.0	0	n/a	n/a	0.060	4.290
6x6A05	Zr	0.696	36	0.5625	0.0350	0.4820	110.0	0	n/a	n/a	0.060	4.290
6x6A06	Zr	0.696	35	0.5625	0.0350	0.4820	110.0	1	0.0	0.0	0.060	4.290
6x6A07	Zr	0.700	36	0.5555	0.03525	0.4780	110.0	0	n/a	n/a	0.060	4.290
6x6A08	Zr	0.710	36	0.5625	0.0260	0.4980	110.0	0	n/a	n/a	0.060	4.290
6x6B (MOX) Assembly Class												
6x6B01	Zr	0.694	36	0.5645	0.0350	0.4820	110.0	0	n/a	n/a	0.060	4.290
6x6B02	Zr	0.694	36	0.5625	0.0350	0.4820	110.0	0	n/a	n/a	0.060	4.290
6x6B03	Zr	0.696	36	0.5625	0.0350	0.4820	110.0	0	n/a	n/a	0.060	4.290
6x6B04	Zr	0.696	35	0.5625	0.0350	0.4820	110.0	1	0.0	0.0	0.060	4.290
6x6B05	Zr	0.710	35	0.5625	0.0350	0.4820	110.0	1	0.0	0.0	0.060	4.290
6x6C Assembly Class												
6x6C01	Zr	0.740	36	0.5630	0.0320	0.4880	77.5	0	n/a	n/a	0.060	4.542
7x7A Assembly Class												
7x7A01	Zr	0.631	49	0.4860	0.0328	0.4110	80	0	n/a	n/a	0.060	4.542

Table 6.2.1 (page 2 of 7)
BWR FUEL CHARACTERISTICS AND ASSEMBLY CLASS DEFINITIONS
(all dimensions are in inches)

Fuel Assembly Designation	Clad Material	Pitch	Number of Fuel Rods	Cladding OD	Cladding Thickness	Pellet Diameter	Active Fuel Length	Number of Water Rods	Water Rod OD	Water Rod ID	Channel Thickness	Channel ID
7x7B Assembly Class												
7x7B01	Zr	0.738	49	0.5630	0.0320	0.4870	150	0	n/a	n/a	0.080	5.278
7x7B02	Zr	0.738	49	0.5630	0.0370	0.4770	150	0	n/a	n/a	0.102	5.291
7x7B03	Zr	0.738	49	0.5630	0.0370	0.4770	150	0	n/a	n/a	0.080	5.278
7x7B04	Zr	0.738	49	0.5700	0.0355	0.4880	150	0	n/a	n/a	0.080	5.278
7x7B05	Zr	0.738	49	0.5630	0.0340	0.4775	150	0	n/a	n/a	0.080	5.278
7x7B06	Zr	0.738	49	0.5700	0.0355	0.4910	150	0	n/a	n/a	0.080	5.278
8x8A Assembly Class												
8x8A01	Zr	0.523	64	0.4120	0.0250	0.3580	110	0	n/a	n/a	0.100	4.290
8x8A02	Zr	0.523	63	0.4120	0.0250	0.3580	120	0	n/a	n/a	0.100	4.290

Table 6.2.1 (page 3 of 7)
BWR FUEL CHARACTERISTICS AND ASSEMBLY CLASS DEFINITIONS
(all dimensions are in inches)

Fuel Assembly Designation	Clad Material	Pitch	Number of Fuel Rods	Cladding OD	Cladding Thickness	Pellet Diameter	Active Fuel Length	Number of Water Rods	Water Rod OD	Water Rod ID	Channel Thickness	Channel ID
8x8B Assembly Class												
8x8B01	Zr	0.641	63	0.4840	0.0350	0.4050	150	1	0.484	0.414	0.100	5.278
8x8B02	Zr	0.636	63	0.4840	0.0350	0.4050	150	1	0.484	0.414	0.100	5.278
8x8B03	Zr	0.640	63	0.4930	0.0340	0.4160	150	1	0.493	0.425	0.100	5.278
8x8B04	Zr	0.642	64	0.5015	0.0360	0.4195	150	0	n/a	n/a	0.100	5.278
8x8C Assembly Class												
8x8C01	Zr	0.641	62	0.4840	0.0350	0.4050	150	2	0.484	0.414	0.100	5.278
8x8C02	Zr	0.640	62	0.4830	0.0320	0.4100	150	2	0.591	0.531	0.000	no channel
8x8C03	Zr	0.640	62	0.4830	0.0320	0.4100	150	2	0.591	0.531	0.080	5.278
8x8C04	Zr	0.640	62	0.4830	0.0320	0.4100	150	2	0.591	0.531	0.100	5.278
8x8C05	Zr	0.640	62	0.4830	0.0320	0.4100	150	2	0.591	0.531	0.120	5.278
8x8C06	Zr	0.640	62	0.4830	0.0320	0.4110	150	2	0.591	0.531	0.100	5.278
8x8C07	Zr	0.640	62	0.4830	0.0340	0.4100	150	2	0.591	0.531	0.100	5.278
8x8C08	Zr	0.640	62	0.4830	0.0320	0.4100	150	2	0.493	0.425	0.100	5.278
8x8C09	Zr	0.640	62	0.4930	0.0340	0.4160	150	2	0.493	0.425	0.100	5.278
8x8C10	Zr	0.640	62	0.4830	0.0340	0.4100	150	2	0.591	0.531	0.120	5.278
8x8C11	Zr	0.640	62	0.4830	0.0340	0.4100	150	2	0.591	0.531	0.120	5.215
8x8C12	Zr	0.636	62	0.4830	0.0320	0.4110	150	2	0.591	0.531	0.120	5.215

Table 6.2.1 (page 4 of 7)
BWR FUEL CHARACTERISTICS AND ASSEMBLY CLASS DEFINITIONS
(all dimensions are in inches)

Fuel Assembly Designation	Clad Material	Pitch	Number of Fuel Rods	Cladding OD	Cladding Thickness	Pellet Diameter	Active Fuel Length	Number of Water Rods	Water Rod OD	Water Rod ID	Channel Thickness	Channel ID
8x8D Assembly Class												
8x8D01	Zr	0.640	60	0.4830	0.0320	0.4110	150	2 large/ 2 small	0.591/ 0.483	0.531/ 0.433	0.100	5.278
8x8D02	Zr	0.640	60	0.4830	0.0320	0.4110	150	4	0.591	0.531	0.100	5.278
8x8D03	Zr	0.640	60	0.4830	0.0320	0.4110	150	4	0.483	0.433	0.100	5.278
8x8D04	Zr	0.640	60	0.4830	0.0320	0.4110	150	1	1.34	1.26	0.100	5.278
8x8D05	Zr	0.640	60	0.4830	0.0320	0.4100	150	1	1.34	1.26	0.100	5.278
8x8D06	Zr	0.640	60	0.4830	0.0320	0.4110	150	1	1.34	1.26	0.120	5.278
8x8D07	Zr	0.640	60	0.4830	0.0320	0.4110	150	1	1.34	1.26	0.080	5.278
8x8D08	Zr	0.640	61	0.4830	0.0300	0.4140	150	3	0.591	0.531	0.080	5.278
8x8E Assembly Class												
8x8E01	Zr	0.640	59	0.4930	0.0340	0.4160	150	5	0.493	0.425	0.100	5.278
8x8F Assembly Class												
8x8F01	Zr	0.609	64	0.4576	0.0290	0.3913	150	4†	0.291†	0.228†	0.055	5.390
9x9A Assembly Class												
9x9A01	Zr	0.566	74	0.4400	0.0280	0.3760	150	2	0.98	0.92	0.100	5.278
9x9A02	Zr	0.566	66	0.4400	0.0280	0.3760	150	2	0.98	0.92	0.100	5.278
9x9A03	Zr	0.566	74/66	0.4400	0.0280	0.3760	150/90	2	0.98	0.92	0.100	5.278
9x9A04	Zr	0.566	66	0.4400	0.0280	0.3760	150	2	0.98	0.92	0.120	5.278

† Four rectangular water cross segments dividing the assembly into four quadrants

Table 6.2.1 (page 5 of 7)
BWR FUEL CHARACTERISTICS AND ASSEMBLY CLASS DEFINITIONS
(all dimensions are in inches)

Fuel Assembly Designation	Clad Material	Pitch	Number of Fuel Rods	Cladding OD	Cladding Thickness	Pellet Diameter	Active Fuel Length	Number of Water Rods	Water Rod OD	Water Rod ID	Channel Thickness	Channel ID
9x9B Assembly Class												
9x9B01	Zr	0.569	72	0.4330	0.0262	0.3737	150	1	1.516	1.459	0.100	5.278
9x9B02	Zr	0.569	72	0.4330	0.0260	0.3737	150	1	1.516	1.459	0.100	5.278
9x9B03	Zr	0.572	72	0.4330	0.0260	0.3737	150	1	1.516	1.459	0.100	5.278
9x9C Assembly Class												
9x9C01	Zr	0.572	80	0.4230	0.0295	0.3565	150	1	0.512	0.472	0.100	5.278
9x9D Assembly Class												
9x9D01	Zr	0.572	79	0.4240	0.0300	0.3565	150	2	0.424	0.364	0.100	5.278
9x9E Assembly Class†												
9x9E01	Zr	0.572	76	0.4170	0.0265	0.3530	150	5	0.546	0.522	0.120	5.215
9x9E02	Zr	0.572	48 28	0.4170 0.4430	0.0265 0.0285	0.3530 0.3745	150	5	0.546	0.522	0.120	5.215

† The 9x9E and 9x9F fuel assembly classes represent a single fuel type containing fuel rods with different dimensions (SPC 9x9-5). In addition to the actual configuration (9x9E02 and 9x9F02), the 9x9E class contains a hypothetical assembly with only small fuel rods (9x9E01), and the 9x9F class contains a hypothetical assembly with only large rods (9x9F01). This was done in order to simplify the specification of this assembly in ~~the CoC~~ *Section 2.1.9*.

Table 6.2.1 (page 6 of 7)
BWR FUEL CHARACTERISTICS AND ASSEMBLY CLASS DEFINITIONS
(all dimensions are in inches)

Fuel Assembly Designation	Clad Material	Pitch	Number of Fuel Rods	Cladding OD	Cladding Thickness	Pellet Diameter	Active Fuel Length	Number of Water Rods	Water Rod OD	Water Rod ID	Channel Thickness	Channel ID
9x9F Assembly Class*												
9x9F01	Zr	0.572	76	0.4430	0.0285	0.3745	150	5	0.546	0.522	0.120	5.215
9x9F02	Zr	0.572	48 28	0.4170 0.4430	0.0265 0.0285	0.3530 0.3745	150	5	0.546	0.522	0.120	5.215
9x9G Assembly Class												
9x9G01	Zr	0.572	72	0.4240	0.0300	0.3565	150	1	1.668	1.604	0.120	5.278
10x10A Assembly Class												
10x10A01	Zr	0.510	92	0.4040	0.0260	0.3450	155	2	0.980	0.920	0.100	5.278
10x10A02	Zr	0.510	78	0.4040	0.0260	0.3450	155	2	0.980	0.920	0.100	5.278
10x10A03	Zr	0.510	92/78	0.4040	0.0260	0.3450	155/90	2	0.980	0.920	0.100	5.278
10x10B Assembly Class												
10x10B01	Zr	0.510	91	0.3957	0.0239	0.3413	155	1	1.378	1.321	0.100	5.278
10x10B02	Zr	0.510	83	0.3957	0.0239	0.3413	155	1	1.378	1.321	0.100	5.278
10x10B03	Zr	0.510	91/83	0.3957	0.0239	0.3413	155/90	1	1.378	1.321	0.100	5.278

* The 9x9E and 9x9F fuel assembly classes represent a single fuel type containing fuel rods with different dimensions (SPC 9x9-5). In addition to the actual configuration (9x9E02 and 9x9F02), the 9x9E class contains a hypothetical assembly with only small fuel rods (9x9E01), and the 9x9F class contains a hypothetical assembly with only large rods (9x9F01). This was done in order to simplify the specification of this assembly in ~~the CoC~~ *Section 2.1.9*.

Table 6.2.1 (page 7 of 7)
BWR FUEL CHARACTERISTICS AND ASSEMBLY CLASS DEFINITIONS
(all dimensions are in inches)

Fuel Assembly Designation	Clad Material	Pitch	Number of Fuel Rods	Cladding OD	Cladding Thickness	Pellet Diameter	Active Fuel Length	Number of Water Rods	Water Rod OD	Water Rod ID	Channel Thickness	Channel ID
10x10C Assembly Class												
10x10C01	Zr	0.488	96	0.3780	0.0243	0.3224	150	5	1.227	1.165	0.055	5.347
10x10D Assembly Class												
10x10D01	SS	0.565	100	0.3960	0.0200	0.3500	83	0	n/a	n/a	0.08	5.663
10x10E Assembly Class												
10x10E01	SS	0.557	96	0.3940	0.0220	0.3430	83	4	0.3940	0.3500	0.08	5.663

Table 6.2.2 (page 1 of 4)
PWR FUEL CHARACTERISTICS AND ASSEMBLY CLASS DEFINITIONS
(all dimensions are in inches)

Fuel Assembly Designation	Clad Material	Pitch	Number of Fuel Rods	Cladding OD	Cladding Thickness	Pellet Diameter	Active Fuel Length	Number of Guide Tubes	Guide Tube OD	Guide Tube ID	Guide Tube Thickness
14x14A Assembly Class											
14x14A01	Zr	0.556	179	0.400	0.0243	0.3444	150	17	0.527	0.493	0.0170
14x14A02	Zr	0.556	179	0.400	0.0243	0.3444	150	17	0.528	0.490	0.0190
14x14A03	Zr	0.556	179	0.400	0.0243	0.3444	150	17	0.526	0.492	0.0170
14x14B Assembly Class											
14x14B01	Zr	0.556	179	0.422	0.0243	0.3659	150	17	0.539	0.505	0.0170
14x14B02	Zr	0.556	179	0.417	0.0295	0.3505	150	17	0.541	0.507	0.0170
14x14B03	Zr	0.556	179	0.424	0.0300	0.3565	150	17	0.541	0.507	0.0170
14x14B04	Zr	0.556	179	0.426	0.0310	0.3565	150	17	0.541	0.507	0.0170
14x14C Assembly Class											
14x14C01	Zr	0.580	176	0.440	0.0280	0.3765	150	5	1.115	1.035	0.0400
14x14C02	Zr	0.580	176	0.440	0.0280	0.3770	150	5	1.115	1.035	0.0400
14x14C03	Zr	0.580	176	0.440	0.0260	0.3805	150	5	1.111	1.035	0.0380
14x14D Assembly Class											
14x14D01	SS	0.556	180	0.422	0.0165	0.3835	144	16	0.543	0.514	0.0145

Table 6.2.2 (page 2 of 4)
PWR FUEL CHARACTERISTICS AND ASSEMBLY CLASS DEFINITIONS
(all dimensions are in inches)

Fuel Assembly Designation	Clad Material	Pitch	Number of Fuel Rods	Cladding OD	Cladding Thickness	Pellet Diameter	Active Fuel Length	Number of Guide Tubes	Guide Tube OD	Guide Tube ID	Guide Tube Thickness
14x14E Assembly Class											
14x14E01†	SS	0.453 and 0.411	162 3 8	0.3415 0.3415 0.3415	0.0120 0.0285 0.0200	0.313 0.280 0.297	102	0	n/a	n/a	n/a
14x14E02†	SS	0.453 and 0.411	173	0.3415	0.0120	0.313	102	0	n/a	n/a	n/a
14x14E03†	SS	0.453 and 0.411	173	0.3415	0.0285	0.0280	102	0	n/a	n/a	n/a
15x15A Assembly Class											
15x15A01	Zr	0.550	204	0.418	0.0260	0.3580	150	21	0.533	0.500	0.0165

† This is the fuel assembly used at Indian Point 1 (IP-1). This assembly is a 14x14 assembly with 23 fuel rods omitted to allow passage of control rods between assemblies. It has a different pitch in different sections of the assembly, and different fuel rod dimensions in some rods.

Table 6.2.2 (page 3 of 4)
PWR FUEL CHARACTERISTICS AND ASSEMBLY CLASS DEFINITIONS
(all dimensions are in inches)

Fuel Assembly Designation	Clad Material	Pitch	Number of Fuel Rods	Cladding OD	Cladding Thickness	Pellet Diameter	Active Fuel Length	Number of Guide Tubes	Guide Tube OD	Guide Tube ID	Guide Tube Thickness
15x15B Assembly Class											
15x15B01	Zr	0.563	204	0.422	0.0245	0.3660	150	21	0.533	0.499	0.0170
15x15B02	Zr	0.563	204	0.422	0.0245	0.3660	150	21	0.546	0.512	0.0170
15x15B03	Zr	0.563	204	0.422	0.0243	0.3660	150	21	0.533	0.499	0.0170
15x15B04	Zr	0.563	204	0.422	0.0243	0.3659	150	21	0.545	0.515	0.0150
15x15B05	Zr	0.563	204	0.422	0.0242	0.3659	150	21	0.545	0.515	0.0150
15x15B06	Zr	0.563	204	0.420	0.0240	0.3671	150	21	0.544	0.514	0.0150
15x15C Assembly Class											
15x15C01	Zr	0.563	204	0.424	0.0300	0.3570	150	21	0.544	0.493	0.0255
15x15C02	Zr	0.563	204	0.424	0.0300	0.3570	150	21	0.544	0.511	0.0165
15x15C03	Zr	0.563	204	0.424	0.0300	0.3565	150	21	0.544	0.511	0.0165
15x15C04	Zr	0.563	204	0.417	0.0300	0.3565	150	21	0.544	0.511	0.0165
15x15D Assembly Class											
15x15D01	Zr	0.568	208	0.430	0.0265	0.3690	150	17	0.530	0.498	0.0160
15x15D02	Zr	0.568	208	0.430	0.0265	0.3686	150	17	0.530	0.498	0.0160
15x15D03	Zr	0.568	208	0.430	0.0265	0.3700	150	17	0.530	0.499	0.0155
15x15D04	Zr	0.568	208	0.430	0.0250	0.3735	150	17	0.530	0.500	0.0150
15x15E Assembly Class											
15x15E01	Zr	0.568	208	0.428	0.0245	0.3707	150	17	0.528	0.500	0.0140
15x15F Assembly Class											
15x15F01	Zr	0.568	208	0.428	0.0230	0.3742	150	17	0.528	0.500	0.0140

Table 6.2.2 (page 4 of 4)
PWR FUEL CHARACTERISTICS AND ASSEMBLY CLASS DEFINITIONS
(all dimensions are in inches)

Fuel Assembly Designation	Clad Material	Pitch	Number of Fuel Rods	Cladding OD	Cladding Thickness	Pellet Diameter	Active Fuel Length	Number of Guide Tubes	Guide Tube OD	Guide Tube ID	Guide Tube Thickness
15x15G Assembly Class											
15x15G01	SS	0.563	204	0.422	0.0165	0.3825	144	21	0.543	0.514	0.0145
15x15H Assembly Class											
15x15H01	Zr	0.568	208	0.414	0.0220	0.3622	150	17	0.528	0.500	0.0140
16x16A Assembly Class											
16x16A01	Zr	0.506	236	0.382	0.0250	0.3255	150	5	0.980	0.900	0.0400
16x16A02	Zr	0.506	236	0.382	0.0250	0.3250	150	5	0.980	0.900	0.0400
17x17A Assembly Class											
~~17x17A01~~	~~Zr~~	~~0.496~~	~~264~~	~~0.360~~	~~0.0225~~	~~0.3088~~	~~144~~	~~25~~	~~0.474~~	~~0.442~~	~~0.0160~~
17x17A0~~2~~*1*	Zr	0.496	264	0.360	0.0225	0.3088	150	25	0.474	0.442	0.0160
17x17A0~~3~~*2*	Zr	0.496	264	0.360	0.0250	0.3030	150	25	0.480	0.448	0.0160
17x17B Assembly Class											
17x17B01	Zr	0.496	264	0.374	0.0225	0.3225	150	25	0.482	0.450	0.0160
17x17B02	Zr	0.496	264	0.374	0.0225	0.3225	150	25	0.474	0.442	0.0160
17x17B03	Zr	0.496	264	0.376	0.0240	0.3215	150	25	0.480	0.448	0.0160
17x17B04	Zr	0.496	264	0.372	0.0205	0.3232	150	25	0.427	0.399	0.0140
17x17B05	Zr	0.496	264	0.374	0.0240	0.3195	150	25	0.482	0.450	0.0160
17x17B06	Zr	0.496	264	0.372	0.0205	0.3232	150	25	0.480	0.452	0.0140
17x17C Assembly Class											
17x17C01	Zr	0.502	264	0.379	0.0240	0.3232	150	25	0.472	0.432	0.0200
17x17C02	Zr	0.502	264	0.377	0.0220	0.3252	150	25	0.472	0.432	0.0200

Table 6.2.3
REACTIVITY EFFECT OF ASSEMBLY PARAMETER VARIATIONS for BWR Fuel in the MPC-68
(all dimensions are in inches)

Fuel Assembly/ Parameter Variation	reactivity effect	calculated k_{eff}	standard deviation	cladding OD	cladding ID	cladding thickness	pellet OD	water rod thickness	channel thickness
8x8C04 (GE8x8R)	reference	0.9307	0.0007	0.483	0.419	0.032	0.410	0.030	0.100
increase pellet OD (+0.001)	+0.0005	0.9312	0.0007	0.483	0.419	0.032	0.411	0.030	0.100
decrease pellet OD (-0.001)	-0.0008	0.9299	0.0009	0.483	0.419	0.032	0.409	0.030	0.100
increase clad ID (+0.004)	+0.0027	0.9334	0.0007	0.483	0.423	0.030	0.410	0.030	0.100
decrease clad ID (-0.004)	-0.0034	0.9273	0.0007	0.483	0.415	0.034	0.410	0.030	0.100
increase clad OD (+0.004)	-0.0041	0.9266	0.0008	0.487	0.419	0.034	0.410	0.030	0.100
decrease clad OD (-0.004)	+0.0023	0.9330	0.0007	0.479	0.419	0.030	0.410	0.030	0.100
increase water rod thickness (+0.015)	-0.0019	0.9288	0.0008	0.483	0.419	0.032	0.410	0.045	0.100
decrease water rod thickness (-0.015)	+0.0001	0.9308	0.0008	0.483	0.419	0.032	0.410	0.015	0.100
remove water rods (i.e., replace the water rod tubes with water)	+0.0021	0.9328	0.0008	0.483	0.419	0.032	0.410	0.000	0.100
remove channel	-0.0039	0.9268	0.0009	0.483	0.419	0.032	0.410	0.030	0.000
increase channel thickness (+0.020)	+0.0005	0.9312	0.0007	0.483	0.419	0.032	0.410	0.030	0.120
reduced active length (120 Inches)	*-0.0007*	*0.9300*	*0.0007*	*0.483*	*0.419*	*0.032*	*0.410*	*0.030*	*0.100*
reduced active length (90 Inches)	*-0.0043*	*0.9264*	*0.0007*	*0.483*	*0.419*	*0.032*	*0.410*	*0.030*	*0.100*

Table 6.2.4
REACTIVITY EFFECT OF ASSEMBLY PARAMETER VARIATIONS in PWR Fuel in the MPC 24 with 400ppm soluble boron concentration
(all dimensions are in inches)

Fuel Assembly/ Parameter Variation	reactivity effect	calculated k_{eff}	standard deviation	cladding OD	Cladding ID	cladding thickness	pellet OD	guide tube thickness
15x15F (15x15 B&W, 5.0% E)	reference	0.9271	0.0005	0.4280	0.3820	0.0230	0.3742	0.0140
increase pellet OD (+0.001)	-0.0008	0.9263	0.0004	0.4280	0.3820	0.0230	0.3752	0.0140
decrease pellet OD (-0.001)	-0.0002	0.9269	0.0005	0.4280	0.3820	0.0230	0.3732	0.0140
increase clad ID (+0.004)	+0.0040	0.9311	0.0005	0.4280	0.3860	0.0210	0.3742	0.0140
decrease clad ID (-0.004)	-0.0033	0.9238	0.0004	0.4280	0.3780	0.0250	0.3742	0.0140
increase clad OD (+0.004)	-0.0042	0.9229	0.0004	0.4320	0.3820	0.0250	0.3742	0.0140
decrease clad OD (-0.004)	+0.0035	0.9306	0.0005	0.4240	0.3820	0.0210	0.3742	0.0140
increase guide tube thickness (+0.004)	-0.0008	0.9263	0.0005	0.4280	0.3820	0.0230	0.3742	0.0180
decrease guide tube thickness (-0.004)	+0.0006	0.9277	0.0004	0.4280	0.3820	0.0230	0.3742	0.0100
remove guide tubes (i.e., replace the guide tubes with water)	+0.0028	0.9299	0.0004	0.4280	0.3820	0.0230	0.3742	0.000
voided guide tubes	-0.0318	0.8953	0.0005	0.4280	0.3820	0.0230	0.3742	0.0140

Table 6.2.5
REACTIVITY EFFECT OF ASSEMBLY PARAMETER VARIATIONS in PWR Fuel in the MPC-32 with 2600ppm soluble boron concentration
(all dimensions are in inches)

Fuel Assembly/ Parameter Variation	reactivity effect	calculated k_{eff}	standard deviation	cladding OD	cladding ID	cladding thickness	pellet OD	guide tube thickness
15x15F (15x15 B&W, 5.0% E)	reference	0.9389	0.0004	0.4280	0.3820	0.0230	0.3742	0.0140
increase pellet OD (+0.001)	+0.0019	0.9408	0.0004	0.4280	0.3820	0.0230	0.3752	0.0140
decrease pellet OD (-0.001)	0.0000	0.9389	0.0004	0.4280	0.3820	0.0230	0.3732	0.0140
increase clad ID (+0.004)	+0.0015	0.9404	0.0004	0.4280	0.3860	0.0210	0.3742	0.0140
decrease clad ID (-0.004)	-0.0015	0.9374	0.0004	0.4280	0.3780	0.0250	0.3742	0.0140
increase clad OD (+0.004)	-0.0002	0.9387	0.0004	0.4320	0.3820	0.0250	0.3742	0.0140
decrease clad OD (-0.004)	+0.0007	0.9397	0.0004	0.4240	0.3820	0.0210	0.3742	0.0140
increase guide tube thickness (+0.004)	-0.0003	0.9387	0.0004	0.4280	0.3820	0.0230	0.3742	0.0180
decrease guide tube thickness (-0.004)	-0.0005	0.9384	0.0004	0.4280	0.3820	0.0230	0.3742	0.0100
remove guide tubes (i.e., replace the guide tubes with water)	-0.0005	0.9385	0.0004	0.4280	0.3820	0.0230	0.3742	0.000
voided guide tubes	+0.0039	0.9428	0.0004	0.4280	0.3820	0.0230	0.3742	0.0140

Table 6.2. 6
MAXIMUM K_{EFF} VALUES FOR THE 14X14A ASSEMBLY CLASS IN THE MPC-24
(all dimensions are in inches)

14x14A (4.6% Enrichment, ~~Boral~~*fixed neutron absorber* ^{10}B minimum loading of 0.02 g/cm^2) 179 fuel rods, 17 guide tubes, pitch=0.556, Zr clad									
Fuel Assembly Designation	maximum k_{eff}	calculated k_{eff}	standard deviation	cladding OD	cladding ID	cladding thickness	pellet OD	fuel length	guide tube thickness
14x14A01	0.9295	0.9252	0.0008	0.400	0.3514	0.0243	0.3444	150	0.017
14x14A02	0.9286	0.9242	0.0008	0.400	0.3514	0.0243	0.3444	150	0.019
14x14A03	0.9296	0.9253	0.0008	0.400	0.3514	0.0243	0.3444	150	0.017
Dimensions Listed *for Authorized Contents*				0.400 (min.)	0.3514 (max.)		0.3444 (max.)	150 (max.)	0.017 (min.)
bounding dimensions (14x14A03)	0.9296	0.9253	0.0008	0.400	0.3514	0.0243	0.3444	150	0.017

Table 6.2.7
MAXIMUM K_{EFF} VALUES FOR THE 14X14B ASSEMBLY CLASS IN THE MPC-24
(all dimensions are in inches)

14x14B (4.6% Enrichment, ~~Boral~~*fixed neutron absorber* ^{10}B minimum loading of 0.02 g/cm^2) 179 fuel rods, 17 guide tubes, pitch=0.556, Zr clad									
Fuel Assembly Designation	maximum k_{eff}	calculated k_{eff}	standard deviation	cladding OD	cladding ID	cladding thickness	pellet OD	fuel length	guide tube thickness
14x14B01	0.9159	0.9117	0.0007	0.422	0.3734	0.0243	0.3659	150	0.017
14x14B02	0.9169	0.9126	0.0008	0.417	0.3580	0.0295	0.3505	150	0.017
14x14B03	0.9110	0.9065	0.0009	0.424	0.3640	0.0300	0.3565	150	0.017
14x14B04	0.9084	0.9039	0.0009	0.426	0.3640	0.0310	0.3565	150	0.017
Dimensions Listed *for Authorized Contents*				0.417 (min.)	0.3734 (max.)		0.3659 (max.)	150 (max.)	0.017 (min.)
bounding dimensions (B14x14B01)	0.9228	0.9185	0.0008	0.417	0.3734	0.0218	0.3659	150	0.017

Table 6.2.8
MAXIMUM K_{EFF} VALUES FOR THE 14X14C ASSEMBLY CLASS IN THE MPC-24
(all dimensions are in inches)

14x14C (4.6% Enrichment, ~~Boral~~*fixed neutron absorber* ^{10}B minimum loading of 0.02 g/cm^2) 176 fuel rods, 5 guide tubes, pitch=0.580, Zr clad									
Fuel Assembly Designation	maximum k_{eff}	calculated k_{eff}	standard deviation	cladding OD	Cladding ID	cladding thickness	pellet OD	fuel length	guide tube thickness
14x14C01	0.9258	0.9215	0.0008	0.440	0.3840	0.0280	0.3765	150	0.040
14x14C02	0.9265	0.9222	0.0008	0.440	0.3840	0.0280	0.3770	150	0.040
14x14C03	0.9287	0.9242	0.0009	0.440	0.3880	0.0260	0.3805	150	0.038
Dimensions Listed *for Authorized Contents*				0.440 (min.)	0.3880 (max.)		0.3805 (max.)	150 (max.)	0.038 (min.)
bounding dimensions (14x14C03)	0.9287	0.9242	0.0009	0.440	0.3880	0.0260	0.3805	150	0.038

Table 6.2.9
MAXIMUM K_{EFF} VALUES FOR THE 14X14D ASSEMBLY CLASS IN THE MPC-24
(all dimensions are in inches)

14x14D (4.0% Enrichment, ~~Boral~~*fixed neutron absorber* ^{10}B minimum loading of 0.02 g/cm^2) 180 fuel rods, 16 guide tubes, pitch=0.556, SS clad									
Fuel Assembly Designation	maximum k_{eff}	calculated k_{eff}	standard deviation	cladding OD	Cladding ID	cladding thickness	pellet OD	fuel length	guide tube thickness
14x14D01	0.8507	0.8464	0.0008	0.422	0.3890	0.0165	0.3835	144	0.0145
Dimensions Listed *for Authorized Contents*				0.422 (min.)	0.3890 (max.)		0.3835 (max.)	144 (max.)	0.0145 (min.)

Table 6.2.10
MAXIMUM K_{EFF} VALUES FOR THE 14X14E ASSEMBLY CLASS IN THE MPC-24
(all dimensions are in inches)

14x14E (5.0% Enrichment, ~~Boral~~*fixed neutron absorber* ^{10}B minimum loading of 0.02 g/cm^2) 173 fuel rods, 0 guide tubes, pitch=0.453 and 0.441, SS clad†									
Fuel Assembly Designation	maximum k_{eff}	calculated k_{eff}	standard deviation	cladding OD	cladding ID	cladding thickness	pellet OD	fuel length††	guide tube thickness
14x14E01	0.7598	0.7555	0.0008	0.3415	0.3175 0.2845 0.3015	0.0120 0.0285 0.0200	0.3130 0.2800 0.2970	102	0.0000
14x14E02	0.7627	0.7586	0.0007	0.3415	0.3175	0.0120	0.3130	102	0.0000
14x14E03	0.6952	0.6909	0.0008	0.3415	0.2845	0.0285	0.2800	102	0.0000
Dimensions Listed *for Authorized Contents*				0.3415 (min.)	0.3175 (max.)		0.3130 (max.)	102 (max.)	0.0000 (min.)
Bounding dimensions (14x14E02)	0.7627	0.7586	0.0007	0.3415	0.3175	0.0120	0.3130	102	0.0000

† This is the IP-1 fuel assembly at Indian Point. This assembly is a 14x14 assembly with 23 fuel rods omitted to allow passage of control rods between assemblies. Fuel rod dimensions are bounding for each of the three types of rods found in the IP-1 fuel assembly.
†† Calculations were conservatively performed for a fuel length of 150 inches.

Table 6.2.11
MAXIMUM K_{EFF} VALUES FOR THE 15X15A ASSEMBLY CLASS IN THE MPC-24
(all dimensions are in inches)

15x15A (4.1% Enrichment, ~~Boral~~*fixed neutron absorber* ^{10}B minimum loading of 0.02 g/cm^2) 204 fuel rods, 21 guide tubes, pitch=0.550, Zr clad									
Fuel Assembly Designation	maximum k_{eff}	calculated k_{eff}	standard deviation	cladding OD	cladding ID	cladding thickness	pellet OD	fuel length	guide tube thickness
15x15A01	0.9204	0.9159	0.0009	0.418	0.3660	0.0260	0.3580	150	0.0165
Dimensions Listed *for Authorized Contents*				0.418 (min.)	0.3660 (max.)		0.3580 (max.)	150 (max.)	0.0165 (min.)

Table 6.2.12
MAXIMUM K_{EFF} VALUES FOR THE 15X15B ASSEMBLY CLASS IN THE MPC-24
(all dimensions are in inches)

15x15B (4.1% Enrichment, ~~Boral~~*fixed neutron absorber* ^{10}B minimum loading of 0.02 g/cm^2) 204 fuel rods, 21 guide tubes, pitch=0.563, Zr clad									
Fuel Assembly Designation	maximum k_{eff}	calculated k_{eff}	standard deviation	cladding OD	cladding ID	cladding thickness	pellet OD	fuel length	guide tube thickness
15x15B01	0.9369	0.9326	0.0008	0.422	0.3730	0.0245	0.3660	150	0.017
15x15B02	0.9338	0.9295	0.0008	0.422	0.3730	0.0245	0.3660	150	0.017
15x15B03	0.9362	0.9318	0.0008	0.422	0.3734	0.0243	0.3660	150	0.017
15x15B04	0.9370	0.9327	0.0008	0.422	0.3734	0.0243	0.3659	150	0.015
15x15B05	0.9356	0.9313	0.0008	0.422	0.3736	0.0242	0.3659	150	0.015
15x15B06	0.9366	0.9324	0.0007	0.420	0.3720	0.0240	0.3671	150	0.015
Dimensions Listed *for Authorized Contents*				0.420 (min.)	0.3736 (max.)		0.3671 (max.)	150 (max.)	0.015 (min.)
bounding dimensions (B15x15B01)	0.9388	0.9343	0.0009	0.420	0.3736	0.0232	0.3671	150	0.015

Table 6.2.13
MAXIMUM K_{EFF} VALUES FOR THE 15X15C ASSEMBLY CLASS IN THE MPC-24
(all dimensions are in inches)

15x15C (4.1% Enrichment, ~~Boral~~*fixed neutron absorber* ^{10}B minimum loading of 0.02 g/cm^2) 204 fuel rods, 21 guide tubes, pitch=0.563, Zr clad									
Fuel Assembly Designation	maximum k_{eff}	calculated k_{eff}	standard deviation	cladding OD	cladding ID	cladding thickness	pellet OD	fuel length	guide tube thickness
15x15C01	0.9255	0.9213	0.0007	0.424	0.3640	0.0300	0.3570	150	0.0255
15x15C02	0.9297	0.9255	0.0007	0.424	0.3640	0.0300	0.3570	150	0.0165
15x15C03	0.9297	0.9255	0.0007	0.424	0.3640	0.0300	0.3565	150	0.0165
15x15C04	0.9311	0.9268	0.0008	0.417	0.3570	0.0300	0.3565	150	0.0165
Dimensions Listed *for Authorized Contents*				0.417 (min.)	0.3640 (max.)		0.3570 (max.)	150 (max.)	0.0165 (min.)
bounding dimensions (B15x15C01)	0.9361	0.9316	0.0009	0.417	0.3640	0.0265	0.3570	150	0.0165

Table 6.2.14
MAXIMUM K_{EFF} VALUES FOR THE 15X15D ASSEMBLY CLASS IN THE MPC-24
(all dimensions are in inches)

15x15D (4.1% Enrichment, ~~Boral~~*fixed neutron absorber* ^{10}B minimum loading of 0.02 g/cm^2) 208 fuel rods, 17 guide tubes, pitch=0.568, Zr clad									
Fuel Assembly Designation	maximum k_{eff}	calculated k_{eff}	standard deviation	cladding OD	cladding ID	cladding thickness	pellet OD	fuel length	guide tube thickness
15x15D01	0.9341	0.9298	0.0008	0.430	0.3770	0.0265	0.3690	150	0.0160
15x15D02	0.9367	0.9324	0.0008	0.430	0.3770	0.0265	0.3686	150	0.0160
15x15D03	0.9354	0.9311	0.0008	0.430	0.3770	0.0265	0.3700	150	0.0155
15x15D04	0.9339	0.9292	0.0010	0.430	0.3800	0.0250	0.3735	150	0.0150
Dimensions Listed *for Authorized Contents*				0.430 (min.)	0.3800 (max.)		0.3735 (max.)	150 (max.)	0.0150 (min.)
bounding dimensions (15x15D04)	0.9339†	0.9292	0.0010	0.430	0.3800	0.0250	0.3735	150	0.0150

† The k_{eff} value listed for the 15x15D02 case is higher than that for the case with the bounding dimensions. Therefore, the 0.9367 (15x15D02) value is listed in Table 6.1.1 as the maximum.

Table 6.2.15
MAXIMUM K_{EFF} VALUES FOR THE 15X15E ASSEMBLY CLASS IN THE MPC-24
(all dimensions are in inches)

15x15E (4.1% Enrichment, ~~Boral~~*fixed neutron absorber* ^{10}B minimum loading of 0.02 g/cm^2) 208 fuel rods, 17 guide tubes, pitch=0.568, Zr clad									
Fuel Assembly Designation	maximum k_{eff}	calculated k_{eff}	standard deviation	cladding OD	cladding ID	cladding thickness	pellet OD	fuel length	guide tube thickness
15x15E01	0.9368	0.9325	0.0008	0.428	0.3790	0.0245	0.3707	150	0.0140
Dimensions Listed *for Authorized Contents*				0.428 (min.)	0.3790 (max.)		0.3707 (max.)	150 (max.)	0.0140 (min.)

Table 6.2.16
MAXIMUM K_{EFF} VALUES FOR THE 15X15F ASSEMBLY CLASS IN THE MPC-24
(all dimensions are in inches)

15x15F (4.1% Enrichment, ~~Boral~~*fixed neutron absorber* ^{10}B minimum loading of 0.02 g/cm^2) 208 fuel rods, 17 guide tubes, pitch=0.568, Zr clad									
Fuel Assembly Designation	maximum k_{eff}	calculated k_{eff}	standard deviation	cladding OD	cladding ID	cladding thickness	pellet OD	fuel length	guide tube thickness
15x15F01	0.9395†	0.9350	0.0009	0.428	0.3820	0.0230	0.3742	150	0.0140
Dimensions Listed *for Authorized Contents*				0.428 (min.)	0.3820 (max.)		0.3742 (max.)	150 (max.)	0.0140 (min.)

† KENO5a verification calculation resulted in a maximum k_{eff} of 0.9383.

Table 6.2.17
MAXIMUM K_{EFF} VALUES FOR THE 15X15G ASSEMBLY CLASS IN THE MPC-24
(all dimensions are in inches)

15x15G (4.0% Enrichment, ~~Boral~~*fixed neutron absorber* ^{10}B minimum loading of 0.02 g/cm^2) 204 fuel rods, 21 guide tubes, pitch=0.563, SS clad									
Fuel Assembly Designation	maximum k_{eff}	calculated k_{eff}	standard deviation	cladding OD	cladding ID	cladding thickness	pellet OD	fuel length	guide tube thickness
15x15G01	0.8876	0.8833	0.0008	0.422	0.3890	0.0165	0.3825	144	0.0145
Dimensions Listed *for Authorized Contents*				0.422 (min.)	0.3890 (max.)		0.3825 (max.)	144 (max.)	0.0145 (min.)

Table 6.2.18
MAXIMUM K_{EFF} VALUES FOR THE 15X15H ASSEMBLY CLASS IN THE MPC-24
(all dimensions are in inches)

15x15H (3.8% Enrichment, ~~Boral~~*fixed neutron absorber* ^{10}B minimum loading of 0.02 g/cm^2) 208 fuel rods, 17 guide tubes, pitch=0.568, Zr clad									
Fuel Assembly Designation	maximum k_{eff}	calculated k_{eff}	standard deviation	cladding OD	cladding ID	cladding thickness	pellet OD	fuel length	guide tube thickness
15x15H01	0.9337	0.9292	0.0009	0.414	0.3700	0.0220	0.3622	150	0.0140
Dimensions Listed *for Authorized Contents*				0.414 (min.)	0.3700 (max.)		0.3622 (max.)	150 (max.)	0.0140 (min.)

Table 6.2.19
MAXIMUM K_{EFF} VALUES FOR THE 16X16A ASSEMBLY CLASS IN THE MPC-24
(all dimensions are in inches)

16x16A (4.6% Enrichment, ~~Boral~~*fixed neutron absorber* ^{10}B minimum loading of 0.02 g/cm^2) 236 fuel rods, 5 guide tubes, pitch=0.506, Zr clad									
Fuel Assembly Designation	maximum k_{eff}	calculated k_{eff}	standard deviation	cladding OD	cladding ID	cladding thickness	pellet OD	fuel length	guide tube thickness
16x16A01	0.9287	0.9244	0.0008	0.382	0.3320	0.0250	0.3255	150	0.0400
16x16A02	0.9263	0.9221	0.0007	0.382	0.3320	0.0250	0.3250	150	0.0400
Dimensions Listed *for Authorized Contents*				0.382 (min.)	0.3320 (max.)		0.3255 (max.)	150 (max.)	0.0400 (min.)
bounding dimensions (16x16A01)	0.9287	0.9244	0.0008	0.382	0.3320	0.0250	0.3255	150	0.0400

Table 6.2.20
MAXIMUM K_{EFF} VALUES FOR THE 17X17A ASSEMBLY CLASS IN THE MPC-24
(all dimensions are in inches)

17x17A (4.0% Enrichment, ~~Boral~~*fixed neutron absorber* ^{10}B minimum loading of 0.02 g/cm^2) 264 fuel rods, 25 guide tubes, pitch=0.496, Zr clad									
Fuel Assembly Designation	maximum k_{eff}	calculated k_{eff}	standard deviation	cladding OD	cladding ID	cladding thickness	pellet OD	fuel length	guide tube thickness
~~17x17A01~~	~~0.9368~~	~~0.9325~~	~~0.0008~~	~~0.360~~	~~0.3150~~	~~0.0225~~	~~0.3088~~	~~144~~	~~0.016~~
17x17A0~~2~~*1*	0.9368	0.9325	0.0008	0.360	0.3150	0.0225	0.3088	150	0.016
17x17A0~~3~~*2*	0.9329	0.9286	0.0008	0.360	0.3100	0.0250	0.3030	150	0.016
Dimensions Listed *for Authorized Contents*				0.360 (min.)	0.3150 (max.)		0.3088 (max.)	150 (max.)	0.016 (min.)
bounding dimensions (17x17A0~~2~~*1*)	0.9368	0.9325	0.0008	0.360	0.3150	0.0225	0.3088	150	0.016

Table 6.2.21
MAXIMUM K_{EFF} VALUES FOR THE 17X17B ASSEMBLY CLASS IN THE MPC-24
(all dimensions are in inches)

17x17B (4.0% Enrichment, ~~Boral~~*fixed neutron absorber* ^{10}B minimum loading of 0.02 g/cm^2) 264 fuel rods, 25 guide tubes, pitch=0.496, Zr clad									
Fuel Assembly Designation	maximum k_{eff}	calculated k_{eff}	standard deviation	Cladding OD	cladding ID	cladding thickness	pellet OD	fuel length	guide tube thickness
17x17B01	0.9288	0.9243	0.0009	0.374	0.3290	0.0225	0.3225	150	0.016
17x17B02	0.9290	0.9247	0.0008	0.374	0.3290	0.0225	0.3225	150	0.016
17x17B03	0.9243	0.9199	0.0008	0.376	0.3280	0.0240	0.3215	150	0.016
17x17B04	0.9324	0.9279	0.0009	0.372	0.3310	0.0205	0.3232	150	0.014
17x17B05	0.9266	0.9222	0.0008	0.374	0.3260	0.0240	0.3195	150	0.016
17x17B06	0.9311	0.9268	0.0008	0.372	0.3310	0.0205	0.3232	150	0.014
Dimensions Listed *for Authorized Contents*				0.372 (min.)	0.3310 (max.)		0.3232 (max.)	150 (max.)	0.014 (min.)
bounding dimensions (17x17B06)	0.9311†	0.9268	0.0008	0.372	0.3310	0.0205	0.3232	150	0.014

† The k_{eff} value listed for the 17x17B04 case is higher than that for the case with the bounding dimensions. Therefore, the 0.9324 (17x17B04) value is listed in Table 6.1.1 as the maximum.

Table 6.2.22
MAXIMUM K_{EFF} VALUES FOR THE 17X17C ASSEMBLY CLASS IN THE MPC-24
(all dimensions are in inches)

17x17C (4.0% Enrichment, ~~Boral~~*fixed neutron absorber* ^{10}B minimum loading of 0.02 g/cm^2) 264 fuel rods, 25 guide tubes, pitch=0.502, Zr clad									
Fuel Assembly Designation	maximum k_{eff}	calculated k_{eff}	standard deviation	cladding OD	cladding ID	cladding thickness	pellet OD	fuel length	guide tube thickness
17x17C01	0.9293	0.9250	0.0008	0.379	0.3310	0.0240	0.3232	150	0.020
17x17C02	0.9336	0.9293	0.0008	0.377	0.3330	0.0220	0.3252	150	0.020
Dimensions Listed *for Authorized Contents*				0.377 (min.)	0.3330 (max.)		0.3252 (max.)	150 (max.)	0.020 (min.)
bounding dimensions (17x17C02)	0.9336	0.9293	0.0008	0.377	0.3330	0.0220	0.3252	150	0.020

Table 6.2.23
MAXIMUM K_{EFF} VALUES FOR THE 7X7B ASSEMBLY CLASS IN THE MPC-68 and MPC-68FF
(all dimensions are in inches)

7x7B (4.2% Enrichment, ~~Boral~~*fixed neutron absorber* ^{10}B minimum loading of 0.0279 g/cm^2) 49 fuel rods, 0 water rods, pitch=0.738, Zr clad										
Fuel Assembly Designation	maximum k_{eff}	calculated k_{eff}	standard deviation	cladding OD	cladding ID	cladding thickness	pellet OD	fuel length	water rod thickness	channel thickness
7x7B01	0.9372	0.9330	0.0007	0.5630	0.4990	0.0320	0.4870	150	n/a	0.080
7x7B02	0.9301	0.9260	0.0007	0.5630	0.4890	0.0370	0.4770	150	n/a	0.102
7x7B03	0.9313	0.9271	0.0008	0.5630	0.4890	0.0370	0.4770	150	n/a	0.080
7x7B04	0.9311	0.9270	0.0007	0.5700	0.4990	0.0355	0.4880	150	n/a	0.080
7x7B05	0.9350	0.9306	0.0008	0.5630	0.4950	0.0340	0.4775	150	n/a	0.080
7x7B06	0.9298	0.9260	0.0006	0.5700	0.4990	0.0355	0.4910	150	n/a	0.080
Dimensions Listed *for Authorized Contents*				0.5630 (min.)	0.4990 (max.)		0.4910 (max.)	150 (max.)	n/a	0.120 (max.)
bounding dimensions (B7x7B01)	0.9375	0.9332	0.0008	0.5630	0.4990	0.0320	0.4910	150	n/a	0.102
bounding dimensions with 120 mil channel (B7x7B02)	0.9386	0.9344	0.0007	0.5630	0.4990	0.0320	0.4910	150	n/a	0.120

Table 6.2.24
MAXIMUM K_{EFF} VALUES FOR THE 8X8B ASSEMBLY CLASS IN THE MPC-68 and MPC-68FF
(all dimensions are in inches)

8x8B (4.2% Enrichment, ~~Boral~~*fixed neutron absorber* ^{10}B minimum loading of 0.0279 g/cm^2) 63 or 64 fuel rods†, 1 or 0 water rods†, pitch† = 0.636-0.642, Zr clad												
Fuel Assembly Designation	maximum k_{eff}	calculated k_{eff}	standard deviation	Fuel rods	Pitch	cladding OD	cladding ID	cladding thickness	pellet OD	fuel length	water rod thickness	channel thickness
8x8B01	0.9310	0.9265	0.0009	63	0.641	0.4840	0.4140	0.0350	0.4050	150	0.035	0.100
8x8B02	0.9227	0.9185	0.0007	63	0.636	0.4840	0.4140	0.0350	0.4050	150	0.035	0.100
8x8B03	0.9299	0.9257	0.0008	63	0.640	0.4930	0.4250	0.0340	0.4160	150	0.034	0.100
8x8B04	0.9236	0.9194	0.0008	64	0.642	0.5015	0.4295	0.0360	0.4195	150	n/a	0.100
Dimensions Listed *for Authorized Contents*				63 or 64	0.636-0.642	0.4840 (min.)	0.4295 (max.)		0.4195 (max.)	150 (max.)	0.034	0.120 (max.)
bounding (pitch=0.636) (B8x8B01)	0.9346	0.9301	0.0009	63	0.636	0.4840	0.4295	0.02725	0.4195	150	0.034	0.120
bounding (pitch=0.640) (B8x8B02)	0.9385	0.9343	0.0008	63	0.640	0.4840	0.4295	0.02725	0.4195	150	0.034	0.120
bounding (pitch=0.642) (B8x8B03)	0.9416	0.9375	0.0007	63	0.642	0.4840	0.4295	0.02725	0.4195	150	0.034	0.120

† This assembly class was analyzed and qualified for a small variation in the pitch and a variation in the number of fuel and water rods.

Table 6.2.25
MAXIMUM K_{EFF} VALUES FOR THE 8X8C ASSEMBLY CLASS IN THE MPC-68 and MPC-68FF
(all dimensions are in inches)

8x8C (4.2% Enrichment, ~~Boral~~*fixed neutron absorber* ^{10}B minimum loading of 0.0279 g/cm^2) 62 fuel rods, 2 water rods, pitch† = 0.636-0.641, Zr clad											
Fuel Assembly Designation	maximum k_{eff}	calculated k_{eff}	standard deviation	pitch	cladding OD	cladding ID	cladding thickness	pellet OD	fuel length	water rod thickness	channel thickness
8x8C01	0.9315	0.9273	0.0007	0.641	0.4840	0.4140	0.0350	0.4050	150	0.035	0.100
8x8C02	0.9313	0.9268	0.0009	0.640	0.4830	0.4190	0.0320	0.4100	150	0.030	0.000
8x8C03	0.9329	0.9286	0.0008	0.640	0.4830	0.4190	0.0320	0.4100	150	0.030	0.800
8x8C04	0.9348††	0.9307	0.0007	0.640	0.4830	0.4190	0.0320	0.4100	150	0.030	0.100
8x8C05	0.9353	0.9312	0.0007	0.640	0.4830	0.4190	0.0320	0.4100	150	0.030	0.120
8x8C06	0.9353	0.9312	0.0007	0.640	0.4830	0.4190	0.0320	0.4110	150	0.030	0.100
8x8C07	0.9314	0.9273	0.0007	0.640	0.4830	0.4150	0.0340	0.4100	150	0.030	0.100
8x8C08	0.9339	0.9298	0.0007	0.640	0.4830	0.4190	0.0320	0.4100	150	0.034	0.100
8x8C09	0.9301	0.9260	0.0007	0.640	0.4930	0.4250	0.0340	0.4160	150	0.034	0.100
8x8C10	0.9317	0.9275	0.0008	0.640	0.4830	0.4150	0.0340	0.4100	150	0.030	0.120
8x8C11	0.9328	0.9287	0.0007	0.640	0.4830	0.4150	0.0340	0.4100	150	0.030	0.120
8x8C12	0.9285	0.9242	0.0008	0.636	0.4830	0.4190	0.0320	0.4110	150	0.030	0.120
Dimensions Listed *for Authorized Contents*				0.636-0.641	0.4830 (min.)	0.4250 (max.)		0.4160 (max.)	150 (max.)	0.000 (min.)	0.120 (max.)
bounding (pitch=0.636) (B8x8C01)	0.9357	0.9313	0.0009	0.636	0.4830	0.4250	0.0290	0.4160	150	0.000	0.120
bounding (pitch=0.640) (B8x8C02)	0.9425	0.9384	0.0007	0.640	0.4830	0.4250	0.0290	0.4160	150	0.000	0.120
Bounding (pitch=0.641) (B8x8C03)	0.9418	0.9375	0.0008	0.641	0.4830	0.4250	0.0290	0.4160	150	0.000	0.120

† This assembly class was analyzed and qualified for a small variation in the pitch.
†† KENO5a verification calculation resulted in a maximum k_{eff} of 0.9343.

Table 6.2.26
MAXIMUM K_{EFF} VALUES FOR THE 8X8D ASSEMBLY CLASS IN THE MPC-68 and MPC-68FF
(all dimensions are in inches)

8x8D (4.2% Enrichment, ~~Boral~~*fixed neutron absorber* ^{10}B minimum loading of 0.0279 g/cm^2) 60-61 fuel rods, 1-4 water rods†, pitch=0.640, Zr clad										
Fuel Assembly Designation	maximum k_{eff}	calculated k_{eff}	standard deviation	Cladding OD	cladding ID	cladding thickness	pellet OD	fuel length	water rod thickness	channel thickness
8x8D01	0.9342	0.9302	0.0006	0.4830	0.4190	0.0320	0.4110	150	0.03/0.025	0.100
8x8D02	0.9325	0.9284	0.0007	0.4830	0.4190	0.0320	0.4110	150	0.030	0.100
8x8D03	0.9351	0.9309	0.0008	0.4830	0.4190	0.0320	0.4110	150	0.025	0.100
8x8D04	0.9338	0.9296	0.0007	0.4830	0.4190	0.0320	0.4110	150	0.040	0.100
8x8D05	0.9339	0.9294	0.0009	0.4830	0.4190	0.0320	0.4100	150	0.040	0.100
8x8D06	0.9365	0.9324	0.0007	0.4830	0.4190	0.0320	0.4110	150	0.040	0.120
8x8D07	0.9341	0.9297	0.0009	0.4830	0.4190	0.0320	0.4110	150	0.040	0.080
8x8D08	0.9376	0.9332	0.0009	0.4830	0.4230	0.0300	0.4140	150	0.030	0.080
Dimensions Listed *for Authorized Contents*				0.4830 (min.)	0.4230 (max.)		0.4140 (max.)	150 (max.)	0.000 (min.)	0.120 (max.)
bounding dimensions (B8x8D01)	0.9403	0.9363	0.0007	0.4830	0.4230	0.0300	0.4140	150	0.000	0.120

† Fuel assemblies 8x8D01 through 8x8D03 have 4 water rods that are similar in size to the fuel rods, while assemblies 8x8D04 through 8x8D07 have 1 large water rod that takes the place of the 4 water rods. Fuel assembly 8x8D08 contains 3 water rods that are similar in size to the fuel rods.

Table 6.2.27
MAXIMUM K_{EFF} VALUES FOR THE 8X8E ASSEMBLY CLASS IN THE MPC-68 and MPC-68FF
(all dimensions are in inches)

8x8E (4.2% Enrichment, ~~Boral~~*fixed neutron absorber* ^{10}B minimum loading of 0.0279 g/cm^2) 59 fuel rods, 5 water rods, pitch=0.640, Zr clad										
Fuel Assembly Designation	maximum k_{eff}	calculated k_{eff}	standard deviation	cladding OD	cladding ID	cladding thickness	pellet OD	fuel length	water rod thickness	channel thickness
8x8E01	0.9312	0.9270	0.0008	0.4930	0.4250	0.0340	0.4160	150	0.034	0.100
Dimensions Listed *for Authorized Contents*				0.4930 (min.)	0.4250 (max.)		0.4160 (max.)	150 (max.)	0.034 (min.)	0.100 (max.)

Table 6.2.28
MAXIMUM K_{EFF} VALUES FOR THE 8X8F ASSEMBLY CLASS IN THE MPC-68 and MPC-68FF
(all dimensions are in inches)

8x8F (4.0% Enrichment, ~~Boral~~*fixed neutron absorber* ^{10}B minimum loading of 0.0279 g/cm^2) 64 fuel rods, 4 rectangular water cross segments dividing the assembly into four quadrants, pitch=0.609, Zr clad										
Fuel Assembly Designation	maximum k_{eff}	calculated k_{eff}	standard deviation	cladding OD	cladding ID	cladding thickness	pellet OD	fuel length	water rod thickness	channel thickness
8x8F01	0.9411	0.9366	0.0009	0.4576	0.3996	0.0290	0.3913	150	0.0315	0.055
Dimensions Listed *for Authorized Contents*				0.4576 (min.)	0.3996 (max.)		0.3913 (max.)	150 (max.)	0.0315 (min.)	0.055 (max.)

Table 6.2.29
MAXIMUM K_{EFF} VALUES FOR THE 9X9A ASSEMBLY CLASS IN THE MPC-68 and MPC-68FF
(all dimensions are in inches)

9x9A (4.2% Enrichment, ~~Boral~~*fixed neutron absorber* ^{10}B minimum loading of 0.0279 g/cm^2) 74/66 fuel rods†, 2 water rods, pitch=0.566, Zr clad										
Fuel Assembly Designation	maximum k_{eff}	calculated k_{eff}	standard deviation	cladding OD	cladding ID	cladding thickness	pellet OD	fuel length	water rod thickness	channel thickness
9x9A01 (axial segment with all rods)	0.9353	0.9310	0.0008	0.4400	0.3840	0.0280	0.3760	150	0.030	0.100
9x9A02 (axial segment with only the full length rods)	0.9388	0.9345	0.0008	0.4400	0.3840	0.0280	0.3760	150	0.030	0.100
9x9A03 (actual three-dimensional representation of all rods)	0.9351	0.9310	0.0007	0.4400	0.3840	0.0280	0.3760	150/90	0.030	0.100
9x9A04 (axial segment with only the full length rods)	0.9396	0.9355	0.0007	0.4400	0.3840	0.0280	0.3760	150	0.030	0.120
Dimensions Listed *for Authorized Contents*				0.4400 (min.)	0.3840 (max.)		0.3760 (max.)	150 (max.)	0.000 (min.)	0.120 (max.)
bounding dimensions (axial segment with only the full length rods) (B9x9A01)	0.9417	0.9374	0.0008	0.4400	0.3840	0.0280	0.3760	150	0.000	0.120

† This assembly class contains 66 full length rods and 8 partial length rods. In order to eliminate a requirement on the length of the partial length rods, separate calculations were performed for the axial segments with and without the partial length rods.

Table 6.2.30
MAXIMUM K_{EFF} VALUES FOR THE 9X9B ASSEMBLY CLASS IN THE MPC-68 and MPC-68FF
(all dimensions are in inches)

9x9B (4.2% Enrichment, ~~Boral~~*fixed neutron absorber* ^{10}B minimum loading of 0.0279 g/cm^2) 72 fuel rods, 1 water rod (square, replacing 9 fuel rods), pitch=0.569 to 0.572†, Zr clad											
Fuel Assembly Designation	maximum k_{eff}	calculated k_{eff}	standard deviation	pitch	cladding OD	cladding ID	cladding thickness	pellet OD	fuel length	water rod thickness	channel thickness
9x9B01	0.9380	0.9336	0.0008	0.569	0.4330	0.3807	0.0262	0.3737	150	0.0285	0.100
9x9B02	0.9373	0.9329	0.0009	0.569	0.4330	0.3810	0.0260	0.3737	150	0.0285	0.100
9x9B03	0.9417	0.9374	0.0008	0.572	0.4330	0.3810	0.0260	0.3737	150	0.0285	0.100
Dimensions Listed *for Authorized Contents*				0.572	0.4330 (min.)	0.3810 (max.)		0.3740 (max.)	150 (max.)	0.000 (min.)	0.120 (max.)
bounding dimensions (B9x9B01)	0.9436	0.9394	0.0008	0.572	0.4330	0.3810	0.0260	0.3740††	150	0.000	0.120

† This assembly class was analyzed and qualified for a small variation in the pitch.
†† This value was conservatively defined to be larger than any of the actual pellet diameters.

Table 6.2.31
MAXIMUM K_{EFF} VALUES FOR THE 9X9C ASSEMBLY CLASS IN THE MPC-68 and MPC-68FF
(all dimensions are in inches)

9x9C (4.2% Enrichment, ~~Boral~~*fixed neutron absorber* ^{10}B minimum loading of 0.0279 g/cm^2) 80 fuel rods, 1 water rods, pitch=0.572, Zr clad										
Fuel Assembly Designation	maximum k_{eff}	calculated k_{eff}	standard deviation	cladding OD	cladding ID	cladding thickness	pellet OD	fuel length	water rod thickness	channel thickness
9x9C01	0.9395	0.9352	0.0008	0.4230	0.3640	0.0295	0.3565	150	0.020	0.100
Dimensions Listed *for Authorized Contents*				0.4230 (min.)	0.3640 (max.)		0.3565 (max.)	150 (max.)	0.020 (min.)	0.100 (max.)

Table 6.2.32
MAXIMUM K_{EFF} VALUES FOR THE 9X9D ASSEMBLY CLASS IN THE MPC-68 and MPC-68FF
(all dimensions are in inches)

9x9D (4.2% Enrichment, ~~Boral~~*fixed neutron absorber* ^{10}B minimum loading of 0.0279 g/cm^2) 79 fuel rods, 2 water rods, pitch=0.572, Zr clad										
Fuel Assembly Designation	maximum k_{eff}	calculated k_{eff}	standard deviation	cladding OD	cladding ID	cladding thickness	pellet OD	fuel length	water rod thickness	channel thickness
9x9D01	0.9394	0.9350	0.0009	0.4240	0.3640	0.0300	0.3565	150	0.0300	0.100
Dimensions Listed *for Authorized Contents*				0.4240 (min.)	0.3640 (max.)		0.3565 (max.)	150 (max.)	0.0300 (min.)	0.100 (max.)

Table 6.2.33
MAXIMUM K_{EFF} VALUES FOR THE 9X9E ASSEMBLY CLASS IN THE MPC-68 and MPC-68FF

(all dimensions are in inches)

9x9E (4.0% Enrichment, ~~Boral~~*fixed neutron absorber* ^{10}B minimum loading of 0.0279 g/cm^2) 76 fuel rods, 5 water rods, pitch=0.572, Zr clad										
Fuel Assembly Designation	maximum k_{eff}	calculated k_{eff}	standard deviation	cladding OD	cladding ID	cladding thickness	pellet OD	fuel length	water rod thickness	channel thickness
9x9E01	0.9334	0.9293	0.0007	0.4170	0.3640	0.0265	0.3530	150	0.0120	0.120
9x9E02	0.9401	0.9359	0.0008	0.4170 0.4430	0.3640 0.3860	0.0265 0.0285	0.3530 0.3745	150	0.0120	0.120
Dimensions Listed *for Authorized Contents*†				0.4170 (min.)	0.3640 (max.)		0.3530 (max.)	150 (max.)	0.0120 (min.)	0.120 (max.)
bounding dimensions (9x9E02)	0.9401	0.9359	0.0008	0.4170 0.4430	0.3640 0.3860	0.0265 0.0285	0.3530 0.3745	150	0.0120	0.120

† This fuel assembly, also known as SPC 9x9-5, contains fuel rods with different cladding and pellet diameters which do not bound each other. To be consistent in the way fuel assemblies are listed ~~in the Certificate of Compliance~~ *for the authorized contents*, two assembly classes (9x9E and 9x9F) are required to specify this assembly. Each class contains the actual geometry (9x9E02 and 9x9F02), as well as a hypothetical geometry with either all small rods (9x9E01) or all large rods (9x9F01). The ~~Certificate of Compliance~~ *Authorized Contents* lists the small rod dimensions for class 9x9E and the large rod dimensions for class 9x9F, and a note that both classes are used to qualify the assembly. The analyses demonstrate that all configurations, including the actual geometry, are acceptable.

Table 6.2.34
MAXIMUM K_{EFF} VALUES FOR THE 9X9F ASSEMBLY CLASS IN THE MPC-68 and MPC-68FF

(all dimensions are in inches)

9x9F (4.0% Enrichment, ~~Boral~~*fixed neutron absorber* ^{10}B minimum loading of 0.0279 g/cm^2) 76 fuel rods, 5 water rods, pitch=0.572, Zr clad										
Fuel Assembly Designation	maximum k_{eff}	calculated k_{eff}	standard deviation	cladding OD	cladding ID	cladding thickness	pellet OD	fuel length	water rod thickness	channel thickness
9x9F01	0.9307	0.9265	0.0007	0.4430	0.3860	0.0285	0.3745	150	0.0120	0.120
9x9F02	0.9401	0.9359	0.0008	0.4170 0.4430	0.3640 0.3860	0.0265 0.0285	0.3530 0.3745	150	0.0120	0.120
Dimensions Listed *for Authorized Contents*†				0.4430 (min.)	0.3860 (max.)		0.3745 (max.)	150 (max.)	0.0120 (min.)	0.120 (max.)
bounding dimensions (9x9F02)	0.9401	0.9359	0.0008	0.4170 0.4430	0.3640 0.3860	0.0265 0.0285	0.3530 0.3745	150	0.0120	0.120

† This fuel assembly, also known as SPC 9x9-5, contains fuel rods with different cladding and pellet diameters which do not bound each other. To be consistent in the way fuel assemblies are listed ~~in the Certificate of Compliance~~ *for the authorized contents*, two assembly classes (9x9E and 9x9F) are required to specify this assembly. Each class contains the actual geometry (9x9E02 and 9x9F02), as well as a hypothetical geometry with either all small rods (9x9E01) or all large rods (9x9F01). The ~~Certificate of Compliance~~ *Authorized Contents* lists the small rod dimensions for class 9x9E and the large rod dimensions for class 9x9F, and a note that both classes are used to qualify the assembly. The analyses demonstrate that all configurations, including the actual geometry, are acceptable.

Table 6.2.35
MAXIMUM K_{EFF} VALUES FOR THE 9X9G ASSEMBLY CLASS IN THE MPC-68 and MPC-68FF
(all dimensions are in inches)

9x9G (4.2% Enrichment, ~~Boral~~*fixed neutron absorber* ^{10}B minimum loading of 0.0279 g/cm^2) 72 fuel rods, 1 water rod (square, replacing 9 fuel rods), pitch=0.572, Zr clad										
Fuel Assembly Designation	maximum k_{eff}	calculated k_{eff}	standard deviation	cladding OD	cladding ID	cladding thickness	pellet OD	fuel length	water rod thickness	channel thickness
9x9G01	0.9309	0.9265	0.0008	0.4240	0.3640	0.0300	0.3565	150	0.0320	0.120
Dimensions Listed *for Authorized Contents*				0.4240 (min.)	0.3640 (max.)		0.3565 (max.)	150 (max.)	0.0320 (min.)	0.120 (max.)

Table 6.2.36
MAXIMUM K_{EFF} VALUES FOR THE 10X10A ASSEMBLY CLASS IN THE MPC-68 and MPC-68FF

(all dimensions are in inches)

10x10A (4.2% Enrichment, ~~Boral~~*fixed neutron absorber* ^{10}B minimum loading of 0.0279 g/cm^2) 92/78 fuel rods†, 2 water rods, pitch=0.510, Zr clad										
Fuel Assembly Designation	maximum k_{eff}	calculated k_{eff}	standard deviation	cladding OD	cladding ID	cladding thickness	pellet OD	fuel length	water rod thickness	channel thickness
10x10A01 (axial segment with all rods)	0.9377	0.9335	0.0008	0.4040	0.3520	0.0260	0.3450	155	0.030	0.100
10x10A02 (axial segment with only the full length rods)	0.9426	0.9386	0.0007	0.4040	0.3520	0.0260	0.3450	155	0.030	0.100
10x10A03 (actual three-dimensional representation of all rods)	0.9396	0.9356	0.0007	0.4040	0.3520	0.0260	0.3450	155/90	0.030	0.100
Dimensions Listed *for Authorized Contents*				0.4040 (min.)	0.3520 (max.)		0.3455 (max.)	150†† (max.)	0.030 (min.)	0.120 (max.)
bounding dimensions (axial segment with only the full length rods) (B10x10A01)	0.9457†††	0.9414	0.0008	0.4040	0.3520	0.0260	0.3455‡	155	0.030	0.120

† This assembly class contains 78 full-length rods and 14 partial-length rods. In order to eliminate the requirement on the length of the partial length rods, separate calculations were performed for axial segments with and without the partial length rods.

†† Although the analysis qualifies this assembly for a maximum active fuel length of 155 inches, the ~~Certificate of Compliance~~ *specification for the authorized contents* limits the active fuel length to 150 inches. This is due to the fact that the ~~Boral~~*fixed neutron absorber* panels are 156 inches in length.

††† KENO5a verification calculation resulted in a maximum k_{eff} of 0.9453.

‡ This value was conservatively defined to be larger than any of the actual pellet diameters.

Table 6.2.37
MAXIMUM K_{EFF} VALUES FOR THE 10X10B ASSEMBLY CLASS IN THE MPC-68 and MPC-68FF
(all dimensions are in inches)

10x10B (4.2% Enrichment, ~~Boral~~*fixed neutron absorber* ^{10}B minimum loading of 0.0279 g/cm^2) 91/83 fuel rods†, 1 water rods (square, replacing 9 fuel rods), pitch=0.510, Zr clad										
Fuel Assembly Designation	maximum k_{eff}	calculated k_{eff}	standard deviation	cladding OD	cladding ID	cladding thickness	pellet OD	fuel length	water rod thickness	channel thickness
10x10B01 (axial segment with all rods)	0.9384	0.9341	0.0008	0.3957	0.3480	0.0239	0.3413	155	0.0285	0.100
10x10B02 (axial segment with only the full length rods)	0.9416	0.9373	0.0008	0.3957	0.3480	0.0239	0.3413	155	0.0285	0.100
10x10B03 (actual three-dimensional representation of all rods)	0.9375	0.9334	0.0007	0.3957	0.3480	0.0239	0.3413	155/90	0.0285	0.100
Dimensions Listed *for Authorized Contents*				0.3957 (min.)	0.3480 (max.)		0.3420 (max.)	150†† (max.)	0.000 (min.)	0.120 (max.)
bounding dimensions (axial segment with only the full length rods) (B10x10B01)	0.9436	0.9395	0.0007	0.3957	0.3480	0.0239	0.3420†††	155	0.000	0.120

† This assembly class contains 83 full length rods and 8 partial length rods. In order to eliminate a requirement on the length of the partial length rods, separate calculations were performed for the axial segments with and without the partial length rods.

†† Although the analysis qualifies this assembly for a maximum active fuel length of 155 inches, the ~~Certificate of Compliance~~ *specification for the authorized contents* limits the active fuel length to 150 inches. This is due to the fact that the ~~Boral~~*fixed neutron absorber* panels are 156 inches in length.

††† This value was conservatively defined to be larger than any of the actual pellet diameters.

Table 6.2.38
MAXIMUM K_{EFF} VALUES FOR THE 10X10C ASSEMBLY CLASS IN THE MPC-68 and MPC-68FF

(all dimensions are in inches)

10x10C (4.2% Enrichment, ~~Boral~~*fixed neutron absorber* ^{10}B minimum loading of 0.0279 g/cm^2) 96 fuel rods, 5 water rods (1 center diamond and 4 rectangular), pitch=0.488, Zr clad										
Fuel Assembly Designation	maximum k_{eff}	calculated k_{eff}	standard deviation	cladding OD	cladding ID	cladding thickness	pellet OD	fuel length	water rod thickness	channel thickness
10x10C01	0.9433	0.9392	0.0007	0.3780	0.3294	0.0243	0.3224	150	0.031	0.055
Dimensions Listed *for Authorized Contents*				0.3780 (min.)	0.3294 (max.)		0.3224 (max.)	150 (max.)	0.031 (min.)	0.055 (max.)

Table 6.2.39
MAXIMUM K_{EFF} VALUES FOR THE 10X10D ASSEMBLY CLASS IN THE MPC-68 and MPC-68FF

(all dimensions are in inches)

10x10D (4.0% Enrichment, ~~Boral~~*fixed neutron absorber* ^{10}B minimum loading of 0.0279 g/cm^2) 100 fuel rods, 0 water rods, pitch=0.565, SS clad										
Fuel Assembly Designation	maximum k_{eff}	calculated k_{eff}	standard deviation	cladding OD	cladding ID	cladding thickness	pellet OD	fuel length	water rod thickness	channel thickness
10x10D01	0.9376	0.9333	0.0008	0.3960	0.3560	0.0200	0.350	83	n/a	0.080
Dimensions Listed *for Authorized Contents*				0.3960 (min.)	0.3560 (max.)		0.350 (max.)	83 (max.)	n/a	0.080 (max.)

Table 6.2.40
MAXIMUM K_{EFF} VALUES FOR THE 10X10E ASSEMBLY CLASS IN THE MPC-68 and MPC-68FF

(all dimensions are in inches)

10x10E (4.0% Enrichment, ~~Boral~~*fixed neutron absorber* ^{10}B minimum loading of 0.0279 g/cm^2) 96 fuel rods, 4 water rods, pitch=0.557, SS clad										
Fuel Assembly Designation	maximum k_{eff}	calculated k_{eff}	standard deviation	cladding OD	cladding ID	cladding thickness	pellet OD	fuel length	water rod thickness	channel thickness
10x10E01	0.9185	0.9144	0.0007	0.3940	0.3500	0.0220	0.3430	83	0.022	0.080
Dimensions Listed *for Authorized Contents*				0.3940 (min.)	0.3500 (max.)		0.3430 (max.)	83 (max.)	0.022 (min.)	0.080 (max.)

Table 6.2.41
MAXIMUM K_{EFF} VALUES FOR THE 6X6A ASSEMBLY CLASS IN THE MPC-68F and MPC-68FF
(all dimensions are in inches)

6x6A (3.0% Enrichment†, ~~Boral~~*fixed neutron absorber* ^{10}B minimum loading of 0.0067 g/cm^2) 35 or 36 fuel rods††, 1 or 0 water rods††, pitch††=0.694 to 0.710, Zr clad												
Fuel Assembly Designation	maximum k_{eff}	calculated k_{eff}	standard deviation	pitch	fuel rods	cladding OD	cladding ID	cladding thickness	pellet OD	fuel length	water rod thickness	channel thickness
6x6A01	0.7539	0.7498	0.0007	0.694	36	0.5645	0.4945	0.0350	0.4940	110	n/a	0.060
6x6A02	0.7517	0.7476	0.0007	0.694	36	0.5645	0.4925	0.0360	0.4820	110	n/a	0.060
6x6A03	0.7545	0.7501	0.0008	0.694	36	0.5645	0.4945	0.0350	0.4820	110	n/a	0.060
6x6A04	0.7537	0.7494	0.0008	0.694	36	0.5550	0.4850	0.0350	0.4820	110	n/a	0.060
6x6A05	0.7555	0.7512	0.0008	0.696	36	0.5625	0.4925	0.0350	0.4820	110	n/a	0.060
6x6A06	0.7618	0.7576	0.0008	0.696	35	0.5625	0.4925	0.0350	0.4820	110	0.0	0.060
6x6A07	0.7588	0.7550	0.0005	0.700	36	0.5555	0.4850	0.03525	0.4780	110	n/a	0.060
6x6A08	0.7808	0.7766	0.0007	0.710	36	0.5625	0.5105	0.0260	0.4980	110	n/a	0.060
Dimensions Listed *for Authorized Contents*				0.710 (max.)	35 or 36	0.5550 (min.)	0.5105 (max.)	0.02225	0.4980 (max.)	120 (max.)	0.0	0.060 (max.)
bounding dimensions (B6x6A01)	0.7727	0.7685	0.0007	0.694	35	0.5550	0.5105	0.02225	0.4980	120	0.0	0.060
bounding dimensions (B6x6A02)	0.7782	0.7738	0.0008	0.700	35	0.5550	0.5105	0.02225	0.4980	120	0.0	0.060
bounding dimensions (B6x6A03)	0.7888	0.7846	0.0007	0.710	35	0.5550	0.5105	0.02225	0.4980	120	0.0	0.060

† Although the calculations were performed for 3.0%, the enrichment is limited in the ~~Certificate of Compliance~~ *specification for the authorized contents* to 2.7%.

†† This assembly class was analyzed and qualified for a small variation in the pitch and a variation in the number of fuel and water rods.

Table 6.2.42
MAXIMUM K_{EFF} VALUES FOR THE 6X6B ASSEMBLY CLASS IN THE MPC-68F and MPC-68FF
(all dimensions are in inches)

6x6B (3.0% Enrichment†, ~~Boral~~*fixed neutron absorber* ^{10}B minimum loading of 0.0067 g/cm^2) 35 or 36 fuel rods†† (up to 9 MOX rods), 1 or 0 water rods††, pitch††=0.694 to 0.710, Zr clad												
Fuel Assembly Designation	maximum k_{eff}	calculated k_{eff}	standard deviation	pitch	fuel rods	cladding OD	cladding ID	cladding thickness	pellet OD	fuel length	water rod thickness	channel thickness
6x6B01	0.7604	0.7563	0.0007	0.694	36	0.5645	0.4945	0.0350	0.4820	110	n/a	0.060
6x6B02	0.7618	0.7577	0.0007	0.694	36	0.5625	0.4925	0.0350	0.4820	110	n/a	0.060
6x6B03	0.7619	0.7578	0.0007	0.696	36	0.5625	0.4925	0.0350	0.4820	110	n/a	0.060
6x6B04	0.7686	0.7644	0.0008	0.696	35	0.5625	0.4925	0.0350	0.4820	110	0.0	0.060
6x6B05	0.7824	0.7785	0.0006	0.710	35	0.5625	0.4925	0.0350	0.4820	110	0.0	0.060
Dimensions Listed *for Authorized Contents*				0.710 (max.)	35 or 36	0.5625 (min.)	0.4945 (max.)		0.4820 (max.)	120 (max.)	0.0	0.060 (max.)
bounding dimensions (B6x6B01)	0.7822†††	0.7783	0.0006	0.710	35	0.5625	0.4945	0.0340	0.4820	120	0.0	0.060

Note:

1. These assemblies contain up to 9 MOX pins. The composition of the MOX fuel pins is given in Table 6.3.4.

† The ^{235}U enrichment of the MOX and UO_2 pins is assumed to be 0.711% and 3.0%, respectively.

†† This assembly class was analyzed and qualified for a small variation in the pitch and a variation in the number of fuel and water rods.

††† The k_{eff} value listed for the 6x6B05 case is slightly higher than that for the case with the bounding dimensions. However, the difference (0.0002) is well within the statistical uncertainties, and thus, the two values are statistically equivalent (within 1σ). Therefore, the 0.7824 value is listed in Tables 6.1.7 and 6.1.8 as the maximum.

Table 6.2.43
MAXIMUM K_{EFF} VALUES FOR THE 6X6C ASSEMBLY CLASS IN THE MPC-68F and MPC-68FF

(all dimensions are in inches)

6x6C (3.0% Enrichment†, ~~Boral~~*fixed neutron absorber* ^{10}B minimum loading of 0.0067 g/cm^2) 36 fuel rods, 0 water rods, pitch=0.740, Zr clad										
Fuel Assembly Designation	maximum k_{eff}	calculated k_{eff}	standard deviation	cladding OD	cladding ID	cladding thickness	pellet OD	fuel length	water rod thickness	channel thickness
6x6C01	0.8021	0.7980	0.0007	0.5630	0.4990	0.0320	0.4880	77.5	n/a	0.060
Dimensions Listed *for Authorized Contents*				0.5630 (min.)	0.4990 (max.)		0.4880 (max.)	77.5 (max.)	n/a	0.060 (max.)

† Although the calculations were performed for 3.0%, the enrichment is limited in the ~~Certificate of Compliance~~ *specification for the authorized contents* to 2.7%.

Table 6.2.44
MAXIMUM K_{EFF} VALUES FOR THE 7X7A ASSEMBLY CLASS IN THE MPC-68F and MPC-68FF

(all dimensions are in inches)

7x7A (3.0% Enrichment†, ~~Boral~~*fixed neutron absorber* ^{10}B minimum loading of 0.0067 g/cm^2) 49 fuel rods, 0 water rods, pitch=0.631, Zr clad										
Fuel Assembly Designation	maximum k_{eff}	calculated k_{eff}	standard deviation	cladding OD	cladding ID	cladding thickness	pellet OD	fuel length	water rod thickness	channel thickness
7x7A01	0.7974	0.7932	0.0008	0.4860	0.4204	0.0328	0.4110	80	n/a	0.060
Dimensions Listed *for Authorized Contents*				0.4860 (min.)	0.4204 (max.)		0.4110 (max.)	80 (max.)	n/a	0.060 (max.)

† Although the calculations were performed for 3.0%, the enrichment is limited in the ~~Certificate of Compliance~~ *specification for the authorized contents* to 2.7%.

Table 6.2.45
MAXIMUM K_{EFF} VALUES FOR THE 8X8A ASSEMBLY CLASS IN THE MPC-68F and MPC-68FF

(all dimensions are in inches)

8x8A (3.0% Enrichment†, ~~Boral~~*fixed neutron absorber* ^{10}B minimum loading of 0.0067 g/cm^2) 63 or 64 fuel rods††, 0 water rods, pitch=0.523, Zr clad											
Fuel Assembly Designation	maximum k_{eff}	calculated k_{eff}	standard deviation	fuel rods	cladding OD	cladding ID	cladding thickness	pellet OD	fuel length	water rod thickness	channel thickness
8x8A01	0.7685	0.7644	0.0007	64	0.4120	0.3620	0.0250	0.3580	110	n/a	0.100
8x8A02	0.7697	0.7656	0.0007	63	0.4120	0.3620	0.0250	0.3580	120	n/a	0.100
Dimensions Listed *for Authorized Contents*				63	0.4120 (min.)	0.3620 (max.)		0.3580 (max.)	120 (max.)	n/a	0.100 (max.)
bounding dimensions (8x8A02)	0.7697	0.7656	0.0007	63	0.4120	0.3620	0.0250	0.3580	120	n/a	0.100

† Although the calculations were performed for 3.0%, the enrichment is limited in the ~~Certificate of Compliance~~ *specification for the authorized contents* to 2.7%.

†† This assembly class was analyzed and qualified for a variation in the number of fuel rods.

Table 6.2.46

SPECIFICATION OF THE THORIA ROD CANISTER AND THE THORIA RODS

Canister ID	4.81"
Canister Wall Thickness	0.11"
Separator Assembly Plates Thickness	0.11"
Cladding OD	0.412"
Cladding ID	0.362"
Pellet OD	0.358"
Active Length	110.5"
Fuel Composition	1.8% UO_2 and 98.2% ThO_2
Initial Enrichment	93.5 wt% ^{235}U for 1.8% of the fuel
Maximum k_{eff}	0.1813
Calculated k_{eff}	0.1779
Standard Deviation	0.0004

6.3 MODEL SPECIFICATION

6.3.1 Description of Calculational Model

Figures 6.3.1, 6.3.1.a, 6.3.2 and 6.3.3 show representative horizontal cross sections of the four types of cells used in the calculations, and Figures 6.3.4 through 6.3.6 illustrate the basket configurations used. Four different MPC fuel basket designs were evaluated as follows:

- a 24 PWR assembly basket
- an optimized 24 PWR assembly basket (24E / 24EF)
- a 32 PWR assembly basket
- a 68 BWR assembly basket.

For all four basket designs, the same techniques and the same level of detail are used in the calculational models.

Full three-dimensional calculations were used, assuming the axial configuration shown in Figure 6.3.7. Although the ~~Boral~~*fixed* neutron absorber panels are 156 inches in length, which is much longer than the active fuel length (maximum of 150 inches), they are assumed equal to *or less than* the active fuel length in the calculations. As shown on the Drawings in Section 1.5, 16 of the 24 periphery ~~Boral~~*fixed neutron absorber* panels on the MPC-24 and MPC-24E/EF have reduced width (i.e., 6.25 inches wide as opposed to 7.5 inches). However, the calculational models for these baskets conservatively assume all of the periphery ~~Boral~~*fixed neutron absorber* panels are 6.25 inches in width. *Note that Figures 6.3.1 through 6.3.3 show Boral as the fixed neutron absorber. The effect of using Metamic as fixed neutron absorber is discussed in Subsection 6.4.11.*

The off-normal and accident conditions defined in Chapter 2 and considered in Chapter 11 have no adverse effect on the design conditions important to criticality safety (see Subsection 6.4.2.5), and thus from a criticality standpoint, the normal, off-normal, and accident conditions are identical and do not require individual models.

The calculational model explicitly defines the fuel rods and cladding, the guide tubes (or water rods for BWR assemblies), the water-gaps and ~~Boral~~*fixed neutron* absorber panels on the stainless steel walls of the storage cells. Under the conditions of storage, when the MPC is dry, the resultant reactivity with the design basis fuel is very low ($k_{eff} < 0.52$). For the flooded condition (loading and unloading), pure, unborated water was assumed to be present in the fuel

rod pellet-to-clad gaps. Appendix 6.D provides sample input files for two of the MPC basket designs (MPC-68 and MPC-24) in the HI-STORM 100 System.

The water thickness above and below the fuel is intentionally maintained less than or equal to the actual water thickness. This assures that any positive reactivity effect of the steel in the MPC is conservatively included. Furthermore, the water above and below the fuel is modeled as unborated water, even when borated water is present in the fuel region.

As indicated in Figures 6.3.1 through 6.3.3 and in Tables 6.3.1 and 6.3.2, calculations were made with dimensions assumed to be at their most conservative value with respect to criticality. CASMO-3 and MCNP4a were used to determine the direction of the manufacturing tolerances, which produced the most adverse effect on criticality. After the directional effect (positive effect with an increase in reactivity; or negative effect with a decrease in reactivity) of the manufacturing tolerances was determined, the criticality analyses were performed using the worst case tolerances in the direction which would increase reactivity.

CASMO-3 was used for one of each of the two principal basket designs, i.e. for the flux trap design MPC-24 and for the non-fluxtrap design MPC-68. The effects are shown in Table 6.3.1 which also identifies the approximate magnitude of the tolerances on reactivity. Generally, the conclusions in Table 6.3.1 are directly applicable to the MPC-24E/EF and the MPC-32. Exceptions are the conclusions for the water temperature and void percentage, which are not directly applicable to the MPC-32 due to the presence of high soluble boron concentrations in this canister. This condition is addressed in Section 6.4.2.1 where the optimum moderation is determined for the MPC-32.

Additionally, MCNP4a calculations are performed to evaluate the tolerances of the various basket dimensions of the MPC-68, MPC-24 and MPC-32 in further detail. The various basket dimensions are inter-dependent, and therefore cannot be individually varied (i.e., reduction in one parameter requires a corresponding reduction or increase in another parameter). Thus, it is not possible to determine the reactivity effect of each individual dimensional tolerance separately. However, it is possible to determine the reactivity effect of the dimensional tolerances by evaluating the various possible dimensional combinations. To this end, an evaluation of the various possible dimensional combinations was performed using MCNP4a. Calculated k_{eff} results (which do not include the bias, uncertainties, or calculational statistics), along with the actual dimensions, for a number of dimensional combinations are shown in Table 6.3.2 for the reference PWR and BWR assemblies. Each of the basket dimensions are evaluated for their minimum, nominal and maximum values from the Drawings of section 1.5. For PWR MPC designs, the reactivity effect of tolerances with soluble boron present in the water is additionally determined. Due to the close similarity between the MPC-24 and MPC-24E, the basket dimensions are only evaluated for the MPC-24, and the same dimensional assumptions are applied to both MPC designs.

Based on the MCNP4a and CASMO-3 calculations, the conservative dimensional assumptions listed in Table 6.3.3 were determined. Because the reactivity effect (positive or negative) of the manufacturing tolerances are not assembly dependent, these dimensional assumptions were employed for the criticality analyses.

As demonstrated in this section, design parameters important to criticality safety are: fuel enrichment, the inherent geometry of the fuel basket structure, the fixed neutron absorbing panels ~~(Boral)~~ and the soluble boron concentration in the water during loading/unloading operations. As shown in Chapter 11, none of these parameters are affected during any of the design basis off-normal or accident conditions involving handling, packaging, transfer or storage.

6.3.2 Cask Regional Densities

Composition of the various components of the principal designs of the HI-STORM 100 System are listed in Table 6.3.4.

The HI-STORM 100 System is designed such that the fixed neutron absorber ~~(Boral)~~ will remain effective for a storage period greater than 20 years, and there are no credible means to lose it. A detailed physical description, historical applications, unique characteristics, service experience, and manufacturing quality assurance of ~~Boral~~*fixed neutron absorber* are provided in Section 1.2.1.3.1.

The continued efficacy of the ~~Boral~~*fixed neutron absorber* is assured by acceptance testing, documented in Section 9.1.5.3, to validate the ^{10}B (poison) concentration in the ~~Boral~~*fixed neutron absorber*. To demonstrate that the neutron flux from the irradiated fuel results in a negligible depletion of the poison material over the storage period, an MCNP4a calculation of the number of neutrons absorbed in the ^{10}B was performed. The calculation conservatively assumed a constant neutron source for 50 years equal to the initial source for the design basis fuel, as determined in Section 5.2, and shows that the fraction of ^{10}B atoms destroyed is only 2.6E-09 in 50 years. Thus, the reduction in ^{10}B concentration in the ~~Boral~~*fixed neutron absorber* by neutron absorption is negligible. In addition, ~~analysis in Appendix 3.M.1 of the HI-STAR 100 FSAR~~ *the results presented in Subsection 3.4.4.3.1.8* demonstrates that the sheathing, which affixes the ~~Boral~~*fixed neutron absorber* panel, remains in place during all credible accident conditions, and thus, the ~~Boral~~*fixed neutron absorber* panel remains permanently fixed. Therefore, in accordance with 10CFR72.124(b), there is no need to provide a surveillance or monitoring program to verify the continued efficacy of the neutron absorber.

6.3.3 Eccentric Positioning of Assemblies in Fuel Storage Cells

Up to and including Revision 1 of this FSAR, all criticality calculations were performed with fuel assemblies centered in the fuel storage locations since the effect of credible eccentric fuel positioning was judged to be not significant. Starting in Revision 2 of this FSAR, the potential reactivity effect of eccentric positioning of assemblies in the fuel storage locations is accounted for in a conservatively bounding fashion, as described further in this subsection, for all new or changed conditions. The calculations in this subsection serve to determine for which of these conditions the eccentric positioning of assemblies in the fuel storage locations results in a higher maximum k_{eff} value than the centered positioning. For the cases where the eccentric positioning results in a higher maximum k_{eff} value, the eccentric positioning is used for all corresponding cases reported in the summary tables in Section 6.1 and the results tables in Section 6.4. All other calculations throughout this chapter, such as studies to determine bounding fuel dimensions, bounding basket dimensions, or bounding moderation conditions, are performed with assemblies centered in the fuel storage locations.

To conservatively account for eccentric fuel positioning in the fuel storage cells, three different configurations are analyzed, and the results are compared to determine the bounding configuration:

- *Cell Center Configuration: All assemblies centered in their fuel storage cell; same configuration that is used in Section 6.2 and Section 6.3.1;*
- *Basket Center Configuration: All assemblies in the basket are moved as close to the center of the basket as permitted by the basket geometry; and*
- *Basket Periphery Configuration: All assemblies in the basket are moved furthest away from the basket center, and as close to the periphery of the basket as possible.*

It should be noted that the two eccentric configurations are hypothetical, since there is no known physical effect that could move all assemblies within a basket consistently to the center or periphery. Instead, the most likely configuration would be that all assemblies are moved in the same direction when the cask is in a horizontal position, and that assemblies are positioned randomly when the cask is in a vertical position. Further, it is not credible to assume that any such configuration could exist by chance. Even if the probability for a single assembly placed in the corner towards the basket center would be 1/5 (i.e. assuming only the center and four corner positions in each cell, all with equal probability), then the probability that all assemblies would be located towards the center would be $(1/5)^{24}$ or approximately 10^{-17} for the MPC-24, $(1/5)^{32}$ or approximately 10^{-23} for the MPC-32, and $(1/5)^{68}$ or approximately 10^{-48} for the MPC-68. However, since the configurations listed above bound all credible configurations, they are conservatively used in the analyses.

In Table 6.3.5, results are presented for all conditions that were introduced in Revision 2 of this FSAR, namely results for the MPC-24E/EF with intact and damaged fuel at 5 wt% ^{235}U, for the MPC-32 with soluble boron levels lower than 2600 ppm for 5 wt% ^{235}U and lower than 1900 ppm for 4.1 wt% ^{235}U, and for the MPC-32 with intact and damaged fuel. The table shows the maximum k_{eff} value for centered and the two eccentric configurations for each condition, and the difference in k_{eff} between the centered and eccentric positioning. The results and conclusions are summarized as follows:

- *In all cases, moving the assemblies to the periphery of the basket results in a reduction in reactivity, compared to the cell centered position.*
- *For the MPC-24E/EF, moving the assemblies and DFCs towards the center of the basket also results in a minor reduction. The cell centered configuration is therefore bounding for this condition and is used in the design basis calculations reported in Section 6.1 and Section 6.4.*
- *For the MPC-32 cases listed in Table 6.3.5, the maximum reactivity is shown for the basket center configuration. However, for some of the cases with intact and damaged fuel in the MPC-32, the cell centered configuration results in a higher maximum reactivity. Therefore, both the cell centered and basket centered configuration are analyzed for the MPC-32 design basis calculation, and the higher results are listed in the tables in Section 6.1. and 6.4. This applies to the cases with intact and damaged fuel, and to cases with intact fuel only and soluble boron levels lower than 2600 ppm for 5 wt% ^{235}U and lower than 1900 ppm for 4.1 wt% ^{235}U.*

Table 6.3.1

CASMO-3 CALCULATIONS FOR EFFECT OF TOLERANCES AND TEMPERATURE

Change in Nominal Parameter†	Δk for Maximum Tolerance		Action/Modeling Assumption
	MPC-24‡	MPC-68	
Reduce ~~Boral~~*Fixed Neutron Absorber* Width to Minimum	N/A††† min.= nom.= 7.5" and 6.25"	N/A††† min. = nom. = 4.75"	Assume minimum ~~Boral~~*fixed neutron absorber* width
Increase UO_2 Density to Maximum	+0.0017 max. = 10.522 g/cc nom. = 10.412 g/cc	+0.0014 max. = 10.522 g/cc nom. = 10.412 g/cc	Assume maximum UO_2 density
Reduce Box Inside Dimension (I.D.) to Minimum	-0.0005 min.= 8.86" nom. = 8.92"	See Table 6.3.2	Assume maximum box I.D. for the MPC-24
Increase Box Inside Dimension (I.D.) to Maximum	+0.0007 max. = 8.98" nom. = 8.92"	-0.0030 max. = 6.113" nom. = 6.053"	Assume minimum box I.D. for the MPC-68
Decrease Water Gap to Minimum	+0.0069 min. = 1.09" nom. = 1.15"	N/A	Assume minimum water gap in the MPC-24

† Reduction (or increase) in a parameter indicates that the parameter is changed to its minimum (or maximum) value.

‡ Calculations for the MPC-24 were performed with CASMO-4 [6.3.1-6.3.3].

††† The ~~Boral~~*fixed neutron absorber* width for the MPC-68 is 4.75" +0.125", -0" , the ~~Boral~~*fixed neutron absorber* widths for the MPC-24 are 7.5" +0.125", -0" and 6.25" +0.125" –0" (i.e., the nominal and minimum values are the same).

Table 6.3.1 (continued)

CASMO-3 CALCULATIONS FOR EFFECT OF TOLERANCES AND TEMPERATURE

Change in Nominal Parameter	Δk Maximum Tolerance		Action/Modeling Assumption
	MPC-24‡	MPC-68	
Increase in Temperature			Assume 20°C
20°C	Ref.	Ref.	
40°C	-0.0030	-0.0039	
70°C	-0.0089	-0.0136	
100°C	-0.0162	-0.0193	
10% Void in Moderator			Assume no void
20°C with no void	Ref.	Ref.	
20°C	-0.0251	-0.0241	
100°C	-0.0412	-0.0432	
Removal of Flow Channel (BWR)	N/A	-0.0073	Assume flow channel present for MPC-68

‡ Calculations for the MPC-24 were performed with CASMO-4 [6.3.1-6.3.3].

Table 6.3.2

MCNP4a EVALUATION OF BASKET MANUFACTURING TOLERANCES†

Pitch		Box I.D.		Box Wall Thickness		MCNP4a Calculated k_{eff}
MPC-24†† (17x17A01 @ 4.0% Enrichment)						
nominal	(10.906")	maximum	(8.98")	nominal	(5/16")	0.9325±0.0008†††
minimum	(10.846")	nominal	(8.92")	nominal	(5/16")	0.9300±0.0008
nominal	(10.906")	nom. - 0.04"	(8.88")	nom. + 0.05"	(0.3625")	0.9305±0.0007
MPC-68 (8x8C04 @ 4.2% Enrichment)						
minimum	(6.43")	minimum	(5.993")	nominal	(1/4")	0.9307±0.0007
nominal	(6.49")	nominal	(6.053")	nominal	(1/4")	0.9274±0.0007
maximum	(6.55")	maximum	(6.113")	nominal	(1/4")	0.9272±0.0008
nom. + 0.05"	(6.54")	nominal	(6.053")	nom. + 0.05"	(0.30")	0.9267±0.0007

Notes:

1. Values in parentheses are the actual value used.

† Tolerance for pitch and box I.D. are ± 0.06".
Tolerance for box wall thickness is +0.05", -0.00".

†† All calculations for the MPC-24 assume minimum water gap thickness (1.09").

††† Numbers are 1σ statistical uncertainties.

Table 6.3.2 (cont.)

MCNP4a EVALUATION OF BASKET MANUFACTURING TOLERANCES†

Pitch	Box I.D.	Box Wall Thickness	MCNP4a Calculated k_{eff}
MPC-24 (17x17A @ 5.0% Enrichment) 400ppm soluble boron			
nominal (10.906")	maximum (8.98")	nominal (5/16")	0.9236±0.0007††
maximum (10.966")	maximum (8.98")	nominal (5/16")	0.9176±0.0008
minimum (10.846")	nominal (8.92")	nominal (5/16")	0.9227±0.0010
minimum (10.846")	minimum (8.86")	nominal (5/16")	0.9159±0.0008
nominal (10.906")	nominal-0.04" (8.88")	nom.+0.05" (0.3625")	0.9232±0.0009
nominal (10.906")	nominal (8.92")	nominal (5/16")	0.9158±0.0007
MPC-32 (17x17A @ 5.0% Enrichment) 2600 ppm soluble boron			
minimum (9.158")	minimum (8.69")	nominal (9/32")	0.9085±0.0007
nominal (9.218")	nominal (8.75")	nominal (9/32")	0.9028±0.0007
maximum (9.278")	maximum (8.81")	nominal (9/32")	0.8996±0.0008
nominal+0.05" (9.268")	nominal (8.75")	nominal+0.05" (0.331")	0.9023±0.0008
minimum+0.05"(9.208")	minimum (8.69")	nominal+0.05" (0.331")	0.9065±0.0007
maximum (9.278")	Maximum-0.05" (8.76")	nominal+0.05" (0.331")	0.9030±0.0008

Notes:

1. Values in parentheses are the actual value used.

† Tolerance for pitch and box I.D. are ± 0.06".
Tolerance for box wall thickness is +0.05", -0.00".

†† Numbers are 1σ statistical uncertainties.

Table 6.3.3

BASKET DIMENSIONAL ASSUMPTIONS

Basket Type	Pitch	Box I.D.	Box Wall Thickness	Water-Gap Flux Trap
MPC-24	nominal (10.906")	maximum (8.98")	nominal (5/16")	minimum (1.09")
MPC-24E	nominal (10.847")	maximum (8.81", 9.11" for DFC Positions)	nominal (5/16")	minimum (1.076", 0.776" for DFC Positions)
MPC-32	Minimum (9.158")	Minimum (8.69")	Nominal (9/32")	N/A
MPC-68	minimum (6.43")	Minimum (5.993")	nominal (1/4")	N/A

Table 6.3.4

COMPOSITION OF THE MAJOR COMPONENTS OF THE HI-STORM 100 SYSTEM

MPC-24, MPC-24E and MPC-32		
UO_2 5.0% ENRICHMENT, DENSITY (g/cc) = 10.522		
Nuclide	**Atom-Density**	**Wgt. Fraction**
8016	4.696E-02	1.185E-01
92235	1.188E-03	4.408E-02
92238	2.229E-02	8.374E-01
UO_2 4.0% ENRICHMENT, DENSITY (g/cc) = 10.522		
Nuclide	**Atom-Density**	**Wgt. Fraction**
8016	4.693E-02	1.185E-01
92235	9.505E-04	3.526E-02
92238	2.252E-02	8.462E-01
BORAL (0.02 g ^{10}B/cm sq), DENSITY (g/cc) = 2.660 (MPC-24)		
Nuclide	**Atom-Density**	**Wgt. Fraction**
5010	8.707E-03	5.443E-02
5011	3.512E-02	2.414E-01
6012	1.095E-02	8.210E-02
13027	3.694E-02	6.222E-01
BORAL (0.0279 g ^{10}B/cm sq), DENSITY (g/cc) = 2.660 (MPC-24E and MPC-32)		
Nuclide	**Atom-Density**	**Wgt. Fraction**
5010	8.071E-03	5.089E-02
5011	3.255E-02	2.257E-01
6012	1.015E-02	7.675E-02
13027	3.805E-02	6.467E-01

Table 6.3.4 (continued)

COMPOSITION OF THE MAJOR COMPONENTS OF THE HI-STORM 100 SYSTEM

METAMIC (0.02 g ^{10}B/cm sq), DENSITY (g/cc) = 2.648 (MPC-24)		
Nuclide	***Atom-Density***	***Wgt. Fraction***
5010	*6.314E-03*	*3.965E-02*
5011	*2.542E-02*	*1.755E-01*
6012	*7.932E-02*	*5.975E-02*
13027	*4.286E-02*	*7.251E-01*
METAMIC (0.0279 g ^{10}B/cm sq), DENSITY (g/cc) = 2.646 (MPC-24E and MPC-32)		
Nuclide	***Atom-Density***	***Wgt. Fraction***
5010	*6.541E-03*	*4.110E-02*
5011	*2.633E-02*	*1.819E-01*
6012	*8.217E-03*	*6.193E-02*
13027	*4.223E-02*	*7.151E-01*

Table 6.3.4 (continued)

COMPOSITION OF THE MAJOR COMPONENTS OF THE HI-STORM 100 SYSTEM

BORATED WATER, 300 PPM, DENSITY (g/cc)=1.00		
Nuclide	**Atom-Density**	**Wt. Fraction**
5010	3.248E-06	5.400E-05
5011	1.346E-05	2.460E-04
1001	6.684E-02	1.1186E-01
8016	3.342E-02	8.8784E-01
BORATED WATER, 400PPM, DENSITY (g/cc)=1.00		
Nuclide	**Atom-Density**	**Wgt. Fraction**
5010	4.330E-06	7.200E-05
5011	1.794E-05	3.280E-04
1001	6.683E-02	1.1185E-01
8016	3.341E-02	8.8775E-01
BORATED WATER, 1900PPM, DENSITY (g/cc)=1.00		
Nuclide	Atom-Density	**Wgt. Fraction**
5010	2.057E-05	3.420E-04
5011	8.522E-05	1.558E-03
1001	6.673E-02	1.1169E-01
8016	3.336E-02	8.8641E-01
BORATED WATER, 2600PPM, DENSITY (g/cc)=0.93		
Nuclide	Atom-Density	**Wgt. Fraction**
5010	2.618e-05	4.680E-04
5011	1.085e-04	2.132E-03
1001	6.201e-02	1.1161E-01
8016	3.101e-02	8.8579E-01

Table 6.3.4 (continued)

COMPOSITION OF THE MAJOR COMPONENTS OF THE HI-STORM 100 SYSTEM

MPC-68		
UO_2 4.2% ENRICHMENT, DENSITY (g/cc) = 10.522		
Nuclide	**Atom-Density**	**Wgt. Fraction**
8016	4.697E-02	1.185E-01
92235	9.983E-04	3.702E-02
92238	2.248E-02	8.445E-01
UO_2 3.0% ENRICHMENT, DENSITY (g/cc) = 10.522		
Nuclide	**Atom-Density**	**Wgt. Fraction**
8016	4.695E-02	1.185E-01
92235	7.127E-04	2.644E-02
92238	2.276E-02	8.550E-01
MOX FUEL†, DENSITY (g/cc) = 10.522		
Nuclide	**Atom-Density**	**Wgt. Fraction**
8016	4.714E-02	1.190E-01
92235	1.719E-04	6.380E-03
92238	2.285E-02	8.584E-01
94239	3.876E-04	1.461E-02
94240	9.177E-06	3.400E-04
94241	3.247E-05	1.240E-03
94242	2.118E-06	7.000E-05

† The Pu-238, which is an absorber, was conservatively neglected in the MOX description for analysis purposes.

Table 6.3.4 (continued)

COMPOSITION OF THE MAJOR COMPONENTS OF THE HI-STORM 100 SYSTEM

BORAL (0.0279 g ^{10}B/cm sq), DENSITY (g/cc) = 2.660		
Nuclide	**Atom-Density**	**Wgt. Fraction**
5010	8.071E-03	5.089E-02
5011	3.255E-02	2.257E-01
6012	1.015E-02	7.675E-02
13027	3.805E-02	6.467E-01
METAMIC (0.0279 g ^{10}B/cm sq), DENSITY (g/cc) = 2.646		
Nuclide	***Atom-Density***	***Wgt. Fraction***
5010	*6.541E-03*	*4.110E-02*
5011	*2.633E-02*	*1.819E-01*
6012	*8.217E-03*	*6.193E-02*
13027	*4.223E-02*	*7.151E-01*
FUEL IN THORIA RODS, DENSITY (g/cc) = 10.522		
Nuclide	**Atom-Density**	**Wgt. Fraction**
~~50~~*08016*	4.798E-02	1.212E-01
~~501~~*92235*	4.001E-04	1.484E-02
~~601~~*92238*	2.742E-05	1.030E-03
~~1302~~*90232*	2.357E-02	8.630E-01
COMMON MATERIALS		
ZR CLAD, DENSITY (g/cc) = 6.550		
Nuclide	**Atom-Density**	**Wgt. Fraction**
40000	4.323E-02	1.000E+00
MODERATOR (H_2O), DENSITY (g/cc) = 1.000		
Nuclide	**Atom-Density**	**Wgt. Fraction**
1001	6.688E-02	1.119E-01
8016	3.344E-02	8.881E-01

Table 6.3.4 (continued)

COMPOSITION OF THE MAJOR COMPONENTS OF THE HI-STORM 100 SYSTEM

STAINLESS STEEL, DENSITY (g/cc) = 7.840		
Nuclide	**Atom-Density**	**Wgt. Fraction**
24000	1.761E-02	1.894E-01
25055	1.761E-03	2.001E-02
26000	5.977E-02	6.905E-01
28000	8.239E-03	1.000E-01
ALUMINUM, DENSITY (g/cc) = 2.700		
Nuclide	**Atom-Density**	**Wgt. Fraction**
13027	6.026E-02	1.000E+00

Table 6.3.4 (continued)

COMPOSITION OF THE MAJOR COMPONENTS OF THE HI-STORM 100 SYSTEM

CONCRETE, DENSITY (g/cc) = 2.35		
Nuclide	**Atom-Density**	**Wgt. Fraction**
1001	8.806E-03	6.000E-03
8016	4.623E-02	5.000E-01
11000	1.094E-03	1.700E-02
13027	2.629E-04	4.800E-03
14000	1.659E-02	3.150E-01
19000	7.184E-04	1.900E-02
20000	3.063E-03	8.300E-02
26000	3.176E-04	1.200E-02
LEAD, DENSITY (g/cc) = 11.34		
Nuclide	**Atom-Density**	**Wgt. Fraction**
82000	3.296E-02	1.0
HOLTITE-A, DENSITY (g/cc) = 1.61		
1001	5.695E-02	5.920E-02
5010	1.365E-04	1.410E-03
5011	5.654E-04	6.420E-03
6012	2.233E-02	2.766E-01
7014	1.370E-03	1.980E-02
8016	2.568E-02	4.237E-01
13027	7.648E-03	2.129E-01

Table 6.3.5

REACTIVITY EFFECTS OF ECCENTRIC POSITIONING OF CONTENT (FUEL ASSEMBLIES AND DFCs) IN BASKET CELLS

CASE	Contents centered (Reference)	Content moved towards center of basket		Content moved towards basket periphery	
	Maximum k_{eff}	Maximum k_{eff}	k_{eff} Difference to Reference	Maximum k_{eff}	k_{eff} Difference to Reference
MPC-24E/EF, Intact Fuel and Damaged Fuel/Fuel Debris, 5% Enrichment, 600ppm Soluble Boron	0.9185	0.9178	-0.0007	0.9132	-0.0053
MPC-32/32F, Intact Fuel, Assembly Class 16x16A, 4.1% Enrichment, 1300ppm Soluble Boron	0.9429	0.9468	0.0039	0.9068	-0.0361
MPC-32/32F, Intact Fuel, Assembly Class 15x15B, 5.0% Enrichment, 2400ppm Soluble Boron	0.9473	0.9493	0.0020	0.9306	-0.0167
MPC-32/32F, Intact Fuel and Damaged Fuel/Fuel Debris, Assembly Class 15x15F (Intact), 5% Enrichment, 2900ppm Soluble Boron	0.9378	0.9397	0.0019	0.9277	-0.0101

6.4 CRITICALITY CALCULATIONS

6.4.1 Calculational or Experimental Method

6.4.1.1 Basic Criticality Safety Calculations

The principal method for the criticality analysis is the general three-dimensional continuous energy Monte Carlo N-Particle code MCNP4a [6.1.4] developed at the Los Alamos National Laboratory. MCNP4a was selected because it has been extensively used and verified and has all of the necessary features for this analysis. MCNP4a calculations used continuous energy cross-section data based on ENDF/B-V, as distributed with the code [6.1.4]. Independent verification calculations were performed with NITAWL-KENO5a [6.1.5], which is a three-dimensional multigroup Monte Carlo code developed at the Oak Ridge National Laboratory. The KENO5a calculations used the 238-group cross-section library, which is based on ENDF/B-V data and is distributed as part of the SCALE-4.3 package [6.4.1], in association with the NITAWL-II program [6.1.6], which adjusts the uranium-238 cross sections to compensate for resonance self-shielding effects. The Dancoff factors required by NITAWL-II were calculated with the CELLDAN code [6.1.13], which includes the SUPERDAN code [6.1.7] as a subroutine.

The convergence of a Monte Carlo criticality problem is sensitive to the following parameters: (1) number of histories per cycle, (2) the number of cycles skipped before averaging, (3) the total number of cycles and (4) the initial source distribution. The MCNP4a criticality output contains a great deal of useful information that may be used to determine the acceptability of the problem convergence. This information was used in parametric studies to develop appropriate values for the aforementioned criticality parameters to be used in the criticality calculations for this submittal. Based on these studies, a minimum of 5,000 histories were simulated per cycle, a minimum of 20 cycles were skipped before averaging, a minimum of 100 cycles were accumulated, and the initial source was specified as uniform over the fueled regions (assemblies). Further, the output was examined to ensure that each calculation achieved acceptable convergence. These parameters represent an acceptable compromise between calculational precision and computational time. Appendix 6.D provides sample input files for the MPC-24 and MPC-68 basket in the HI-STORM 100 System.

CASMO-3 [6.1.9] was used for determining the small incremental reactivity effects of manufacturing tolerances. Although CASMO-3 has been extensively benchmarked, these calculations are used only to establish direction of reactivity uncertainties due to manufacturing tolerances (and their magnitude). This allows the MCNP4a calculational model to use the worst combination of manufacturing tolerances. Table 6.3.1 shows results of the CASMO-3 calculations.

6.4.2 Fuel Loading or Other Contents Loading Optimization

The basket designs are intended to safely accommodate fuel with enrichments indicated in Tables 6.1.1 through 6.1.8 . These calculations were based on the assumption that the HI-STORM 100 System (HI-TRAC transfer cask) was fully flooded with clean unborated water or water containing specific minimum soluble boron concentrations. In all cases, the calculations include bias and calculational uncertainties, as well as the reactivity effects of manufacturing tolerances, determined by assuming the worst case geometry.

~~Nominally, the fuel assemblies would be centrally positioned in each MPC basket cell. However, in accordance with NUREG-1536, the consequence of eccentric positioning was also evaluated and found to be negligible. To simulate eccentric positioning (and possible closer approach to the MPC steel shield), calculations were performed analytically decreasing the inner radius until it was 1 cm away† from the nearest fuel. Results showed a minor increase in reactivity of 0.0026 Δk maximum (MPC-68) which implies that the effect of eccentric location of fuel is negligible at the actual reflector spacing.~~

6.4.2.1 Internal and External Moderation

As required by NUREG-1536, calculations in this section demonstrate that the HI-STORM 100 System remains subcritical for all credible conditions of moderation.

6.4.2.1.1 Unborated Water

With a neutron absorber present (i.e., the ~~Boral~~*fixed neutron absorber* sheets or the steel walls of the storage compartments), the phenomenon of a peak in reactivity at a hypothetical low moderator density (sometimes called "optimum" moderation) does not occur to any significant extent. In a definitive study, Cano, et al. [6.4.2] has demonstrated that the phenomenon of a peak in reactivity at low moderator densities does not occur in the presence of strong neutron absorbing material or in the absence of large water spaces between fuel assemblies in storage. Nevertheless, calculations for a single reflected cask were made to confirm that the phenomenon does not occur with low density water inside or outside the casks.

Calculations for the MPC designs with internal and external moderators of various densities are shown in Table 6.4.1. For comparison purposes, a calculation for a single unreflected cask (Case

† PNL critical experiments have shown a small positive reactivity effect of thick steel reflectors, with the maximum effect at 1 cm distance from the fuel. In the cask designs, the fuel is mechanically prohibited from being positioned at a 1 cm spacing from the overpack steel.

1) is also included in Table 6.4.1. At 100% external moderator density, Case 2 corresponds to a single fully-flooded cask, fully reflected by water. Figure 6.4.10 plots calculated k_{eff} values ($\pm 2\sigma$) as a function of internal moderator density for both MPC designs with 100% external moderator density (i.e., full water reflection). Results listed in Table 6.4.1 support the following conclusions:

- For each type of MPC, the calculated k_{eff} for a fully-flooded cask is independent of the external moderator (the small variations in the listed values are due to statistical uncertainties which are inherent to the calculational method (Monte Carlo)), and

- For each type of MPC, reducing the internal moderation results in a monotonic reduction in reactivity, with no evidence of any optimum moderation. Thus, the fully flooded condition corresponds to the highest reactivity, and the phenomenon of optimum low-density moderation does not occur and is not applicable to the HI-STORM 100 System.

For each of the MPC designs, the maximum k_{eff} values are shown to be less than or statistically equal to that of a single internally flooded unreflected cask and are below the regulatory limit of 0.95.

6.4.2.1.2 Borated Water

With the presence of a soluble neutron absorber in the water, the discussion in the previous section is not always applicable. Calculations were made to determine the optimum moderator density for the MPC designs that require a minimum soluble boron concentration.

Calculations for the MPC designs with various internal moderator densities are shown in Table 6.4.6. As shown in the previous section, the external moderator density has a negligible effect on the reactivity, and is therefore not varied. Water containing soluble boron has a slightly higher density than pure water. Therefore, water densities up to 1.005 g/cm^3 were analyzed for the higher soluble boron concentrations. Additionally, for the higher soluble boron concentrations, analysis have been performed with empty (voided) guide tubes. This variation is discussed in detail in Section 6.4.8. Results listed in the Table 6.4.6 support the following conclusions:

- For all cases with a soluble boron concentration of up to 1900ppm, and for a soluble boron concentration of 2600ppm assuming voided guide tubes, the conclusion of the Section 6.4.2.1.1 applies, i.e. the maximum reactivity is corresponds to 100% moderator density.

- For 2600ppm soluble boron concentration with filled guide tubes, the results presented in Table 6.4.6 indicate that there is a maximum of the reactivity somewhere between 0.90 g/cm^3 and 1.00 g/cm^3 moderator density. However, a distinct maximum can not be

identified, as the reactivities in this range are very close. For the purpose of the calculations with 2600ppm soluble boron concentration, a moderator density of 0.93 g/cm^3 was chosen, which corresponds to the highest calculated reactivity listed in Table 6.4.6.

The calculations documented in this chapter also use soluble boron concentrations other than 1900 ppm and 2600 ppm in the MPC-32/32F. For the MPC-32 loaded with intact fuel only, soluble boron concentrations between 1300 ppm and 2600 ppm are used. For the MPC-32/32F loaded with intact fuel, damaged fuel and fuel debris, soluble boron concentrations between 1500 ppm and 2900 ppm are used. In order to determine the optimum moderation condition for each assembly class at the corresponding soluble boron level, evaluations are performed with filled and voided guide tubes, and for water densities of 1.0 g/cm^3 and 0.93 g/cm^3 for each class and enrichment level. Results for the MPC-32 loaded with intact fuel only are listed in Table 6.4.10 for an initial enrichment of 5.0 wt% ^{235}U and in Table 6.4.11 for an initial enrichment of 4.1 wt% ^{235}U. Corresponding results for the MPC-32/32F loaded with intact fuel, damaged fuel and fuel debris are listed in Table 6.4.14. The highest value listed in these tables for each assembly class is listed as the bounding value in Section 6.1.

6.4.2.2 Partial Flooding

As required by NUREG-1536, calculations in this section address partial flooding in the HI-STORM 100 System and demonstrate that the fully flooded condition is the most reactive.

The reactivity changes during the flooding process were evaluated in both the vertical and horizontal positions for all MPC designs. For these calculations, the cask is partially filled (at various levels) with full density (1.0 g/cc) water and the remainder of the cask is filled with steam consisting of ordinary water at partial density (0.002 g/cc), as suggested in NUREG-1536. Results of these calculations are shown in Table 6.4.2. In all cases, the reactivity increases monotonically as the water level rises, confirming that the most reactive condition is fully flooded.

6.4.2.3 Clad Gap Flooding

As required by NUREG-1536, the reactivity effect of flooding the fuel rod pellet-to-clad gap regions, in the fully flooded condition, has been investigated. Table 6.4.3 presents maximum k_{eff} values that demonstrate the positive reactivity effect associated with flooding the pellet-to-clad gap regions. These results confirm that it is conservative to assume that the pellet-to-clad gap regions are flooded. For all cases that involve flooding, the pellet-to-clad gap regions are assumed to be flooded with clean, unborated water.

6.4.2.4 Preferential Flooding

Two different potential conditions of preferential flooding are considered: preferential flooding of the MPC basket itself (i.e. different water levels in different basket cells), and preferential flooding involving Damaged Fuel Containers.

Preferential flooding of the MPC basket itself for any of the MPC fuel basket designs is not possible because flow holes are present on all four walls of each basket cell and on the two flux trap walls at both the top and bottom of the MPC basket. The flow holes are sized to ensure that they cannot be blocked by crud deposits (see Chapter 11). Because the fuel cladding temperatures remain below their design limits (as demonstrated in Chapter 4) and the inertial loading remains below 63g's (the inertial loadings associated with the design basis drop accidents discussed in Chapter 11 are limited to 45g's), the cladding remains intact (see Section 3.5). For damaged fuel assemblies and fuel debris, the assemblies or debris are pre-loaded into stainless steel Damaged Fuel Containers fitted with 250x250 fine mesh screens which prevent damaged fuel assemblies or fuel debris from blocking the basket flow holes. Therefore, the flow holes cannot be blocked.

However, when DFCs are present in the MPC, a condition could exist during the draining of the MPC, where the DFCs are still partly filled with water while the remainder of the MPC is dry. This condition would be the result of the water tension across the mesh screens. The maximum water level inside the DFCs for this condition is calculated from the dimensions of the mesh screen and the surface tension of water. The wetted perimeter of the screen openings is 50 ft per square inch of screen. With a surface tension of water of 0.005 lbf/ft, this results in a maximum pressure across the screen of 0.25 psi, corresponding to a maximum water height in the DFC of 7 inches. For added conservativism, a value of 12 inches is used. Assuming this condition, calculations are performed for all three possible DFC configurations:

- MPC-68 or MPC-68F with 68 DFCs (Assembly Classes 6x6A/B/C, 7x7A and 8x8A)
- MPC-68 or MPC-68FF with 16 DFCs (All BWR Assembly Classes)
- MPC-24E or MPC-24EF with 4 DFCs (All PWR Assembly Classes)
- *MPC-32 or MPC-32F with 8 DFCs (All PWR Assembly Classes)*

For each configuration, the case resulting in the highest maximum k_{eff} for the fully flooded condition (see Section 6.4.4) is re-analyzed assuming the preferential flooding condition. For these analyses, the lower 12 inches of the active fuel in the DFCs and the water region below the active fuel (see Figure 6.3.7) are filled with full density water (1.0 g/cc). The remainder of the cask is filled with steam consisting of ordinary water at partial density (0.002 g/cc). Table 6.4.4 lists the maximum k_{eff} for the ~~three~~*four* configurations in comparison with the maximum k_{eff} for the fully flooded condition. For all configurations, the preferential flooding condition results in a lower maximum k_{eff} than the fully flooded condition. Thus, the preferential flooding condition is bounded by the fully flooded condition.

Once established, the integrity of the MPC confinement boundary is maintained during all credible off-normal and accident conditions, and thus, the MPC cannot be flooded. In summary, it is concluded that the MPC fuel baskets cannot be preferentially flooded, and that the potential preferential flooding conditions involving DFCs are bounded by the result for the fully flooded condition listed in Section 6.4.4.

6.4.2.5 Design Basis Accidents

The analyses presented in Chapters 3 and 11 demonstrate that the damage resulting from the design basis accidents is limited to a loss of the water jacket for the HI-TRAC transfer cask and minor damage to the concrete radiation shield for the HI-STORM storage cask, which have no adverse effect on the design parameters important to criticality safety.

As reported in Chapter 3, Table 3.4.4, the minimum factor of safety for either MPC as a result of the hypothetical cask drop or tip-over accident is 1.1 against the Level D allowables for Subsection NG, Section III of the ASME Code. Therefore, because the maximum box wall stresses are well within the ASME Level D allowables, the flux-trap gap change will be insignificant compared to the characteristic dimension of the flux trap.

In summary, the design basis accidents have no adverse effect on the design parameters important to criticality safety, and therefore, there is no increase in reactivity as a result of any of the credible off-normal or accident conditions involving handling, packaging, transfer or storage. Consequently, the HI-STORM 100 System is in full compliance with the requirement of 10CRF72.124, which states that "before a nuclear criticality accident is possible, at least two unlikely, independent, and concurrent or sequential changes have occurred in the conditions essential to nuclear criticality safety."

6.4.3 Criticality Results

Results of the design basis criticality safety calculations for the condition of full flooding with water (limiting cases) are presented in section 6.2 and summarized in Section 6.1. To demonstrate the applicability of the HI-STAR analyses, results of the design basis criticality safety calculations for the HI-STAR cask (limiting cases) are also summarized in Section 6.1 for comparison. These data confirm that for each of the candidate fuel types and basket configurations the effective multiplication factor (k_{eff}), including all biases and uncertainties at a 95-percent confidence level, do not exceed 0.95 under all credible normal, off-normal, and accident conditions.

Additional calculations (CASMO-3) at elevated temperatures confirm that the temperature coefficients of reactivity are negative as shown in Table 6.3.1. This confirms that the calculations for the storage baskets are conservative.

In calculating the maximum reactivity, the analysis used the following equation:

$$k_{eff}^{max} = k_c + K_c\sigma_c + Bias + \sigma_B$$

where:

- ⇒ k_c is the calculated k_{eff} under the worst combination of tolerances;
- ⇒ K_c is the K multiplier for a one-sided statistical tolerance limit with 95% probability at the 95% confidence level [6.1.8]. Each final k_{eff} value calculated by MCNP4a (or KENO5a) is the result of averaging 100 (or more) cycle k_{eff} values, and thus, is based on a sample size of 100. The K multiplier corresponding to a sample size of 100 is 1.93. However, for this analysis a value of 2.00 was assumed for the K multiplier, which is larger (more conservative) than the value corresponding to a sample size of 100;
- ⇒ σ_c is the standard deviation of the calculated k_{eff}, as determined by the computer code (MCNP4a or KENO5a);
- ⇒ ***Bias*** is the systematic error in the calculations (code dependent) determined by comparison with critical experiments in Appendix 6.A; and
- ⇒ σ_B is the standard error of the bias (which includes the K multiplier for 95% probability at the 95% confidence level; see Appendix 6.A).

The critical experiment benchmarking and the derivation of the bias and standard error of the bias (95% probability at the 95% confidence level) are presented in Appendix 6.A.

6.4.4 Damaged Fuel and Fuel Debris

Damaged fuel assemblies and fuel debris are required to be loaded into Damaged Fuel Containers (DFCs) prior to being loaded into the MPC. ~~Four (4)~~*Five (5)* different DFC types with different cross sections are analyzed. Three (3) of these DFCs are designed for BWR fuel assemblies, ~~one (1) is~~*two (2) are* designed for PWR fuel assemblies. Two of the DFCs for BWR fuel are specifically designed for fuel assembly classes 6x6A, 6x6B, 6x6C, 7x7A and 8x8A. These assemblies have a smaller cross section, a shorter active length and a low initial enrichment of 2.7 wt% ^{235}U, and therefore a low reactivity. The analysis for these assembly classes is presented in the following Section 6.4.4.1. The remaining ~~two~~*three* DFCs are generic DFCs designed for all BWR and PWR assembly classes. The criticality analysis for these generic DFCs is presented in Section 6.4.4.2.

6.4.4.1 MPC-68, MPC-68F or MPC-68FF loaded with Assembly Classes 6x6A, 6x6B, 6x6C, 7x7A and 8x8A

This section only addresses criticality calculations and results for assembly classes 6x6A, 6x6B, 6x6C, 7x7A and 8x8A, loaded into the MPC-68, MPC-68F or MPC-68FF. Up to 68 DFCs with these assembly classes are permissible to be loaded into the MPC. Two different DFC types with slightly different cross-sections are analyzed. DFCs containing fuel debris must be stored in the MPC-68F or MPC-68FF. DFCs containing damaged fuel assemblies may be stored in either the MPC-68, MPC-68F or MPC-68FF. Evaluation of the capability of storing damaged fuel and fuel debris (loaded in DFCs) is limited to very low reactivity fuel in the MPC-68F. Because the MPC-68 and MPC-68FF have a higher specified ^{10}B loading, the evaluation of the MPC-68F conservatively bounds the storage of damaged BWR fuel assemblies in a standard MPC-68 or MPC-68FF. Although the maximum planar-average enrichment of the damaged fuel is limited to 2.7% ^{235}U as specified in ~~the Certificate of Compliance~~ *Section 2.1.9*, analyses have been made for three possible scenarios, conservatively assuming fuel[††] of 3.0% enrichment. The scenarios considered included the following:

1. Lost or missing fuel rods, calculated for various numbers of missing rods in order to determine the maximum reactivity. The configurations assumed for analysis are illustrated in Figures 6.4.2 through 6.4.8.
2. Broken fuel assembly with the upper segments falling into the lower segment creating a close-packed array (described as a 8x8 array). For conservatism, the array analytically retained the same length as the original fuel assemblies in this analysis. This configuration is illustrated in Figure 6.4.9.
3. Fuel pellets lost from the assembly and forming powdered fuel dispersed through a volume equivalent to the height of the original fuel. (Flow channel and clad material assumed to disappear).

Results of the analyses, shown in Table 6.4.5, confirm that, in all cases, the maximum reactivity is well below the regulatory limit. There is no significant difference in reactivity between the two DFC types. Collapsed fuel reactivity (simulating fuel debris) is low because of the reduced moderation. Dispersed powdered fuel results in low reactivity because of the increase in ^{238}U neutron capture (higher effective resonance integral for ^{238}U absorption).

The loss of fuel rods results in a small increase in reactivity (i.e., rods assumed to collapse, leaving a smaller number of rods still intact). The peak reactivity occurs for 8 missing rods, and a smaller (or larger) number of intact rods will have a lower reactivity, as indicated in Table 6.4.5.

---

†† 6x6A01 and 7x7A01 fuel assemblies were used as representative assemblies.

The analyses performed and summarized in Table 6.4.5 provides the relative magnitude of the effects on the reactivity. This information coupled with the maximum k_{eff} values listed in Table 6.1.3 and the conservatism in the analyses, demonstrate that the maximum k_{eff} of the damaged fuel in the most adverse post-accident condition will remain well below the regulatory requirement of $k_{eff} < 0.95$.

6.4.4.2 Generic BWR and PWR Damaged Fuel and Fuel Debris

The MPC-24E, MPC-24EF, *MPC-32, MPC-32F,* MPC-68 and MPC-68FF are designed to contain PWR and BWR damaged fuel and fuel debris, loaded into generic DFCs. The number of generic DFCs is limited to 16 for the MPC-68 and MPC-68FF, ~~and~~ to 4 for the MPC-24E and MPC-24EF, *and to 8 for the MPC-32 and MPC-32F.* The permissible locations of the DFCs are shown in Figure 6.4.11 for the MPC-68/68FF, ~~and~~ in Figure 6.4.12 for the MPC-24E/24EF *and in Figure 6.4.16 for the MPC-32/32F.*

Damaged fuel assemblies are assemblies with known or suspected cladding defects greater than pinholes or hairlines, or with missing rods, but excluding fuel assemblies with gross defects (for a full definition see ~~Section 1.1 of the Certificate of Compliance~~ *Table 1.0.1*). Therefore, apart from possible missing fuel rods, damaged fuel assemblies have the same geometric configuration as intact fuel assemblies and consequently the same reactivity. Missing fuel rods can result in a slight increase of reactivity. After a drop accident, however, it can not be assumed that the initial geometric integrity is still maintained. For a drop on either the top or bottom of the cask, the damaged fuel assemblies could collapse. This would result in a configuration with a reduced length, but increased amount of fuel per unit length. For a side drop, fuel rods could be compacted to one side of the DFC. In either case, a significant relocation of fuel within the DFC is possible, which creates a greater amount of fuel in some areas of the DFC, whereas the amount of fuel in other areas is reduced. Fuel debris can include a large variety of configurations ranging from whole fuel assemblies with severe damage down to individual fuel pellets.

In the cases of fuel debris or relocated damaged fuel, there is the potential that fuel could be present in axial sections of the DFCs that are outside the basket height covered with ~~Boral~~*the fixed neutron absorber*. However, in these sections, the DFCs are not surrounded by any intact fuel, only by basket cell walls, non-fuel hardware, water and for the MPC-68/68FF by a maximum of one other DFC. Studies have shown that this condition does not result in any significant effect on reactivity, compared to a condition where the damaged fuel and fuel debris is restricted to the axial section of the basket covered by ~~Boral~~*the fixed neutron absorber*. All calculations for generic BWR and PWR damaged fuel and fuel debris are therefore performed assuming that fuel is present only in the axial sections covered by ~~Boral~~*the fixed neutron absorber*, and the results are directly applicable to any situation where damaged fuel and fuel debris is located outside these sections in the DFCs.

To address all the situations listed above and identify the configuration or configurations leading to the highest reactivity, it is impractical to analyze a large number of different geometrical configurations for each of the fuel classes. Instead, a bounding approach is taken which is based on the analysis of regular arrays of bare fuel rods without cladding. Details and results of the analyses are discussed in the following sections.

All calculations for generic damaged fuel and fuel debris are performed using a full cask model with the maximum permissible number of Damaged Fuel Containers. For the MPC-68 and MPC-68FF, the model therefore contains 52 intact assemblies, and 16 DFCs in the locations shown in Figure 6.4.11. For the MPC-24E and MPC-24EF, the model consists of 20 intact assemblies, and 4 DFCs in the locations shown in Figure 6.4.12. *For the MPC-32 and MPC-32, the model consists of 24 intact assemblies, and 8 DFCs in the locations shown in Figure 6.4.16.* The bounding assumptions regarding the intact assemblies and the modeling of the damaged fuel and fuel debris in the DFCs are discussed in the following sections.

Note that since a modeling approach is used that bounds both damaged fuel and fuel debris without distinguishing between these two conditions, the term 'damaged fuel' as used throughout this chapter designates both damaged fuel and fuel debris.

6.4.4.2.1 Bounding Intact Assemblies

Intact BWR assemblies stored together with DFCs are limited to a maximum planar average enrichment of 3.7 wt% ^{235}U, regardless of the fuel class. The results presented in Table 6.1.7 are for different enrichments for each class, ranging between 2.7 and 4.2 wt% ^{235}U, making it difficult to identify the bounding assembly. Therefore, additional calculations were performed for the bounding assembly in each assembly class with a planar average enrichment of 3.7 wt%. The results are summarized in Table 6.4.7 and demonstrate that the assembly classes 9x9E and 9x9F have the highest reactivity. These two classes share the same bounding assembly (see footnotes for Tables 6.2.33 and 6.2.34 for further details). This bounding assembly is used as the intact BWR assembly for all calculations with DFCs.

Intact PWR assemblies stored together with DFCs *in the MPC-24E* are limited to a maximum enrichment of 4.0 wt% ^{235}U *without credit for soluble boron and to a maximum enrichment of 5.0 wt% with credit for soluble boron,* regardless of the fuel class. The results presented in Table 6.1.3 are for different enrichments for each class, ranging between 4.2 and 5.0 wt% ^{235}U, making it difficult to directly identify the bounding assembly. However, Table 6.1.4 shows results for an enrichment of 5.0 wt% for all fuel classes, with a soluble boron concentration of 300 ppm. The assembly class 15x15H has the highest reactivity. This is consistent with the results in Table 6.1.3, where the assembly class 15x15H is among the classes with the highest reactivity, but has the lowest initial enrichment. Therefore, *in the MPC-24E,* the 15x15H assembly is used as the intact PWR assembly for all calculations with DFCs.

Intact PWR assemblies stored together with DFCs in the MPC-32 are limited to a maximum enrichment of 5.0 wt%, regardless of the fuel class. Table 6.1.5 and Table 6.1.6 shows results for enrichments of 4.1 wt% and 5.0 wt%, respectively, for all fuel classes. Since different minimum soluble boron concentrations are used for different groups of assembly classes, the assembly class with the highest reactivity in each group is used as the intact assembly for the calculations with DFCs in the MPC-32. These assembly classes are

- *14x14C for all 14x14 assembly classes;*
- *15x15B for assembly classes 15x15A, B, C and G;*
- *15x15F for assembly classes 15x15D, E, F and H;*
- *16x16A; and*
- *17x17C for all 17x17 assembly classes.*

6.4.4.2.2 Bare Fuel Rod Arrays

A conservative approach is used to model both damaged fuel and fuel debris in the DFCs, using arrays of bare fuel rods:

- Fuel in the DFCs is arranged in regular, rectangular arrays of bare fuel rods, i.e. all cladding and other structural material in the DFC is replaced by water.

- *For cases with soluble boron, additional calculations are performed with reduced water density in the DFC. This is to demonstrate that replacing all cladding and other structural material with borated water is conservative.*

- The active length of these rods is chosen to be the maximum active fuel length of all fuel assemblies listed in Section 6.2, which is 155 inch for BWR fuel and 150 inch for PWR fuel.

- To ensure the configuration with optimum moderation and highest reactivity is analyzed, the amount of fuel per unit length of the DFC is varied over a large range. This is achieved by changing the number of rods in the array and the rod pitch. The number of rods are varied between 9 (3x3) and 189 (17x17) for BWR fuel, and between 64 (8x8) and 729 (27x27) for PWR fuel.

- Analyses are performed for the minimum, maximum and typical pellet diameter of PWR and BWR fuel.

This is a very conservative approach to model damaged fuel, and to model fuel debris configurations such as severely damaged assemblies and bundles of individual fuel rods, as the absorption in the cladding and structural material is neglected.

This is also a conservative approach to model fuel debris configurations such as bare fuel pellets due to the assumption of an active length of 155 inch (BWR) or 150 inch (PWR). The actual height of bare fuel pellets in a DFC would be significantly below these values due to the limitation of the fuel mass for each basket position.

To demonstrate the level of conservatism, additional analyses are performed with the DFC containing various realistic assembly configurations such as intact assemblies, assemblies with missing fuel rods and collapsed assemblies, i.e. assemblies with increased number of rods and decreased rod pitch.

As discussed in Section 6.4.4.2, all calculations are performed for full cask models, containing the maximum permissible number of DFCs together with intact assemblies.

As an example of the damaged fuel model used in the analyses, Figure 6.4.17 shows the basket cell of an MPC-32 with a DFC containing a 17x17 array of bare fuel rods.

Graphical presentations of the calculated maximum k_{eff} for ~~each~~*typical* case*s* as a function of the fuel mass per unit length of the DFC are shown in Figures 6.4.13 (BWR) and 6.4.14 (PWR, *MPC-24E/EF with pure water*). The results for the bare fuel rods show a distinct peak in the maximum k_{eff} at about 2 kg UO_2/inch for BWR fuel, and at about 3.5 kgUO_2/inch for PWR fuel.

The realistic assembly configurations are typically about 0.01 (delta-k) or more below the peak results for the bare fuel rods, demonstrating the conservatism of this approach to model damaged fuel and fuel debris configurations such as severely damaged assemblies and bundles of fuel rods.

For fuel debris configurations consisting of bare fuel pellets only, the fuel mass per unit length would be beyond the value corresponding to the peak reactivity. For example, for DFCs filled with a mixture of 60 vol% fuel and 40 vol% water the fuel mass per unit length is 3.36 kgUO_2/inch for the BWR DFC and 7.92 kgUO_2/inch for the PWR DFC. The corresponding reactivities are significantly below the peak reactivies. The difference is about 0.005 (delta-k) for BWR fuel and 0.01 (delta-k) or more for PWR fuel. Furthermore, the filling height of the DFC would be less than 70 inches in these examples due to the limitation of the fuel mass per basket position, whereas the calculation is conservatively performed for a height of 155 inch (BWR) or 150 inch (PWR). These results demonstrate that even for the fuel debris configuration of bare fuel pellets, the model using bare fuel rods is a conservative approach.

6.4.4.2.3 Distributed Enrichment in BWR Fuel

BWR fuel usually has an enrichment distribution in each planar cross section, and is characterized by the maximum planar average enrichment. For intact fuel it has been shown that

using the average enrichment for each fuel rod in a cross section is conservative, i.e. the reactivity is higher than calculated for the actual enrichment distribution (See Appendix 6.B). For damaged fuel assemblies, additional configurations are analyzed to demonstrate that the distributed enrichment does not have a significant impact on the reactivity of the damaged assembly under accident conditions. Specifically, the following two scenarios were analyzed:

- As a result of an accident, fuel rods with lower enrichment relocate from the top part to the bottom part of the assembly. This results in an increase of the average enrichment in the top part, but at the same time the amount of fuel in that area is reduced compared to the intact assembly.
- As a result of an accident, fuel rods with higher enrichment relocate from the top part to the bottom part of the assembly. This results in an increase of the average enrichment in the bottom part, and at the same time the amount of fuel in that area is increased compared to the intact assembly, leading to a reduction of the water content.

In both scenarios, a compensation of effects on reactivity is possible, as the increase of reactivity due to the increased planar average enrichment might be offset by the possible reduction of reactivity due to the change in the fuel to water ratio. A selected number of calculations have been performed for these scenarios and the results show that there is only a minor change in reactivity. These calculations are shown in Figure 6.4.13 in the group of the explicit assemblies. Consequently, it is appropriate to qualify damaged BWR fuel assemblies and fuel debris based on the maximum planar average enrichment. For assemblies with missing fuel rods, this maximum planar average enrichment has to be determined based on the enrichment and number of rods still present in the assembly when loaded into the DFC.

6.4.4.2.4 Results for MPC-68 and MPC-68FF

The MPC-68 and MPC-68FF allows the storage of up to sixteen DFCs in the shaded cells on the periphery of the basket shown in Figure 6.4.11. In the MPC-68FF, up to 8 of these cells may contain DFCs with fuel debris. The various configurations outlined in Sections 6.4.4.2.2 and 6.4.4.2.3 are analyzed with an enrichment of the intact fuel of 3.7% ^{235}U and an enrichment of damaged fuel or fuel debris of 4.0% ^{235}U. For the intact assembly, the bounding assembly of the 9x9E and 9x9F fuel classes was chosen. This assembly has the highest reactivity of all BWR assembly classes for the initial enrichment of 3.7 wt% ^{235}U, as demonstrated in Table 6.4.7. The results for the various configurations are summarized in Figure 6.4.13 and in Table 6.4.8. Figure 6.4.13 shows the maximum k_{eff}, including bias and calculational uncertainties, for various actual and hypothetical damaged fuel or fuel debris configurations as a function of the fuel mass per unit length of the DFC. Table 6.4.8 lists the highest maximum k_{eff} for the various configurations. All maximum k_{eff} values are below the 0.95 regulatory limit.

6.4.4.2.5 Results for MPC-24E and MPC-24EF

The MPC-24E allows the storage of up to four DFCs with damaged fuel in the four outer fuel baskets cells shaded in Figure 6.4.12. The MPC-24EF allows storage of up to four DFCs with damaged fuel or fuel debris in these locations. These locations are designed with a larger box ID to accommodate the DFCs. For an enrichment of 4.0 wt% ^{235}U for the intact fuel, damaged fuel and fuel debris, *and assuming no soluble boron,* the results for the various configurations outlined in Section 6.4.4.2.2 are summarized in Figure 6.4.14 and in Table 6.4.9. Figure 6.4.14 shows the maximum k_{eff}, including bias and calculational uncertainties, for various actual and hypothetical damaged fuel and fuel debris configurations as a function of the fuel mass per unit length of the DFC. For the intact assemblies, the 15x15H assembly class was chosen. This assembly class has the highest reactivity of all PWR assembly classes for a given initial enrichment. This is demonstrated in Table 6.1.4. Table 6.4.9 lists the highest maximum k_{eff} for the various configurations. All maximum k_{eff} values are below the 0.95 regulatory limit.

For an enrichment of 5.0 wt% ^{235}U for the intact fuel, damaged fuel and fuel debris, a minimum soluble boron concentration of 600 ppm is required. For this condition, calculations are performed for various hypothetical fuel debris configurations (i.e. bare fuel rods) as a function of the fuel mass per unit length of the DFC. Additionally, calculations are performed with reduced water densities in the DFC. The various conditions of damaged fuel, such as assemblies with missing rods or collapsed assemblies, were not analyzed, since the results in Figure 6.4.14 clearly demonstrate that these conditions are bounded by the hypothetical model for fuel debris based on regular arrays of bare fuel rods. Again, the 15x15H assembly class was chosen as the intact assembly since this assembly class has the highest reactivity of all PWR assembly classes as demonstrated in Table 6.1.4. The results are summarized in Table 6.4.12. Similar to the calculations with pure water (see Figure 6.4.14), the results for borated water show a distinct peak of the maximum k_{eff} as a function of the fuel mass per unit length. Therefore, f~~F~~or each condition, the table lists only the highest maximum k_{eff}, including bias and calculational uncertainties, i.e. the point of optimum moderation. The results show that the reactivity decreases with decreasing water density. This demonstrates that replacing all cladding and other structural material with water is conservative even in the presence of soluble boron in the water. All maximum k_{eff} values are below the 0.95 regulatory limit.

6.4.4.2.6 Results for MPC-32 and MPC-32F

The MPC-32 allows the storage of up to eight DFCs with damaged fuel in the outer fuel basket cells shaded in Figure 6.4.16. The MPC-32F allows storage of up to eight DFCs with damaged fuel or fuel debris in these locations. For the MPC-32 and MPC-32F, additional cases are analyzed due to the high soluble boron level required for this basket:

- *The assembly classes of the intact assemblies are grouped, and minimum required soluble boron levels are determined separately for each group. The analyses are*

performed for the bounding assembly class in each group. The bounding assembly classes are listed in Section 6.4.4.2.1.

- *Evaluations of conditions with voided and filled guide tubes and various water densities in the MPC and DFC are performed to identify the most reactive condition.*

In general, all calculations performed for the MPC-32 show the same principal behavior as for the MPC-24 (see Figure 6.4.14), i.e. the reactivity as a function of the fuel mass per unit length for the bare fuel rod array shows a distinct peak. Therefore, for each condition analyzed, only the highest maximum k_{eff}, i.e. the calculated peak reactivity, is listed in the tables. Evaluations of different diameters of the bare fuel pellets and the reduced water density in the DFC have been performed for a representative case using the 15x15F assembly class as the intact assembly, with voided guide tubes, a water density of 1.0 g/cc in the DFC and MPC, 2900 ppm soluble boron, and an enrichment of 5.0 wt% ^{235}U for the intact and damaged fuel and fuel debris. For this case, results are summarized in Table 6.4.13. For each condition, the table lists the highest maximum k_{eff}, including bias and calculational uncertainties, i.e. the point of optimum moderation. The results show that the fuel pellet diameter in the DFC has an insignificant effect on reactivity, and that reactivity decreases with decreasing water density. The latter demonstrates that replacing all cladding and other structural material with water is conservative even in the presence of soluble boron in the water. Therefore, a typical fuel pellet diameter and a water density of 1.0 in the DFCs are used for all further analyses. Two enrichment levels are analyzed, 4.1 wt% ^{235}U and 5.0 wt% ^{235}U, consistent with the analyses for intact fuel only. In any calculation, the same enrichment is used for the intact fuel and the damaged fuel and fuel debris. For both enrichment levels, analyses are performed with voided and filled guide tubes, each with water densities of 0.93 and 1.0 g/cm^3 in the MPC. In all cases, the water density inside the DFCs is assumed to be 1.0 g/cm^3, since this is the most reactive condition as shown in Table 6.4.13. Results are summarized in Table 6.4.14. For each group of assembly classes, the table shows the soluble boron level and the highest maximum k_{eff} for the various moderation conditions of the intact assembly. The highest maximum k_{eff} is the highest value of any of the hypothetical fuel debris configurations, i.e. various arrays of bare fuel rods. All maximum k_{eff} values are below the 0.95 regulatory limit. Conditions of damaged fuel such as assemblies with missing rods or collapsed assemblies were not analyzed in the MPC-32, since the results in Figure 6.4.14 clearly demonstrate that these conditions are bounded by the hypothetical model for fuel debris based on regular arrays of bare fuel rods.

6.4.5 Fuel Assemblies with Missing Rods

For fuel assemblies that are qualified for damaged fuel storage, missing and/or damaged fuel rods are acceptable. However, for fuel assemblies to meet the limitations of intact fuel assembly storage, missing fuel rods must be replaced with dummy rods that displace a volume of water that is equal to, or larger than, that displaced by the original rods.

6.4.6 Thoria Rod Canister

The Thoria Rod Canister is similar to a DFC with an internal separator assembly containing 18 intact fuel rods. The configuration is illustrated in Figure 6.4.15. The k_{eff} value for an MPC-68F filled with Thoria Rod Canisters is calculated to be 0.1813. This low reactivity is attributed to the relatively low content in ^{235}U (equivalent to UO_2 fuel with an enrichment of approximately 1.7 wt% ^{235}U), the large spacing between the rods (the pitch is approximately 1", the cladding OD is 0.412") and the absorption in the separator assembly. Together with the maximum k_{eff} values listed in Tables 6.1.7 and 6.1.8 this result demonstrates, that the k_{eff} for a Thoria Rod Canister loaded into the MPC-68 or the MPC-68F together with other approved fuel assemblies or DFCs will remain well below the regulatory requirement of $k_{eff} < 0.95$.

6.4.7 Sealed Rods replacing BWR Water Rods

Some BWR fuel assemblies contain sealed rods filled with a non-fissile material instead of water rods. Compared to the configuration with water rods, the configuration with sealed rods has a reduced amount of moderator, while the amount of fissile material is maintained. Thus, the reactivity of the configuration with sealed rods will be lower compared to the configuration with water rods. Any configuration containing sealed rods instead of water rods is therefore bounded by the analysis for the configuration with water rods and no further analysis is required to demonstrate the acceptability. Therefore, for all BWR fuel assemblies analyzed, it is permissible that water rods are replaced by sealed rods filled with a non-fissile material.

6.4.8 Non-fuel Hardware in PWR Fuel Assemblies

Non-fuel hardware such as Thimble Plugs (TPs), Burnable Poison Rod Assemblies (BPRAs), Control Rod Assemblies (CRAs), Axial Power Shaping Rods (APSRs) and similar devices are permitted for storage with all PWR fuel types. Non-fuel hardware is inserted in the guide tubes of the assemblies. For pure water, the reactivity of any PWR assembly with inserts is bounded by (i.e. lower than) the reactivity of the same assembly without the insert. This is due to the fact that the insert reduces the amount of moderator in the assembly, while the amount of fissile material remains unchanged. This conclusion is supported by the calculation listed in Table 6.2.4, which shows a significant reduction in reactivity as a result of voided guide tubes, i.e. the removal of the water from the guide tubes.

With the presence of soluble boron in the water, non-fuel hardware not only displaces water, but also the neutron absorber in the water. It is therefore possible that the insertion results in an increase of reactivity, specifically for higher soluble boron concentrations. As a bounding

approach for the presence of non-fuel hardware, analyses were performed with empty (voided) guide tubes, i.e. any absorption of the hardware is neglected. If assemblies contain an instrument tube, this tube remains filled with borated water. Table 6.4.6 shows results for the variation in water density for cases with filled and voided guide tubes. These results show that the optimum moderator density depends on the soluble boron concentration, and on whether the guide tubes are filled or assumed empty. For the MPC-24 with 400 ppm and the MPC-32 with 1900 ppm, voiding the guide tubes results in a reduction of reactivity. All calculations for the MPC-24 and MPC-24E~~, and for the MPC-32 with 1900 ppm~~ are therefore performed with water in the guide tubes. For the MPC-32 with 2600 ppm, the reactivity for voided guide tubes slightly exceeds the reactivity for filled guide tubes. However, this effect is not consistent across all assembly classes. Table 6.4.10, *Table 6.4.11 and Table 6.4.14* shows results with filled and voided guide tubes for all assembly classes in the MPC-32/*32F* at ~~2600ppm~~*4.1 wt%* ^{235}U *and 5.0 wt%* ^{235}U. Some classes show an increase, other classes show a decrease as a result of voiding the guide tubes. Therefore, for the results presented in the Section 6.1, Table *6.1.5, Table* 6.1.6 *and Table 6.1.12*, the maximum value for each class is chosen *for each enrichment level.*

In summary, from a criticality safety perspective, non-fuel hardware inserted into PWR assemblies are acceptable for all allowable PWR types, and, depending on the assembly class, can increase the safety margin.

6.4.9 Neutron Sources in Fuel Assemblies

Fuel assemblies containing start-up neutron sources are permitted for storage in the HI-STORM 100 System. The reactivity of a fuel assembly is not affected by the presence of a neutron source (other than by the presence of the material of the source, which is discussed later). This is true because in a system with a keff less than 1.0, any given neutron population at any time, regardless of its origin or size, will decrease over time. Therefore, a neutron source of any strength will not increase reactivity, but only the neutron flux in a system, and no additional criticality analyses are required. Sources are inserted as rods into fuel assemblies, i.e. they replace either a fuel rod or water rod (moderator). Therefore, the insertion of the material of the source into a fuel assembly will not lead to an increase of reactivity either.

6.4.10 Applicability of HI-STAR Analyses to HI-STORM 100 System

Calculations previously supplied to the NRC in applications for the HI-STAR 100 System (Docket Numbers 71-9261 and 72-1008) are directly applicable to the HI-STORM storage and HI-TRAC transfer casks. The MPC designs are identical. The cask systems differ only in the overpack shield material. The limiting condition for the HI-STORM 100 System is the fully flooded HI-TRAC transfer cask. As demonstrated by the comparative calculations presented in Tables 6.1.1 through 6.1.8, the shield material in the overpack (steel and lead for HI-TRAC, steel

for HI-STAR) has a negligible impact on the eigenvalue of the cask systems. As a result, this analysis for the 125-ton HI-TRAC transfer cask is applicable to the 100-ton HI-TRAC transfer cask. In all cases, for the reference fuel assemblies, the maximum k_{eff} values are in good agreement and are conservatively less than the limiting k_{eff} value (0.95).

6.4.11 Fixed Neutron Absorber Material

The MPCs in the HI-STORM 100 System can be manufactured with one of two possible neutron absorber materials: Boral or Metamic. Both materials are made of aluminum and B_4C powder. Boral has an inner core consisting of B_4C and aluminum between two outer layers consisting of aluminum only. This configuration is explicitly modeled in the criticality evaluation and shown in Figures 6.3.1 through 6.3.3 for each basket. Metamic is a single layer material with the same overall thickness and the same credited ^{10}B loading (in g/cm^2) for each basket. The majority of the criticality evaluations documented in this chapter are performed using Boral as the fixed neutron absorber. For a selected number of bounding cases, analyses are also performed using Metamic instead of Boral. The results for these cases are listed in Table 6.4.15, together with the corresponding result using Boral and the difference between the two materials for each case. Individual cases show small differences for the two materials. However, the differences are mostly below two times the standard deviation (the standard deviation is about 0.0008 for all cases in Table 6.4.15), indicating that the results are statistically equivalent. Furthermore, the average difference is well below one standard deviation, and all cases are below the regulatory limit of 0.95. In some cases listed in Table 6.4.15, the reactivity difference between Metamic and Boral might be larger than expected for two equivalent materials. Also, for four out of the five cases with MPC-24 type baskets, Metamic shows the higher reactivity, which could potentially indicate a trend rather than a statistical variation. Therefore, in order to confirm that the materials are equivalent, a second set of calculations was performed for Metamic, which was statistically independent from the set shown in Table 6.4.15. This was achieved by selecting a different starting value for the random number generator in the Monte Carlo calculations. The second set also shows some individual variations of the differences, and a low average difference. However, there is no apparent trend regarding the MPC-24 type baskets compared to the MPC-32 and MPC-68, and the maximum positive reactivity difference for Metamic in an MPC-24 type basket is only 0.0005. Overall, the calculations demonstrate that the two fixed neutron absorber materials are identical from a criticality perspective. All results obtained for Boral are therefore directly applicable to Metamic and no further evaluations using Metamic are required.

Table 6.4.1

MAXIMUM REACTIVITIES WITH REDUCED WATER DENSITIES FOR CASK ARRAYS†

	Water Density		MCNP4a Maximum k_{eff} ††	
Case Number	Internal	External	MPC-24 (17x17A01 @ 4.0%)	MPC-68 (8x8C04 @ 4.2%)
1	100%	single cask	0.9368	0.9348
2	100%	100%	0.9354	0.9339
3	100%	70%	0.9362	0.9339
4	100%	50%	0.9352	0.9347
5	100%	20%	0.9372	0.9338
6	100%	10%	0.9380	0.9336
7	100%	5%	0.9351	0.9333
8	100%	0%	0.9342	0.9338
9	70%	0%	0.8337	0.8488
10	50%	0%	0.7426	0.7631
11	20%	0%	0.5606	0.5797
12	10%	0%	0.4834	0.5139
13	5%	0%	0.4432	0.4763
14	10%	100%	0.4793	0.4946

† For an infinite square array of casks with 60cm spacing between cask surfaces.

†† Maximum k_{eff} includes the bias, uncertainties, and calculational statistics, evaluated for the worst case combination of manufacturing tolerances.

Table 6.4.2

REACTIVITY EFFECTS OF PARTIAL CASK FLOODING

MPC-24 (17x17A01 @ 4.0% ENRICHMENT) (no soluble boron)			
Flooded Condition (% Full)	Vertical Orientation	Flooded Condition (% Full)	Horizontal Orientation
25	0.9157	25	0.8766
50	0.9305	50	0.9240
75	0.9330	75	0.9329
100	0.9368	100	0.9368
MPC-68 (8x8C04 @ 4.2% ENRICHMENT)			
Flooded Condition (% Full)	Vertical Orientation	Flooded Condition (% Full)	Horizontal Orientation
25	0.9132	23.5	0.8586
50	0.9307	50	0.9088
75	0.9312	76.5	0.9275
100	0.9348	100	0.9348
MPC-32 (15x15F @ 5.0 % ENRICHMENT) 2600ppm Soluble Boron			
Flooded Condition (% Full)	Vertical Orientation	Flooded Condition (% Full)	Horizontal Orientation
25	0.8927	31.25	0.9213
50	0.9215	50	0.9388
75	0.9350	68.75	0.9401
100	0.9445	100	0.9445

Notes:

1. All values are maximum k_{eff} which include bias, uncertainties, and calculational statistics, evaluated for the worst case combination of manufacturing tolerances.

Table 6.4.3

REACTIVITY EFFECT OF FLOODING THE PELLET-TO-CLAD GAP

Pellet-to-Clad Condition	MPC-24 17x17A01 4.0% Enrichment	MPC-68 8x8C04 4.2% Enrichment
dry	0.9295	0.9279
flooded with unborated water	0.9368	0.9348

Notes:

1. All values are maximum k_{eff} which includes bias, uncertainties, and calculational statistics, evaluated for the worst case combination of manufacturing tolerances.

Table 6.4.4

REACTIVITY EFFECT OF PREFERENTIAL FLOODING OF THE DFCs

DFC Configuration	Preferential Flooding	Fully Flooded
MPC-68 or MPC-68F with 68 DFCs (Assembly Classes 6x6A/B/C, 7x7A and 8x8A)	0.6560	0.7857
MPC-68 or MPC-68FF with 16 DFCs (All BWR Assembly Classes)	0.6646	0.9328
MPC-24E or MPC-24EF with 4 DFCs (All PWR Assembly Classes)	0.7895	0.9480
MPC-32 or MPC-32 with 8 DFCs (All PWR Assembly Classes)	*0.7213*	*0.9378*

Notes:

1. All values are maximum k_{eff} which includes bias, uncertainties, and calculational statistics, evaluated for the worst case combination of manufacturing tolerances.

Table 6.4.5

MAXIMUM k_{eff} VALUES† IN THE DAMAGED FUEL CONTAINER

Condition	MCNP4a Maximum†† k_{eff}	
	DFC Dimensions: ID 4.93" THK. 0.12"	DFC Dimensions: ID 4.81" THK. 0.11"
6x6 Fuel Assembly		
6x6 Intact Fuel	0.7086	0.7016
w/32 Rods Standing	0.7183	0.7117
w/28 Rods Standing	0.7315	0.7241
w/24 Rods Standing	0.7086	0.7010
w/18 Rods Standing	0.6524	0.6453
Collapsed to 8x8 array	0.7845	0.7857
Dispersed Powder	0.7628	0.7440
7x7 Fuel Assembly		
7x7 Intact Fuel	0.7463	0.7393
w/41 Rods Standing	0.7529	0.7481
w/36 Rods Standing	0.7487	0.7444
w/25 Rods Standing	0.6718	0.6644

† These calculations were performed with a planar-average enrichment of 3.0% and a ^{10}B loading of 0.0067 g/cm^2, which is 75% of a minimum ^{10}B loading of 0.0089 g/cm^2. The minimum ^{10}B loading in the MPC-68F is 0.010 g/cm^2. Therefore, the listed maximum k_{eff} values are conservative

†† Maximum k_{eff} includes bias, uncertainties, and calculational statistics, evaluated for the worst case combination of manufacturing tolerances.

Table 6.4.6

MAXIMUM k_{eff} VALUES WITH REDUCED BORATED WATER DENSITIES

Internal Water Density† in g/cm³	Maximum k_{eff}				
	MPC-24 (400ppm) @ 5.0 %	MPC-32 (1900ppm) @ 4.1 %		MPC-32 (2600ppm) @ 5.0 %	
Guide Tubes	filled	filled	void	filled	void
1.005	NC††	0.9403	0.9395	NC	0.9481
1.00	0.9314	0.9411	0.9400	0.9445	0.9483
0.99	NC	0.9393	0.9396	0.9438	0.9462
0.98	0.9245	0.9403	0.9376	0.9447	0.9465
0.97	NC	0.9397	0.9391	0.9453	0.9476
0.96	NC	NC	NC	0.9446	0.9466
0.95	0.9186	0.9380	0.9384	0.9451	0.9468
0.94	NC	NC	NC	0.9445	0.9467
0.93	0.9130	0.9392	0.9352	0.9465	0.9460
0.92	NC	NC	NC	0.9458	0.9450
0.91	NC	NC	NC	0.9447	0.9452
0.90	0.9061	0.9384	NC	0.9449	0.9454
0.80	0.8774	0.9322	NC	0.9431	0.9390
0.70	0.8457	0.9190	NC	0.9339	0.9259
0.60	0.8095	0.8990	NC	0.9194	0.9058
0.40	0.7225	0.8280	NC	0.8575	0.8410
0.20	0.6131	0.7002	NC	0.7421	0.7271
0.10	0.5486	0.6178	NC	0.6662	0.6584

† External moderator is modeled at 0%. This is consistent with the results demonstrated in Table 6.4.1.
†† NC: Not Calculated

Table 6.4.7

MAXIMUM k_{eff} VALUES FOR INTACT BWR FUEL ASSEMBLIES WITH A MAXIMUM PLANAR AVERAGE ENRICHMENT OF 3.7 wt% ^{235}U

Fuel Assembly Class	Maximum k_{eff}
6x6A	0.8287
6x6C	0.8436
7x7A	0.8399
7x7B	0.9109
8x8A	0.8102
8x8B	0.9131
8x8C	0.9115
8x8D	0.9125
8x8E	0.9049
8x8F	0.9233
9x9A	0.9111
9x9B	0.9134
9x9C	0.9103
9x9D	0.9096
9x9E	0.9237
9x9F	0.9237
9x9G	0.9005
10x10A	0.9158
10x10B	0.9156
10x10C	0.9152
10x10D	0.9182
10x10E	0.8970

Table 6.4.8

MAXIMUM k_{eff} VALUES IN THE GENERIC BWR DAMAGED FUEL CONTAINER FOR A MAXIMUM INITIAL ENRICHMENT OF 4.0 wt% ^{235}U FOR DAMAGED FUEL AND 3.7 wt% ^{235}U FOR INTACT FUEL

Model Configuration inside the DFC	**Maximum k_{eff}**
Intact Assemblies (4 assemblies analyzed)	0.9241
Assemblies with missing rods (7 configurations analyzed)	0.9240
Assemblies with distributed enrichment (4 configurations analyzed)	0.9245
Collapsed Assemblies (6 configurations analyzed)	0.9258
Regular Arrays of Bare Fuel Rods (31 configurations analyzed)	0.9328

Table 6.4.9

MAXIMUM k_{eff} VALUES IN *THE MPC-24E/EF WITH* THE GENERIC PWR DAMAGED FUEL CONTAINER FOR A MAXIMUM INITIAL ENRICHMENT OF 4.0 wt% ^{235}U *AND NO SOLUBLE BORON.*

Model Configuration inside the DFC	**Maximum k_{eff}**
Intact Assemblies (2 assemblies analyzed)	0.9340
Assemblies with missing rods (4 configurations analyzed)	0.9350
Collapsed Assemblies (6 configurations analyzed)	0.9360
Regular Arrays of Bare Fuel Rods (36 configurations analyzed)	0.9480

Table 6.4.10

MAXIMUM k_{eff} VALUES WITH FILLED AND VOIDED GUIDE TUBES
FOR THE MPC-32 AT 5.0 wt% ENRICHMENT

Fuel Class	***Minimum Soluble Boron Content (ppm)***	**MPC-32 ~~(2600ppm)~~@ 5.0 %**			
		Guide Tubes Filled, ~~Moderator Density 0.93~~		**Guide Tubes Voided, ~~Moderator Density 1.00~~**	
		1.0 g/cm³	***0.93 g/cm³***	***1.0 g/cm³***	***0.93 g/cm³***
14x14A	*1900*	*0.8984*	*0.9000*	*0.8953*	*0.8943*
14x14B	*1900*	*0.9210*	*0.9214*	*0.9164*	*0.9118*
14x14C	*1900*	*0.9371*	*0.9376*	*0.9480*	*0.9421*
14x14D	*1900*	*0.9050*	*0.9027*	*0.8947*	*0.8904*
14x14E	*1900*	*0.7415*	*0.7301*	*n/a*	*n/a*
15x15A	*2500*	*0.9210*	*0.9223*	*0.9230*	*0.9210*
15x15B	*2500*	*0.9402*	*0.9420*	*0.9429*	*0.9421*
15x15C	*2500*	*0.9258*	*0.9292*	*0.9307*	*0.9293*
15x15D	*2600*	*0.9426*	0.9419	0.9466	*0.9440*
15x15E	*2600*	*0.9394*	0.9415	0.9434	*0.9442*
15x15F	*2600*	*0.9445*	0.9465	0.9483	*0.9460*
15x15G	*2500*	*0.9228*	*0.9244*	*0.9251*	*0.9243*
15X15H	*2600*	*0.9271*	0.9301	0.9317	*0.9333*
16X16A	*1900*	*0.9460*	*0.9450*	*0.9474*	*0.9434*
17x17A	*2600*	*0.9105*	0.9145	0.9160	*0.9161*
17x17B	*2600*	*0.9345*	0.9358	0.9371	*0.9356*
17X17C	*2600*	*0.9417*	0.9431	0.9437	*0.9430*

Table 6.4.11

MAXIMUM k_{eff} VALUES WITH FILLED AND VOIDED GUIDE TUBES FOR THE MPC-32 AT 4.1 wt% ENRICHMENT

Fuel Class	Minimum Soluble Boron Content (ppm)	MPC-32 @ 4.1 %			
		Guide Tubes Filled		Guide Tubes Voided	
		1.0 g/cm^3	0.93 g/cm^3	1.0 g/cm^3	0.93 g/cm^3
14x14A	1300	0.9041	0.9029	0.8954	0.8939
14x14B	1300	0.9257	0.9205	0.9128	0.9074
14x14C	1300	0.9402	0.9384	0.9423	0.9365
14x14D	1300	0.8970	0.8943	0.8836	0.8788
14x14E	1300	0.7340	0.7204	n/a	n/a
15x15A	1800	0.9199	0.9206	0.9193	0.9134
15x15B	1800	0.9397	0.9387	0.9385	0.9347
15x15C	1800	0.9266	0.9250	0.9264	0.9236
15x15D	1900	0.9375	0.9384	0.9380	0.9329
15x15E	1900	0.9348	0.9340	0.9365	0.9336
15x15F	1900	0.9411	0.9392	0.9400	0.9352
15x15G	1800	0.9147	0.9128	0.9125	0.9062
15X15H	1900	0.9267	0.9274	0.9276	0.9268
16X16A	1300	0.9468	0.9425	0.9433	0.9384
17x17A	1900	0.9105	0.9111	0.9106	0.9091
17x17B	1900	0.9309	0.9307	0.9297	0.9243
17X17C	1900	0.9355	0.9347	0.9350	0.9308

Table 6.4.12

MAXIMUM k_{eff} VALUES IN THE MPC-24E/24EF WITH THE GENERIC PWR DAMAGED FUEL CONTAINER FOR A MAXIMUM INITIAL ENRICHMENT OF 5.0 wt% ^{235}U AND 600 PPM SOLUBLE BORON.

Water Density inside the DFC	***Bare Fuel Pellet Diameter***	***Maximum k_{eff}***
1.00	*minimum*	*0.9185*
1.00	*typical*	*0.9181*
1.00	*maximum*	*0.9171*
0.95	*typical*	*0.9145*
0.90	*typical*	*0.9125*
0.60	*typical*	*0.9063*
0.10	*typical*	*0.9025*
0.02	*typical*	*0.9025*

Table 6.4.13

MAXIMUM k_{eff} VALUES IN THE MPC-32/32F WITH THE GENERIC PWR DAMAGED FUEL CONTAINER FOR A MAXIMUM INITIAL ENRICHMENT OF 5.0 wt% ^{235}U, 2900 PPM SOLUBLE BORON AND THE 15x15F ASSEMBLY CLASS AS INTACT ASSEMBLY.

Water Density inside the DFC	*Bare Fuel Pellet Diameter*	*Maximum k_{eff}*
1.00	*minimum*	*0.9374*
1.00	*typical*	*0.9372*
1.00	*maximum*	*0.9373*
0.95	*typical*	*0.9369*
0.90	*typical*	*0.9365*
0.60	*typical*	*0.9308*
0.10	*typical*	*0.9295*
0.02	*typical*	*0.9283*

Table 6.4.14

BOUNDING MAXIMUM k_{eff} VALUES FOR THE MPC-32 AND MPC-32F WITH UP TO 8 DFCs UNDER VARIOUS MODERATION CONDITIONS.

Fuel Assembly Class of Intact Fuel	Initial Enrichment (wt% ^{235}U)	Minimum Soluble Boron Content (ppm)	Maximum k_{eff}			
			Filled Guide Tubes		Voided Guide Tubes	
			1.0 g/cm^3	0.93 g/cm^3	1.0 g/cm^3	0.93 g/cm^3
14x14A through 14x14E	4.1	1500	0.9277	0.9283	0.9336	0.9298
	5.0	2300	0.9139	0.9180	0.9269	0.9262
15x15A, B, C, G	4.1	1900	0.9345	0.9350	0.9350	0.9326
	5.0	2700	0.9307	0.9346	0.9347	0.9365
15x15D, E, F, H	4.1	2100	0.9322	0.9336	0.9340	0.9329
	5.0	2900	0.9342	0.9375	0.9385	0.9397
16x16A	4.1	1500	0.9322	0.9321	0.9335	0.9302
	5.0	2300	0.9198	0.9239	0.9289	0.9267
17x17A, B, C	4.1	2100	0.9284	0.9290	0.9294	0.9285
	5.0	2900	0.9308	0.9338	0.9355	0.9367

Table 6.4.15

COMPARISON OF MAXIMUM k_{eff} VALUES FOR DIFFERENT FIXED NEUTRON ABSORBER MATERIALS

Case	*Maximum k_{eff}*		*Reactivity Difference*
	BORAL	*METAMIC*	
MPC-68, Intact Assemblies	*0.9457*	*0.9452*	*-0.0005*
MPC-68, with 16 DFCs	*0.9328*	*0.9315*	*-0.0013*
MPC-68F with 68 DFCs	*0.8021*	*0.8019*	*-0.0002*
MPC-24, 0ppm	*0.9478*	*0.9491*	*+0.0013*
MPC-24, 400ppm	*0.9447*	*0.9457*	*+0.0010*
MPC-24E, Intact Assemblies, 0ppm	*0.9468*	*0.9494*	*+0.0026*
MPC-24E, Intact Assemblies, 300ppm	*0.9399*	*0.9410*	*+0.0011*
MPC-24E, with 4 DFCs, 0ppm	*0.9480*	*0.9471*	*-0.0009*
MPC-32, Intact Assemblies, 1900ppm	*0.9411*	*0.9397*	*-0.0014*
MPC-32, Intact Assemblies, 2600ppm	*0.9483*	*0.9471*	*-0.0012*
Average Difference			***+0.0001***

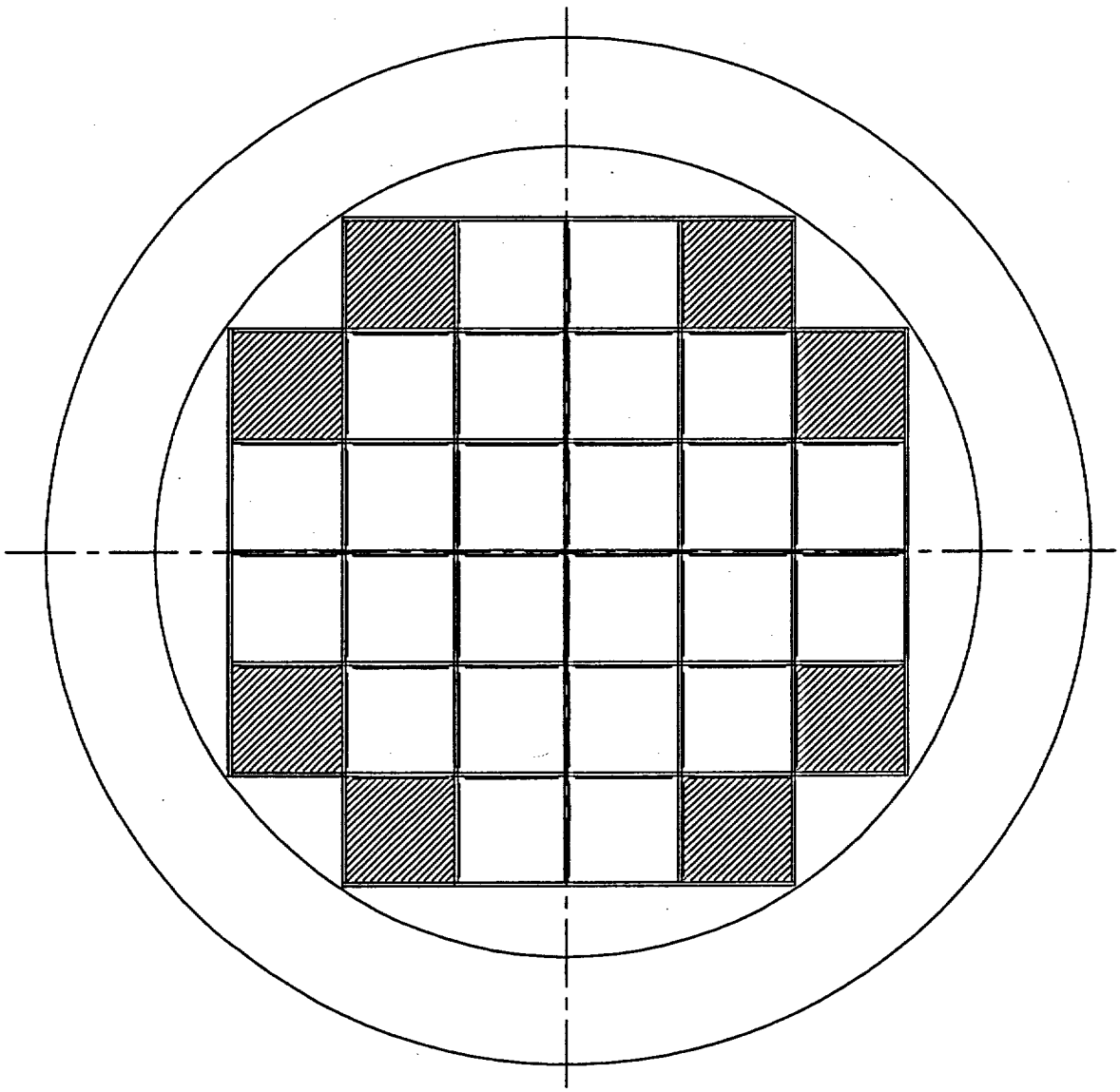


FIGURE 6.4.16; LOCATIONS OF THE DAMAGED FUEL CONTAINERS IN THE MPC-32.

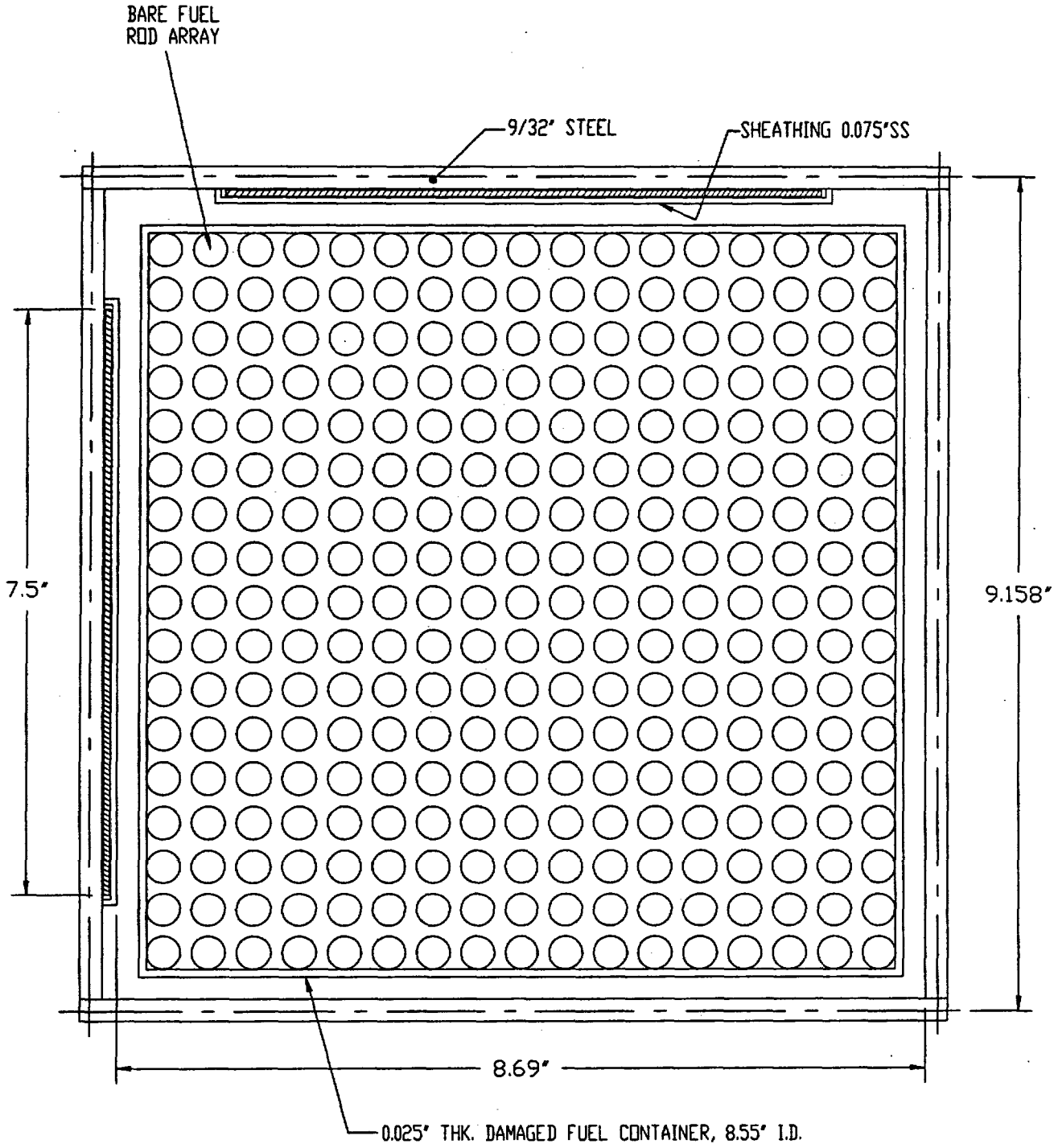


FIGURE 6.4.17; DAMAGED FUEL/FUEL DERIS CALCULATION MODEL (PLANAR CROSS-SECTION) WITH BARE FUEL RODS IN THE MPC-32 BASKET

(SEE CHAPTER 1 FOR TRUE BASKET DIMENSIONS)

APPENDIX 6.C: CALCULATIONAL SUMMARY

The following table lists the maximum k_{eff} (including bias, uncertainties, and calculational statistics), MCNP calculated k_{eff}, standard deviation, and energy of average lethargy causing fission (EALF) for each of the candidate fuel types and basket configurations.

Table 6.C.1
CALCULATIONAL SUMMARY FOR ALL CANDIDATE FUEL TYPES AND BASKET CONFIGURATIONS

MPC-24					
Fuel Assembly Designation	**Cask**	**Maximum k_{eff}**	**Calculated k_{eff}**	**Std. Dev. (1-sigma)**	**EALF (eV)**
14x14A01	HI-STAR	0.9295	0.9252	0.0008	0.2084
14x14A02	HI-STAR	0.9286	0.9242	0.0008	0.2096
14x14A03	HI-STORM	0.3080	0.3047	0.0003	3.37E+04
14x14A03	HI-TRAC	0.9283	0.9239	0.0008	0.2096
14x14A03	HI-STAR	0.9296	0.9253	0.0008	0.2093
14x14B01	HI-STAR	0.9159	0.9117	0.0007	0.2727
14x14B02	HI-STAR	0.9169	0.9126	0.0008	0.2345
14x14B03	HI-STAR	0.9110	0.9065	0.0009	0.2545
14x14B04	HI-STAR	0.9084	0.9039	0.0009	0.2563
B14x14B01	HI-TRAC	0.9237	0.9193	0.0008	0.2669
B14x14B01	HI-STAR	0.9228	0.9185	0.0008	0.2675
14x14C01	HI-TRAC	0.9273	0.9230	0.0008	0.2758
14x14C01	HI-STAR	0.9258	0.9215	0.0008	0.2729
14x14C02	HI-STAR	0.9265	0.9222	0.0008	0.2765
14x14C03	HI-TRAC	0.9274	0.9231	0.0008	0.2839
14x14C03	HI-STAR	0.9287	0.9242	0.0009	0.2825

Table 6.C.1 (continued)
CALCULATIONAL SUMMARY FOR ALL CANDIDATE FUEL TYPES
AND BASKET CONFIGURATIONS

MPC-24					
Fuel Assembly Designation	Cask	Maximum k_{eff}	Calculated k_{eff}	Std. Dev. (1-sigma)	EALF (eV)
14x14D01	HI-TRAC	0.8531	0.8488	0.0008	0.3316
14x14D01	HI-STAR	0.8507	0.8464	0.0008	0.3308
14x14E01	HI-STAR	0.7598	0.7555	0.0008	0.3890
14x14E02	HI-TRAC	0.7627	0.7586	0.0007	0.3591
14x14E02	HI-STAR	0.7627	0.7586	0.0007	0.3607
14x14E03	HI-STAR	0.6952	0.6909	0.0008	0.2905
15x15A01	HI-TRAC	0.9205	0.9162	0.0008	0.2595
15x15A01	HI-STAR	0.9204	0.9159	0.0009	0.2608
15x15B01	HI-STAR	0.9369	0.9326	0.0008	0.2632
15C15B02	HI-STAR	0.9338	0.9295	0.0008	0.2640
15x15B03	HI-STAR	0.9362	0.9318	0.0008	0.2632
15x15B04	HI-STAR	0.9370	0.9327	0.0008	0.2612
15x15B05	HI-STAR	0.9356	0.9313	0.0008	0.2606
15x15B06	HI-STAR	0.9366	0.9324	0.0007	0.2638
B15x15B01	HI-TRAC	0.9387	0.9344	0.0008	0.2616
B15x15B01	HI-STAR	0.9388	0.9343	0.0009	0.2626
15x15C01	HI-STAR	0.9255	0.9213	0.0007	0.2493
15x15C02	HI-STAR	0.9297	0.9255	0.0007	0.2457
15x15C03	HI-STAR	0.9297	0.9255	0.0007	0.2440
15x15C04	HI-STAR	0.9311	0.9268	0.0008	0.2435
B15x15C01	HI-TRAC	0.9362	0.9319	0.0008	0.2374
B15x15C01	HI-STAR	0.9361	0.9316	0.0009	0.2385

Table 6.C.1 (continued)
CALCULATIONAL SUMMARY FOR ALL CANDIDATE FUEL TYPES
AND BASKET CONFIGURATIONS

MPC-24					
Fuel Assembly Designation	**Cask**	**Maximum k_{eff}**	**Calculated k_{eff}**	**Std. Dev. (1-sigma)**	**EALF (eV)**
15x15D01	HI-STAR	0.9341	0.9298	0.0008	0.2822
15x15D02	HI-STAR	0.9367	0.9324	0.0008	0.2802
15x15D03	HI-STAR	0.9354	0.9311	0.0008	0.2844
15x15D04	HI-TRAC	0.9354	0.9309	0.0009	0.2963
15x15D04	HI-STAR	0.9339	0.9292	0.0010	0.2958
15x15E01	HI-TRAC	0.9392	0.9349	0.0008	0.2827
15x15E01	HI-STAR	0.9368	0.9325	0.0008	0.2826
15x15F01	HI-STORM	0.3648	0.3614	0.0003	3.03E+04
15x15F01	HI-TRAC	0.9393	0.9347	0.0009	0.2925
15x15F01	HI-STAR	0.9395	0.9350	0.0009	0.2903
15x15G01	HI-TRAC	0.8878	0.8836	0.0007	0.3347
15x15G01	HI-STAR	0.8876	0.8833	0.0008	0.3357
15x15H01	HI-TRAC	0.9333	0.9288	0.0009	0.2353
15x15H01	HI-STAR	0.9337	0.9292	0.0009	0.2349
16x16A01	HI-STORM	0.3447	0.3412	0.0004	3.15E+04
16x16A01	HI-TRAC	0.9273	0.9228	0.0009	0.2710
16x16A01	HI-STAR	0.9287	0.9244	0.0008	0.2704
16x16A02	HI-STAR	0.9263	0.9221	0.0007	0.2702
~~17x17A01~~	~~HI-STAR~~	~~0.9368~~	~~0.9325~~	~~0.0008~~	~~0.2131~~
17x17A0~~2~~*1*	HI-STORM	0.3243	0.3210	0.0003	3.23E+04
17x17A0~~2~~*1*	HI-TRAC	0.9378	0.9335	0.0008	0.2133
17x17A0~~2~~*1*	HI-STAR	0.9368	0.9325	0.0008	0.2131

Table 6.C.1 (continued)
CALCULATIONAL SUMMARY FOR ALL CANDIDATE FUEL TYPES
AND BASKET CONFIGURATIONS

MPC-24					
Fuel Assembly Designation	**Cask**	**Maximum k_{eff}**	**Calculated k_{eff}**	**Std. Dev. (1-sigma)**	**EALF (eV)**
17x17A0~~3~~2	HI-STAR	0.9329	0.9286	0.0008	0.2018
17x17B01	HI-STAR	0.9288	0.9243	0.0009	0.2607
17x17B02	HI-STAR	0.9290	0.9247	0.0008	0.2596
17x17B03	HI-STAR	0.9243	0.9199	0.0008	0.2625
17x17B04	HI-STAR	0.9324	0.9279	0.0009	0.2576
17x17B05	HI-STAR	0.9266	0.9222	0.0008	0.2539
17x17B06	HI-TRAC	0.9318	0.9275	0.0008	0.2570
17x17B06	HI-STAR	0.9311	0.9268	0.0008	0.2593
17x17C01	HI-STAR	0.9293	0.9250	0.0008	0.2595
17x17C02	HI-TRAC	0.9319	0.9274	0.0009	0.2610
17x17C02	HI-STAR	0.9336	0.9293	0.0008	0.2624

Table 6.C.1 (continued)
CALCULATIONAL SUMMARY FOR ALL CANDIDATE FUEL TYPES
AND BASKET CONFIGURATIONS

MPC-68					
Fuel Assembly Designation	**Cask**	**Maximum k_{eff}**	**Calculated k_{eff}**	**Std. Dev. (1-sigma)**	**EALF (eV)**
6x6A01	HI-STAR	0.7539	0.7498	0.0007	0.2754
6x6A02	HI-STAR	0.7517	0.7476	0.0007	0.2510
6x6A03	HI-STAR	0.7545	0.7501	0.0008	0.2494
6x6A04	HI-STAR	0.7537	0.7494	0.0008	0.2494
6x6A05	HI-STAR	0.7555	0.7512	0.0008	0.2470
6x6A06	HI-STAR	0.7618	0.7576	0.0008	0.2298
6x6A07	HI-STAR	0.7588	0.7550	0.0005	0.2360
6x6A08	HI-STAR	0.7808	0.7766	0.0007	0.2527
B6x6A01	HI-TRAC	0.7732	0.7691	0.0007	0.2458
B6x6A01	HI-STAR	0.7727	0.7685	0.0007	0.2460
B6x6A02	HI-TRAC	0.7785	0.7741	0.0008	0.2411
B6x6A02	HI-STAR	0.7782	0.7738	0.0008	0.2408
B6x6A03	HI-TRAC	0.7886	0.7846	0.0007	0.2311
B6x6A03	HI-STAR	0.7888	0.7846	0.0007	0.2310
6x6B01	HI-STAR	0.7604	0.7563	0.0007	0.2461
6x6B02	HI-STAR	0.7618	0.7577	0.0007	0.2450
6x6B03	HI-STAR	0.7619	0.7578	0.0007	0.2439
6x6B04	HI-STAR	0.7686	0.7644	0.0008	0.2286
6x6B05	HI-STAR	0.7824	0.7785	0.0006	0.2184
B6x6B01	HI-TRAC	0.7833	0.7794	0.0006	0.2181
B6x6B01	HI-STAR	0.7822	0.7783	0.0006	0.2190

Table 6.C.1 (continued)
CALCULATIONAL SUMMARY FOR ALL CANDIDATE FUEL TYPES
AND BASKET CONFIGURATIONS

MPC-68					
Fuel Assembly Designation	**Cask**	**Maximum k_{eff}**	**Calculated k_{eff}**	**Std. Dev. (1-sigma)**	**EALF (eV)**
6x6C01	HI-STORM	0.2759	0.2726	0.0003	1.59E+04
6x6C01	HI-TRAC	0.8024	0.7982	0.0008	0.2135
6x6C01	HI-STAR	0.8021	0.7980	0.0007	0.2139
7x7A01	HI-TRAC	0.7963	0.7922	0.0007	0.2016
7x7A01	HI-STAR	0.7974	0.7932	0.0008	0.2015
7x7B01	HI-STAR	0.9372	0.9330	0.0007	0.3658
7x7B02	HI-STAR	0.9301	0.9260	0.0007	0.3524
7x7B03	HI-STAR	0.9313	0.9271	0.0008	0.3438
7x7B04	HI-STAR	0.9311	0.9270	0.0007	0.3816
7x7B05	HI-STAR	0.9350	0.9306	0.0008	0.3382
7x7B06	HI-STAR	0.9298	0.9260	0.0006	0.3957
B7x7B01	HI-TRAC	0.9367	0.9324	0.0008	0.3899
B7x7B01	HI-STAR	0.9375	0.9332	0.0008	0.3887
B7x7B02	HI-STORM	0.4061	0.4027	0.0003	2.069E+04
B7x7B02	HI-TRAC	0.9385	0.9342	0.0008	0.3952
B7x7B02	HI-STAR	0.9386	0.9344	0.0007	0.3983
8x8A01	HI-TRAC	0.7662	0.7620	0.0008	0.2250
8x8A01	HI-STAR	0.7685	0.7644	0.0007	0.2227
8x8A02	HI-TRAC	0.7690	0.7650	0.0007	0.2163
8x8A02	HI-STAR	0.7697	0.7656	0.0007	0.2158
8x8B01	HI-STAR	0.9310	0.9265	0.0009	0.2935
8x8B02	HI-STAR	0.9227	0.9185	0.0007	0.2993

Table 6.C.1 (continued)
CALCULATIONAL SUMMARY FOR ALL CANDIDATE FUEL TYPES
AND BASKET CONFIGURATIONS

MPC-68					
Fuel Assembly Designation	Cask	Maximum k_{eff}	Calculated k_{eff}	Std. Dev. (1-sigma)	EALF (eV)
8x8B03	HI-STAR	0.9299	0.9257	0.0008	0.3319
8x8B04	HI-STAR	0.9236	0.9194	0.0008	0.3700
B8x8B01	HI-TRAC	0.9352	0.9310	0.0008	0.3393
B8x8B01	HI-STAR	0.9346	0.9301	0.0009	0.3389
B8x8B02	HI-TRAC	0.9401	0.9359	0.0007	0.3331
B8x8B02	HI-STAR	0.9385	0.9343	0.0008	0.3329
B8x8B03	HI-STORM	0.3934	0.3900	0.0004	1.815E+04
B8x8B03	HI-TRAC	0.9427	0.9385	0.0008	0.3278
B8x8B03	HI-STAR	0.9416	0.9375	0.0007	0.3293
8x8C01	HI-STAR	0.9315	0.9273	0.0007	0.2822
8x8C02	HI-STAR	0.9313	0.9268	0.0009	0.2716
8x8C03	HI-STAR	0.9329	0.9286	0.0008	0.2877
8x8C04	HI-STAR	0.9348	0.9307	0.0007	0.2915
8x8C05	HI-STAR	0.9353	0.9312	0.0007	0.2971
8x8C06	HI-STAR	0.9353	0.9312	0.0007	0.2944
8x8C07	HI-STAR	0.9314	0.9273	0.0007	0.2972
8x8C08	HI-STAR	0.9339	0.9298	0.0007	0.2915
8x8C09	HI-STAR	0.9301	0.9260	0.0007	0.3183
8x8C10	HI-STAR	0.9317	0.9275	0.0008	0.3018
8x8C11	HI-STAR	0.9328	0.9287	0.0007	0.3001
8x8C12	HI-STAR	0.9285	0.9242	0.0008	0.3062
B8x8C01	HI-TRAC	0.9348	0.9305	0.0008	0.3114

Table 6.C.1 (continued)
CALCULATIONAL SUMMARY FOR ALL CANDIDATE FUEL TYPES
AND BASKET CONFIGURATIONS

MPC-68					
Fuel Assembly Designation	**Cask**	**Maximum k_{eff}**	**Calculated k_{eff}**	**Std. Dev. (1-sigma)**	**EALF (eV)**
B8x8C01	HI-STAR	0.9357	0.9313	0.0009	0.3141
B8x8C02	HI-STORM	0.3714	0.3679	0.0004	2.30E+04
B8x8C02	HI-TRAC	0.9402	0.9360	0.0008	0.3072
B8x8C02	HI-STAR	0.9425	0.9384	0.0007	0.3081
B8x8C03	HI-TRAC	0.9429	0.9386	0.0008	0.3045
B8x8C03	HI-STAR	0.9418	0.9375	0.0008	0.3056
8x8D01	HI-STAR	0.9342	0.9302	0.0006	0.2733
8x8D02	HI-STAR	0.9325	0.9284	0.0007	0.2750
8x8D03	HI-STAR	0.9351	0.9309	0.0008	0.2731
8x8D04	HI-STAR	0.9338	0.9296	0.0007	0.2727
8x8D05	HI-STAR	0.9339	0.9294	0.0009	0.2700
8x8D06	HI-STAR	0.9365	0.9324	0.0007	0.2777
8x8D07	HI-STAR	0.9341	0.9297	0.0009	0.2694
8x8D08	HI-STAR	0.9376	0.9332	0.0009	0.2841
B8x8D01	HI-TRAC	0.9408	0.9368	0.0006	0.2773
B8x8D01	HI-STAR	0.9403	0.9363	0.0007	0.2778
8x8E01	HI-TRAC	0.9309	0.9266	0.0008	0.2834
8x8E01	HI-STAR	0.9312	0.9270	0.0008	0.2831
8x8F01	HI-TRAC	0.9396	0.9356	0.0006	0.2255
8x8F01	HI-STAR	0.9411	0.9366	0.0009	0.2264
9x9A01	HI-STAR	0.9353	0.9310	0.0008	0.2875
9x9A02	HI-STAR	0.9388	0.9345	0.0008	0.2228

Table 6.C.1 (continued)
CALCULATIONAL SUMMARY FOR ALL CANDIDATE FUEL TYPES
AND BASKET CONFIGURATIONS

MPC-68					
Fuel Assembly Designation	Cask	Maximum k_{eff}	Calculated k_{eff}	Std. Dev. (1-sigma)	EALF (eV)
9x9A03	HI-STAR	0.9351	0.9310	0.0007	0.2837
9x9A04	HI-STAR	0.9396	0.9355	0.0007	0.2262
B9x9A01	HI-STORM	0.3365	0.3331	0.0003	1.78E+04
B9x9A01	HI-TRAC	0.9434	0.9392	0.0007	0.2232
B9x9A01	HI-STAR	0.9417	0.9374	0.0008	0.2236
9x9B01	HI-STAR	0.9380	0.9336	0.0008	0.2576
9x9B02	HI-STAR	0.9373	0.9329	0.0009	0.2578
9x9B03	HI-STAR	0.9417	0.9374	0.0008	0.2545
B9x9B01	HI-TRAC	0.9417	0.9376	0.0007	0.2504
B9x9B01	HI-STAR	0.9436	0.9394	0.0008	0.2506
9x9C01	HI-TRAC	0.9377	0.9335	0.0008	0.2697
9x9C01	HI-STAR	0.9395	0.9352	0.0008	0.2698
9x9D01	HI-TRAC	0.9387	0.9343	0.0008	0.2635
9x9D01	HI-STAR	0.9394	0.9350	0.0009	0.2625
9x9E01	HI-STAR	0.9334	0.9293	0.0007	0.2227
9x9E02	HI-STORM	0.3676	0.3642	0.0003	2.409E+04
9x9E02	HI-TRAC	0.9402	0.9360	0.0008	0.2075
9x9E02	HI-STAR	0.9401	0.9359	0.0008	0.2065
9x9F01	HI-STAR	0.9307	0.9265	0.0007	0.2899
9x9F02	HI-STORM	0.3676	0.3642	0.0003	2.409E+04
9x9F02	HI-TRAC	0.9402	0.9360	0.0008	0.2075
9x9F02	HI-STAR	0.9401	0.9359	0.0008	0.2065

Table 6.C.1 (continued)
CALCULATIONAL SUMMARY FOR ALL CANDIDATE FUEL TYPES
AND BASKET CONFIGURATIONS

MPC-68					
Fuel Assembly Designation	**Cask**	**Maximum k_{eff}**	**Calculated k_{eff}**	**Std. Dev. (1-sigma)**	**EALF (eV)**
9x9G01	HI-TRAC	0.9307	0.9265	0.0007	0.2193
9x9G01	HI-STAR	0.9309	0.9265	0.0008	0.2191
10x10A01	HI-STAR	0.9377	0.9335	0.0008	0.3170
10x10A02	HI-STAR	0.9426	0.9386	0.0007	0.2159
10x10A03	HI-STAR	0.9396	0.9356	0.0007	0.3169
B10x10A01	HI-STORM	0.3379	0.3345	0.0003	1.74E+04
B10x10A01	HI-TRAC	0.9448	0.9405	0.0008	0.2214
B10x10A01	HI-STAR	0.9457	0.9414	0.0008	0.2212
10x10B01	HI-STAR	0.9384	0.9341	0.0008	0.2881
10x10B02	HI-STAR	0.9416	0.9373	0.0008	0.2333
10x10B03	HI-STAR	0.9375	0.9334	0.0007	0.2856
B10x10B01	HI-TRAC	0.9443	0.9401	0.0007	0.2380
B10x10B01	HI-STAR	0.9436	0.9395	0.0007	0.2366
10x10C01	HI-TRAC	0.9430	0.9387	0.0008	0.2424
10x10C01	HI-STAR	0.9433	0.9392	0.0007	0.2416
10x10D01	HI-TRAC	0.9383	0.9343	0.0007	0.3359
10x10D01	HI-STAR	0.9376	0.9333	0.0008	0.3355
10x10E01	HI-TRAC	0.9157	0.9116	0.0007	0.3301
10x10E01	HI-STAR	0.9185	0.9144	0.0007	0.2936

Table 6.C.1 (continued)
CALCULATIONAL SUMMARY FOR ALL CANDIDATE FUEL TYPES
AND BASKET CONFIGURATIONS

MPC-24 400PPM SOLUBLE BORON					
Fuel Assembly Designation	Cask	Maximum k_{eff}	Calculated k_{eff}	Std. Dev. (1-sigma)	EALF (eV)
14x14A03	HI-STAR	0.8884	0.8841	0.0008	0.2501
B14x14B01	HI-STAR	0.8900	0.8855	0.0009	0.3173
14x14C03	HI-STAR	0.8950	0.8907	0.0008	0.3410
14x14D01	HI-STAR	0.8518	0.8475	0.0008	0.4395
14x14E02	HI-STAR	0.7132	0.7090	0.0007	0.4377
15x15A01	HI-STAR	0.9119	0.9076	0.0008	0.3363
B15x15B01	HI-STAR	0.9284	0.9241	0.0008	0.3398
B15x15C01	HI-STAR	0.9236	0.9193	0.0008	0.3074
15x15D04	HI-STAR	0.9261	0.9218	0.0008	0.3841
15x15E01	HI-STAR	0.9265	0.9221	0.0008	0.3656
15x15F01	HI-STORM (DRY)	0.4013	0.3978	0.0004	28685
15x15F01	HI-TRAC	0.9301	0.9256	0.0009	0.3790
15x15F01	HI-STAR	0.9314	0.9271	0.0008	0.3791
15x15G01	HI-STAR	0.8939	0.8897	0.0007	0.4392
15x15H01	HI-TRAC	0.9345	0.9301	0.0008	0.3183
15x15H01	HI-STAR	0.9366	0.9320	0.0009	0.3175
16x16A01	HI-STAR	0.8955	0.8912	0.0008	0.3227
17x17A0~~2~~*1*	HI-STAR	0.9264	0.9221	0.0008	0.2801
17x17B06	HI-STAR	0.9284	0.9241	0.0008	0.3383
17x17C02	HI-TRAC	0.9296	0.9250	0.0009	0.3447
17x17C02	HI-STAR	0.9294	0.9249	0.0009	0.3433

Table 6.C.1 (continued)
CALCULATIONAL SUMMARY FOR ALL CANDIDATE FUEL TYPES
AND BASKET CONFIGURATIONS

MPC-24E/MPC-24EF, UNBORATED WATER					
Fuel Assembly Designation	**Cask**	**Maximum k_{eff}**	**Calculated k_{eff}**	**Std. Dev. (1-sigma)**	**EALF (eV)**
14x14A03	HI-STAR	0.9380	0.9337	0.0008	0.2277
B14x14B01	HI-STAR	0.9312	0.9269	0.0008	0.2927
14x14C01	HI-STAR	0.9356	0.9311	0.0009	0.3161
14x14D01	HI-STAR	0.8875	0.8830	0.0009	0.4026
14x14E02	HI-STAR	0.7651	0.7610	0.0007	0.3645
15x15A01	HI-STAR	0.9336	0.9292	0.0008	0.2879
B15x15B01	HI-STAR	0.9465	0.9421	0.0008	0.2924
B15x15C01	HI-STAR	0.9462	0.9419	0.0008	0.2631
15x15D04	HI-STAR	0.9440	0.9395	0.0009	0.3316
15x15E01	HI-STAR	0.9455	0.9411	0.0009	0.3178
15x15F01	HI-STORM (DRY)	0.3699	0.3665	0.0004	3.280e+04
15x15F01	HI-TRAC	0.9465	0.9421	0.0009	0.3297
15x15F01	HI-STAR	0.9468	0.9424	0.0008	0.3270
15x15G01	HI-STAR	0.9054	0.9012	0.0007	0.3781
15x15H01	HI-STAR	0.9423	0.9381	0.0008	0.2628
16x16A01	HI-STAR	0.9341	0.9297	0.0009	0.3019
17x17A0~~2~~*1*	HI-TRAC	0.9467	0.9425	0.0008	0.2372
17x17A0~~2~~*1*	HI-STAR	0.9447	0.9406	0.0007	0.2374
17x17B06	HI-STAR	0.9421	0.9377	0.0008	0.2888
17x17C02	HI-STAR	0.9433	0.9390	0.0008	0.2932

Table 6.C.1 (continued)
CALCULATIONAL SUMMARY FOR ALL CANDIDATE FUEL TYPES
AND BASKET CONFIGURATIONS

MPC-24E/MPC-24EF, 300PPM BORATED WATER					
Fuel Assembly Designation	**Cask**	**Maximum k_{eff}**	**Calculated k_{eff}**	**Std. Dev. (1-sigma)**	**EALF (eV)**
14x14A03	HI-STAR	0.8963	0.8921	0.0008	0.2231
B14x14B01	HI-STAR	0.8974	0.8931	0.0008	0.3214
14x14C01	HI-STAR	0.9031	0.8988	0.0008	0.3445
14x14D01	HI-STAR	0.8588	0.8546	0.0007	0.4407
14x14E02	HI-STAR	0.7249	0.7205	0.0008	0.4186
15x15A01	HI-STAR	0.9161	0.9118	0.0008	0.3408
B15x15B01	HI-STAR	0.9321	0.9278	0.0008	0.3447
B15x15C01	HI-STAR	0.9271	0.9227	0.0008	0.3121
15x15D04	HI-STAR	0.9290	0.9246	0.0009	0.3950
15x15E01	HI-STAR	0.9309	0.9265	0.0009	0.3754
15x15F01	HI-STORM (DRY)	0.3897	0.3863	0.0003	3.192E+04
15x15F01	HI-TRAC	0.9333	0.9290	0.0008	0.3900
15x15F01	HI-STAR	0.9332	0.9289	0.0008	0.3861
15x15G01	HI-STAR	0.8972	0.8930	0.0007	0.4473
15x15H01	HI-TRAC	0.9399	0.9356	0.0008	0.3235
15x15H01	HI-STAR	0.9399	0.9357	0.0008	0.3248
16x16A01	HI-STAR	0.9021	0.8977	0.0009	0.3274
17x17A0~~2~~*1*	HI-STAR	0.9332	0.9287	0.0009	0.2821
17x17B06	HI-STAR	0.9316	0.9273	0.0008	0.3455
17x17C02	HI-TRAC	0.9320	0.9277	0.0008	0.2819
17x17C02	HI-STAR	0.9312	0.9270	0.0007	0.3530

Table 6.C.1 (continued)
CALCULATIONAL SUMMARY FOR ALL CANDIDATE FUEL TYPES
AND BASKET CONFIGURATIONS

MPC-32, ~~1900 PPM BORATED WATER~~*4.1% Enrichment, Bounding Cases*					
Fuel Assembly Designation	**Cask**	**Maximum k_{eff}**	**Calculated k_{eff}**	**Std. Dev. (1-sigma)**	**EALF (eV)**
14x14A03	HI-STAR	*0.9041*	*0.9001*	*0.0006*	*0.3185*
B14x14B01	HI-STAR	*0.9257*	*0.9216*	*0.0007*	*0.4049*
14x14C01	HI-STAR	*0.9423*	*0.9382*	*0.0007*	*0.4862*
14x14D01	HI-STAR	*0.8970*	*0.8931*	*0.0006*	*0.5474*
14x14E02	HI-STAR	*0.7340*	*0.7300*	*0.0006*	*0.6817*
15x15A01	HI-STAR	*0.9206*	*0.9167*	*0.0006*	*0.5072*
B15x15B01	HI-STAR	*0.9397*	*0.9358*	*0.0006*	*0.4566*
B15x15C01	HI-STAR	*0.9266*	*0.9227*	*0.0006*	*0.4167*
15x15D04	HI-STAR	*0.9384*	*0.9345*	*0.0006*	*0.5594*
15x15E01	HI-STAR	*0.9365*	*0.9326*	*0.0006*	*0.5403*
15x15F01	HI-STORM (DRY)	0.4691	0.4658	0.0003	1.207E+04
15x15F01	HI-TRAC	0.9403	0.9364	0.0006	0.4938
15x15F01	HI-STAR	0.9411	0.9371	0.0006	0.4923
15x15G01	HI-STAR	*0.9147*	*0.9108*	*0.0006*	*0.5880*
15x15H01	HI-STAR	*0.9276*	*0.9237*	*0.0006*	*0.4710*
16x16A01	HI-STAR	*0.9468*	*0.9427*	*0.0007*	*0.3925*
17x17A0~~2~~*1*	HI-STAR	*0.9111*	*0.9072*	*0.0006*	*0.4055*
17x17B06	HI-STAR	0.9309	0.9269	0.0006	0.4365
17x17C02	HI-TRAC	0.9365	0.9327	0.0006	0.4468
17x17C02	HI-STAR	0.9355	0.9317	0.0006	0.4469

Table 6.C.1 (continued)
CALCULATIONAL SUMMARY FOR ALL CANDIDATE FUEL TYPES
AND BASKET CONFIGURATIONS

~~MPC-32 2600 PPM BORATED WATER, GUIDE TUBES FILLED~~					
~~Fuel Assembly Designation~~	~~Cask~~	~~Maximum k_{eff}~~	~~Calculated k_{eff}~~	~~Std. Dev. (1-sigma)~~	~~EALF (eV)~~
~~14x14A03~~	~~HI-STAR~~	~~0.8362~~	~~0.8324~~	~~0.0006~~	~~0.4651~~
~~B14x14B01~~	~~HI-STAR~~	~~0.8633~~	~~0.8594~~	~~0.0006~~	~~0.5923~~
~~14x14C01~~	~~HI-STAR~~	~~0.8808~~	~~0.8768~~	~~0.0007~~	~~0.6567~~
~~14x14D01~~	~~HI-STAR~~	~~0.8485~~	~~0.8446~~	~~0.0006~~	~~0.7957~~
~~14x14E02~~	~~HI-STAR~~	~~0.6240~~	~~0.6200~~	~~0.0006~~	~~0.9061~~
~~15x15A01~~	~~HI-STAR~~	~~0.9121~~	~~0.9082~~	~~0.0006~~	~~0.6343~~
~~B15x15B01~~	~~HI-STAR~~	~~0.9286~~	~~0.9247~~	~~0.0006~~	~~0.6613~~
~~B15x15C01~~	~~HI-STAR~~	~~0.9150~~	~~0.9110~~	~~0.0007~~	~~0.5997~~
~~15x15D04~~	~~HI-STAR~~	~~0.9419~~	~~0.9379~~	~~0.0006~~	~~0.7572~~
~~15x15E01~~	~~HI-STAR~~	~~0.9415~~	~~0.9376~~	~~0.0006~~	~~0.7194~~
~~15x15F01~~	~~HI-STORM (DRY)~~	~~0.5142~~	~~0.5108~~	~~0.0004~~	~~1.228E+04~~
~~15x15F01~~	~~HI-TRAC~~	~~0.9463~~	~~0.9423~~	~~0.0007~~	~~0.7409~~
~~15x15F01~~	~~HI-STAR~~	~~0.9465~~	~~0.9425~~	~~0.0006~~	~~0.7421~~
~~15x15G01~~	~~HI-STAR~~	~~0.9109~~	~~0.9070~~	~~0.0006~~	~~0.8486~~
~~15x15H01~~	~~HI-STAR~~	~~0.9301~~	~~0.9263~~	~~0.0006~~	~~0.6257~~
~~16x16A01~~	~~HI-STAR~~	~~0.8868~~	~~0.8829~~	~~0.0006~~	~~0.6105~~
~~17x17A02~~	~~HI-STAR~~	~~0.9145~~	~~0.9105~~	~~0.0006~~	~~0.5382~~
~~17x17B06~~	~~HI-STAR~~	~~0.9358~~	~~0.9318~~	~~0.0007~~	~~0.6500~~
~~17x17C02~~	~~HI-TRAC~~	~~0.9424~~	~~0.9385~~	~~0.0006~~	~~0.6659~~
~~17x17C02~~	~~HI-STAR~~	~~0.9431~~	~~0.9391~~	~~0.0006~~	~~0.6628~~

Table 6.C.1 (continued)
CALCULATIONAL SUMMARY FOR ALL CANDIDATE FUEL TYPES
AND BASKET CONFIGURATIONS

MPC-32, ~~2600 PPM BORATED WATER, GUIDE TUBES VOIDED~~*5.0% Enrichment, Bounding Cases*					
Fuel Assembly Designation	**Cask**	**Maximum k_{eff}**	**Calculated k_{eff}**	**Std. Dev. (1-sigma)**	**EALF (eV)**
14x14A03	HI-STAR	*0.9000*	*0.8959*	*0.0007*	*0.4651*
B14x14B01	HI-STAR	*0.9214*	*0.9175*	*0.0006*	*0.6009*
14x14C01	HI-STAR	*0.9480*	*0.9440*	*0.0006*	*0.6431*
14x14D01	HI-STAR	*0.9050*	*0.9009*	*0.0007*	*0.7276*
14x14E02	*HI-STAR*	*0.7415*	*0.7375*	*0.0006*	*0.9226*
15x15A01	HI-STAR	*0.9230*	*0.9189*	*0.0007*	*0.7143*
B15x15B01	HI-STAR	*0.9429*	*0.9390*	*0.0006*	*0.7234*
B15x15C01	HI-STAR	*0.9307*	*0.9268*	*0.0006*	*0.6439*
15x15D04	HI-STAR	0.9466	0.9425	0.0007	0.7525
15x15E01	HI-STAR	0.9434	0.9394	0.0007	0.7215
15x15F01	*HI-STORM (DRY)*	*0.5142*	*0.5108*	*0.0004*	*1.228E+04*
15x15F01	HI-TRAC	0.9470	0.9431	0.0006	0.7456
15x15F01	HI-STAR	0.9483	0.9443	0.0007	0.7426
15x15G01	HI-STAR	*0.9251*	*0.9212*	*0.0006*	*0.9303*
15x15H01	HI-STAR	*0.9333*	*0.9292*	*0.0007*	*0.7015*
16x16A01	HI-STAR	*0.9474*	*0.9434*	*0.0006*	*0.5936*
17x17A0~~2~~*1*	HI-STAR	*0.9161*	*0.9122*	*0.0006*	*0.6141*
17x17B06	HI-STAR	0.9371	0.9331	0.0006	0.6705
17x17C02	HI-TRAC	0.9436	0.9396	0.0006	0.6773
17x17C02	HI-STAR	0.9437	0.9399	0.0006	0.6780

Table 6.C.1 (continued)
CALCULATIONAL SUMMARY FOR ALL CANDIDATE FUEL TYPES
AND BASKET CONFIGURATIONS

Note: Maximum k_{eff} = Calculated k_{eff} + $K_c \times \sigma_c$ + Bias + σ_B
where:

K_c = 2.0
σ_c = Std. Dev. (1-sigma)
Bias = 0.0021
σ_B = 0.0006

See Subsection 6.4.3 for further explanation.

CHAPTER 7†: CONFINEMENT

7.0 INTRODUCTION

Confinement of all radioactive materials in the HI-STORM 100 System is provided by the MPC. The design of the HI-STORM 100 confinement boundary assures that there are no credible design basis events that would result in a radiological release to the environment. The HI-STORM 100 Overpack and HI-TRAC Transfer Cask are designed to provide physical protection for an MPC during normal, off-normal, and postulated accident conditions to assure that the integrity of the MPC confinement boundary is maintained. The inert atmosphere in the MPC and the passive heat removal capabilities of the HI-STORM 100 also assure that the SNF assemblies remain protected from degradation, which might otherwise lead to gross cladding ruptures during dry storage.

A detailed description of the confinement structures, systems, and components important to safety is provided in Chapter 2. The structural adequacy of the MPC is demonstrated by the analyses documented in Chapter 3. The physical protection of the MPC provided by the Overpack and the HI-TRAC Transfer Cask is demonstrated by the structural analyses documented in Chapter 3 and for off-normal and postulated accident conditions in Chapter 11. The heat removal capabilities of the HI-STORM 100 System are demonstrated by the thermal analyses documented in Chapter 4.

This chapter describes the HI-STORM 100 confinement boundary design and describes how the design satisfies the confinement requirements of 10CFR72 [7.0.1]. It also provides an evaluation of *the MPC confinement boundary as it relates to the criteria contained in Interim Staff Guidance (ISG)–18 as justification for determining that leakage from the confinement boundary is not credible and, therefore, no confinement analysis is required.* ~~postulated radiological releases to the environment under normal, off-normal, and accident conditions of storage to ensure compliance with the limits established by the regulations~~.

This chapter is in compliance with NUREG-1536 except as noted in Table 1.0.3.

† This chapter has been prepared in the format and section organization set forth in Regulatory Guide 3.61. However, the material content of this chapter also fulfills the requirements of NUREG-1536. Pagination and numbering of sections, figures, and tables are consistent with the convention set down in *Chapter 1,* Section 1.0, herein. Finally, all terms-of-art used in this chapter are consistent with the terminology of the glossary (Table 1.0.1) and component nomenclature of the Bill-of-Materials (Section 1.5).

7.1 CONFINEMENT BOUNDARY

The primary confinement boundary against the release of radionuclides is the cladding of the individual fuel rods. The spent fuel rods are protected from degradation by maintaining an inert gas atmosphere (helium) inside the MPC and keeping the fuel cladding temperatures below the design basis values specified in Chapter 2.

The HI-STORM 100 confinement boundary consists of any one of the ~~seven~~ fully-welded MPC designs described in Chapter 1. Each MPC is identical from a confinement perspective so the following discussion applies to all MPCs. The confinement boundary of the MPC consists of:

- MPC shell
- bottom baseplate
- MPC lid (including the vent and drain port cover plates)
- MPC closure ring
- associated welds

The above items form a totally seal-welded vessel for the storage of design basis spent fuel assemblies.

The MPC requires no valves, gaskets or mechanical seals for confinement. Figure 7.1.1 shows an elevation cross-section of the MPC confinement boundary. All components of the confinement boundary are Important to Safety, Category A, as specified in Table 2.2.6. The MPC confinement boundary is designed and fabricated in accordance with the ASME Code, Section III, Subsection NB [7.1.1] to the maximum extent practicable. Chapter 2 provides design criteria for the confinement design. Section 2.2.4 provides applicable Code requirements. ~~Exceptions~~ *NRC-approved alternatives* to specific Code requirements with complete justifications are presented in Table 2.2.15.

7.1.1 Confinement Vessel

The HI-STORM 100 *System* confinement vessel is the MPC. The MPC is designed to provide confinement of all radionuclides under normal, off-normal and accident conditions. The MPC is designed, fabricated, *inspected,* and tested in accordance with the applicable requirements of ASME, Section III, Subsection NB [7.1.1] *including certain NRC-approved alternatives.* ~~to the maximum extent practicable.~~ The MPC shell and baseplate assembly and basket structure are delivered to the loading facility as one complete component. The MPC lid, vent and drain port cover plates, and closure ring are supplied separately and are installed following fuel loading. The MPC lid and closure ring are welded to the upper part of the MPC shell ~~at the~~*after fuel* loading ~~site~~ to provide redundant sealing of the confinement boundary. The vent and drain port cover plates are welded to the MPC lid after the lid is welded to the MPC. The welds forming the confinement boundary are described in detail in Section 7.1.3.

The MPC lid is made intentionally thick to minimize radiation exposure to workers during MPC closure operations, and is welded to the MPC shell. The vent and drain port cover plates are welded to the MPC lid following completion of MPC draining, moisture removal, and helium backfill activities to close the MPC vent and drain openings. The MPC lid has a stepped recess around the perimeter for accommodating the closure ring. The MPC closure ring is welded to the MPC lid on the inner diameter of the ring and to the MPC shell on the outer diameter. The combination of the welded MPC lid and closure ring form the redundant closure of the MPC.

Table 7.1.1 provides a summary of the design ratings for normal, off-normal and accident conditions for the MPC confinement vessel. Tables 1.2.2, 2.2.1, and 2.2.3 provide additional design basis information.

~~The design basis leakage rate for the MPC confinement boundary is provided in Table 7.1.1.~~ The MPC shell and baseplate are helium leakage tested during fabrication in accordance with the requirements defined in Chapter 9. Following fuel loading and MPC lid welding, the MPC lid-to-shell weld is examined by liquid penetrant method ~~(root and final)~~, volumetrically examined (*or,* if volumetric examination is not performed, multi-layer liquid penetrant examination must be performed), ~~helium leakage tested,~~ and ~~hydrostatically~~*pressure* tested. If the MPC lid weld is acceptable, the vent and drain port cover plates are welded in place~~,~~ *and* examined by the liquid penetrant method. ~~(root and final), and a leakage rate test is performed.~~ Finally, the MPC closure ring is installed, welded and inspected by the liquid penetrant method. ~~(root, if multiple pass, and final).~~ Chapters 8, 9, and 12 provide procedural guidance, acceptance criteria, and *operating controls*~~Technical Specifications~~, respectively, for performance and acceptance of liquid penetrant examinations, volumetric examination, ~~hydrostatic~~ *and pressure* testing~~, and leakage rate testing~~ of the field welds on the MPC.

After moisture removal, the MPC cavity is backfilled with helium. The helium backfill provides an inert atmosphere within the MPC cavity that precludes oxidation and hydride attack of the SNF cladding. Use of a helium atmosphere within the MPC contributes to the long-term integrity of the fuel cladding, reducing the potential for release of fission gas or other radioactive products to the MPC cavity. Helium also aids in heat transfer within the MPC and reduces the maximum fuel cladding temperatures. MPC inerting, in conjunction with the thermal design features of the MPC and storage cask, assures that the fuel assemblies are sufficiently protected against degradation, which might otherwise lead to gross cladding ruptures during long-term storage.

7.1.2 Confinement Penetrations

The MPC penetrations are designed to prevent the release of radionuclides under all normal, off-normal and accident conditions of storage. Two penetrations (the MPC vent and drain ports) are provided in the MPC lid for MPC draining, moisture removal and backfilling during MPC loading operations, and for fuel cool-down and MPC flooding during unloading operations. No other confinement penetrations exist in the MPC. The MPC vent and drain ports are equipped with metal-to-metal seals to minimize leakage and withstand the long-term effects of temperature and radiation. The vent and drain connectors allow the vent and drain ports to be operated like valves and prevent

the need to hot tap into the penetrations during unloading operations. The MPC vent and drain ports are sealed by cover plates ~~which~~*that* are seal welded to the MPC lid. No credit is taken for the seal provided by the vent and drain port *caps*. The MPC closure ring covers the vent and drain port cover plate welds and the MPC lid-to-shell weld providing the redundant closure of the MPC vessel. The redundant closures of the MPC satisfy the requirements of 10CFR72.236(e) [7.0.1].

The MPC has no bolted closures or mechanical seals. The confinement boundary contains no external penetrations for pressure monitoring or overpressure protection.

7.1.3 Seals and Welds

The MPC is designed, fabricated, and tested in accordance with the applicable requirements of ASME, Section III, Subsection NB [7.1.1], *with certain NRC-approved alternatives.* ~~to the maximum extent practicable.~~ The MPC has no bolted closures or mechanical seals. Section 7.1.1 describes the design of the confinement vessel welds. The welds forming the confinement boundary are summarized in Table 7.1.2.

Confinement boundary welds are performed, inspected, and tested in accordance with the applicable requirements of ASME Section III, Subsection NB [7.1.1] *with certain NRC-approved alternatives.* ~~to the maximum extent practicable.~~ The use of multi-pass welds, root pass, for multiple pass welds, and final surface liquid penetrant inspection, and volumetric examination essentially eliminates the chance of a pinhole leak through the weld. If volumetric examination is not performed, multi-layer liquid penetrant examination must be performed. Welds *other than the field closure welds* are also helium leak tested *in the fabrication shop*, providing added assurance of weld integrity. Additionally, a ~~hydrostatic~~*Code pressure* test is performed on the MPC lid-to-shell weld to confirm the weld's structural integrity *after fuel loading*. The ductile stainless steel material used for the MPC confinement boundary is not susceptible to delamination or hydrogen-induced weld degradation. The closure weld redundancy assures that failure of any single MPC confinement boundary closure weld does not result in release of radioactive material to the environment. Table ~~7.1.3~~*9.1.4* provides a summary of the closure weld examinations and tests.

7.1.4 Closure

The MPC is a totally seal-welded pressure vessel. The MPC has no bolted closure or mechanical seals. The MPC's redundant closures are designed to maintain confinement integrity during normal conditions of storage, and off-normal and postulated accident conditions. There are no unique or special closure devices. Primary closure welds (lid-to-shell and vent/drain port cover plate-to-lid) are examined ~~and leakage tested to~~*using the liquid penetrant technique to* ensure their integrity. A description of the MPC weld examinations is provided in Chapter 9.

Since the MPC uses an entirely welded redundant closure system, no direct monitoring of the closure is required. Section 11.2.1.4 describes requirements for verifying the continued confinement capabilities of the MPC in the event of off-normal or accident conditions. As discussed in Section 2.3.3.2, no instrumentation is required or provided for HI-STORM 100 storage operations, other than normal security service instruments and TLDs.

7.1.5 Damaged Fuel Container

The MPC is designed to allow for the storage of specified damaged fuel assemblies and fuel debris in a specially designed damaged fuel container (DFC). Fuel assemblies classified as damaged fuel or fuel debris as specified in ~~the Approved Contents Section of Appendix B to the CoC~~ *Section 2.1.9* have been evaluated.

To aid in loading and unloading, damaged fuel assemblies and fuel debris will be loaded into stainless steel DFCs *for storage* ~~prior to placement~~ in the HI-STORM 100 System. The DFCs that may be loaded into the MPCs are *discussed in Section 2.1.3.*~~shown in Figures 2.1.1 through Figure 2.1.2c.~~ The DFC is designed to provide SNF loose component retention and handling capabilities. The DFC consists of a smooth-walled, welded stainless steel square container with a removable lid. The container lid provides the means of DFC closure and handling. The DFC is provided with stainless steel wire mesh screens in the top and bottom for draining, moisture removal and helium backfill operations. The screens are specified as a 250-by-250-mesh with an effective opening of 0.0024 inches. There are no other openings in the DFC. *Section 2.1.9*~~The CoC~~ specifies the fuel assembly characteristics for damaged fuel acceptable for loading in the MPC-24E, MPC-24EF, *MPC-32F,* MPC-68, MPC-68F or MPC-68FF and for fuel debris acceptable for loading in the MPC-24EF, *MPC-32F,* MPC-68F or MPC-68FF.

Since the DFC has screens on the top and bottom, the DFC provides no pressure retention function. The confinement function of the DFC is limited to minimizing the release of loose particulates within the sealed MPC. The storage design basis leakage rates are not altered by the presence of the DFCs. The radioactive material available for release from the specified fuel assemblies are bounded by the design basis fuel assemblies analyzed herein.

7.1.6 Design and Qualification of Final MPC Closure Welds

The Holtec MPC final closure welds meet the criteria of NRC Interim Staff Guidance (ISG)18 [7.1.2] such that leakage from the MPC confinement boundary is not considered credible. Table 7.1.4 provides the matrix of ISG-18 criteria and how the Holtec MPC design and associated inspection, testing, and QA requirements meet each one. In addition, because proper execution of the MPC lid-to-shell weld is vital to ensuring leak-tightness of each field-welded MPC, Holtec shall review the closure welding procedures for conformance to Code and FSAR requirements.

Table 7.1.1

SUMMARY OF CONFINEMENT BOUNDARY DESIGN SPECIFICATIONS

Design Condition	Design Pressure (psig)	Design Temperature (°F)
Normal	100	MPC Lid: 550
		MPC Shell: 450
		MPC Baseplate: 400
Off-Normal	*110*~~100~~	MPC Lid: 775
		MPC Shell: 775
		MPC Baseplate: 775
Accident	200	MPC Lid: 775
		MPC Shell: 775
		MPC Baseplate: 775

Table 7.1.2

MPC CONFINEMENT BOUNDARY WELDS

Confinement Boundary Welds		
MPC Weld Location	**Weld Type†**	**ASME Code Category (Section III, Subsection NB)**
Shell longitudinal seam	Full Penetration Groove (shop weld)	A
Shell circumferential seam	Full Penetration Groove (shop weld)	B
Baseplate to shell	Full Penetration Groove (shop weld)	C
MPC lid to shell	Partial Penetration Groove (field weld)	C
MPC closure ring to shell	Fillet (field weld)	††
Vent and drain port cover plates to MPC lid	Partial Penetration Groove (field weld)	D
MPC closure ring to closure ring radial	Partial Penetration Groove (field weld)	††
MPC closure ring to MPC lid	Partial Penetration Groove (field weld)	C

† The tests and inspections for the confinement boundary welds are listed in Section 9.1.1.

†† This joint is governed by NB-5271 (liquid penetrant examination).

Table 7.1.3

~~CLOSURE WELD EXAMINATIONS AND TESTS~~

TABLE DELETED

~~Closure Weld Description~~	~~Inspections/Tests~~	~~ASME Acceptance Criteria~~
~~MPC Lid to Shell~~	~~VT on Tack Welds~~ ~~PT Root Pass~~ ~~PT Final Pass~~ ~~VT Final Pass~~ ~~Volumetric Examination of Weld (UT) or multi-layer PT~~ ~~Hydrostatic Test~~ ~~Post Hydrostatic Test – PT~~ ~~Helium Leakage Test~~	~~NF-5360~~ ~~NB-5350~~ ~~NB-5350~~ ~~NF-5360~~ ~~NB-5332~~ ~~NB-6000~~ ~~NB-5350~~ ~~Sect. V and ANSI N14.5~~
~~Vent/Drain Cover Plate to MPC Lid~~	~~VT on Tack Welds~~ ~~PT Root Pass~~ ~~PT Final Pass~~ ~~VT Final Pass~~ ~~Helium Leakage Test~~	~~NF-5360~~ ~~NB-5350~~ ~~NB-5350~~ ~~NF-5360~~ ~~Sect. V and ANSI N14.5~~
~~Closure Ring Radial Welds~~	~~VT on Tack Welds~~ ~~PT Root Pass~~ ~~(if multiple pass)~~ ~~PT Final Pass~~ ~~VT Final Pass~~	~~NF-5360~~ ~~NB-5350~~ ~~NB-5350~~ ~~NF-5360~~
~~Closure Ring to MPC Shell~~	~~VT on Tack Welds~~ ~~PT Root Pass~~ ~~(if multiple pass)~~ ~~PT Final Pass~~ ~~VT Final Pass~~	~~NF-5360~~ ~~NB-5350~~ ~~NB-5350~~ ~~NF-5360~~
~~Closure Ring to MPC Lid~~	~~VT on Tack Welds~~ ~~PT Root Pass~~ ~~PT Final Pass~~ ~~VT Final Pass~~	~~NF-5360~~ ~~NB-5350~~ ~~NB-5350~~ ~~NF-5360~~

Table 7.1.4

COMPARISON OF HOLTEC MPC DESIGN WITH ISG-18 GUIDANCE FOR STORAGE

DESIGN/QUALIFICATION GUIDANCE	*HOLTEC MPC DESIGN*	*FSAR REFERENCE*
The canister is constructed from austenitic stainless steel	*The MPC enclosure vessel is constructed entirely from austenitic stainless steel (Alloy X). Alloy X is defined as Type 304, 304LN, 316, or 316LN material*	*Section 1.2.1.1 and Appendix 1.A*
The canister closure welds meet the guidance of ISG-15 (or approved alternative), Section X.5.2.3	*The MPC lid-to-shell (LTS) closure weld meets ISG-15, Section X.5.2.3 for austenitic stainless steels. UT examination is permitted and NB-5332 acceptance criteria are required. An optional multi-layer PT examination is also permitted. The multi-layer PT is performed at each approximately 3/8" of weld depth, which corresponds to the critical flaw size. A weld quality factor of 0.45 (45% of actual weld capacity) has been used in the stress analysis.*	*Section 9.1.1.1 and Tables 2.2.15 and 9.1.4.* *HI-STAR FSAR Section 3.4.4.3.1.5 and Appendix 3.E (Docket 72-1008)*
The canister maintains its confinement integrity during normal conditions, anticipated occurrences, and credible accidents, and natural phenomena	*The MPC is shown by analysis to maintain confinement integrity for all normal, off-normal, and accident conditions, including natural phenomena. The MPC is design to withstand 45 g deceleration loadings and the cask system is analyzed to verify that decelerations due to credible drops and non-mechanistic tipovers will be less than 45 g's.*	*Section 3.4.4.3 and Appendix 3.A.* *HI-STAR FSAR Section 3.4.4.3*
Records documenting the fabrication and closure welding of canisters shall comply with the provisions 10 CFR 72.174 and ISG-15. Record storage shall comply with ANSI N45.2.9.	*Records documenting the fabrication and closure welding of MPCs meet the requirements of ISG-15 via controls required by the FSAR and HI-STORM CoC. Compliance with 10 CFR 72.174 and ANSI N.45.2.9 is achieved via Holtec QA program and implementing procedures.*	*Section 9.1.1.1 and Table 2.2.15* *Section 13.0*
Activities related to inspection, evaluation, documentation of fabrication, and closure welding of canisters shall be performed in accordance with an NRC-approved quality assurance program.	*The NRC has approved the Holtec quality assurance program under 10 CFR 71. That QA program approval has been adopted for activities governed by 10 CFR 72 as permitted by 10 CFR 72.140(d)*	*Section 13.0*

7.2 REQUIREMENTS FOR NORMAL AND OFF-NORMAL CONDITIONS OF STORAGE

The MPC uses multiple confinement barriers provided by the fuel cladding and the MPC enclosure vessel to assure that there is no release of radioactive material to the environment. Chapter 3 shows that all confinement boundary components are maintained within their Code-allowable stress limits during normal *and off-normal* storage conditions. Chapter 4 shows that the peak confinement boundary component temperatures and pressures are within the design basis limits for all normal *and off-normal* conditions of storage. *Section 7.1 provides a discussion as to how the Holtec MPC design, welding, testing and inspection requirements meet the guidance of ISG-18 such that leakage from the confinement boundary may be considered non-credible.* Since the MPC confinement vessel remains intact, and the design bases temperatures and pressure are not exceeded, *leakage from the MPC confinement boundary is not credible* ~~the design basis leakage rate is not exceeded~~ during normal *and off-normal* conditions of storage.

7.3 CONFINEMENT REQUIREMENTS FOR HYPOTHETICAL ACCIDENT CONDITIONS

The MPC uses redundant confinement closures to assure that there is no release of radioactive materials, including fission gases, volatiles, fuel fines or crud, for postulated storage accident conditions. The analyses presented in Chapters 3 and 11 demonstrate that the MPC remains intact during all ~~normal, off-normal and~~ postulated accident conditions, including the associated increased internal pressure due to decay heat generated by the stored fuel. The MPC is designed, fabricated, and tested in accordance with the applicable requirements of ASME, Section III, Subsection NB [7.1.1], *with certain NRC-approved alternatives as listed in Table 2.2.15.* ~~to the maximum extent practicable.~~ *Section 7.1 provides a discussion as to how the Holtec MPC design, welding, testing and inspection requirements meet the guidance of ISG-18 such that leakage from the confinement boundary may be considered non-credible.* In summary, there is no mechanistic failure that results in a breach of, *and associated leakage of radioactive material from* the MPC confinement boundary.

7.4 REFERENCES

[7.0.1] 10CFR72, Licensing Requirements for the Independent Storage of Spent Nuclear Fuel and High-Level Radioactive Waste.

[7.0.2] NUREG-1536, "Standard Review Plan for Dry Cask Storage Systems", January, 1997.

[7.1.1] American Society of Mechanical Engineers (ASME), Boiler and Pressure Vessel Code, Section III, Division 1, Subsection NB, Class 1 Components, 1995 Edition.

[7.~~12.1~~2] *Interim Staff Guidance 18, "The Design/Qualification of Final Closure Welds on Austenitic Stainless Steel Canisters as Confinement Boundary for Spent Fuel Storage and Containment Boundary for Spent Fuel Transportation," May 2003.* ~~Interim Staff Guidance 11, Revision 1, "Transportation and Storage of Spent Fuel Having Burnups in Excess of 45GWD/MTU", May 16, 2000.~~

[7.2.2] ~~Interim Staff Guidance 5, Revision 1, "Normal, Off Normal, and Hypothetical Dose Estimate Calculations", June 18, 1999~~*Deleted.*

[7.3.1] Deleted.

[7.3.2] ~~Anderson, B.L. et al. *Containment Analysis for Type B Packages Used to Transport Various Contents.* NUREG/CR-6487, UCRL-ID-124822. Lawrence Livermore National Laboratory, November 1996~~*Deleted.*

[7.3.3] ~~Shleien, B, *The Health Physics and Radiological Health Handbook*, Scinta, Inc. Silver Spring, MD, 1992~~*Deleted.*

[7.3.4] ~~U.S. Nuclear Regulatory Commission, "Atmospheric Dispersement Models for Potential Accident Consequence Assessments at Nuclear Power Plants," Regulatory Guide 1.145, February 1989~~*Deleted.*

[7.3.5] ~~U.S. EPA, Federal Guidance Report No. 11, *Limiting Values of Radionuclide Intake and Air Concentration and Dose Conversion Factors for Inhalation, Submersion, and Ingestion*, DE89-011065, 1988~~*Deleted.*

[7.3.6] ~~U.S. EPA, Federal Guidance Report No. 12, *External Exposure to Radionuclides in Air, Water, and Soil*, EPA 402-R-93-081, 1993~~*Deleted.*

[7.3.7] ~~International Commission on Radiological Protection, Limits for Intakes of Radionuclides by Workers, ICRP Publication 30, Part 1; Pergamon Press; Oxford; 1978~~*Deleted.*

[7.3.8] ~~ANSI N14.5-1997. "American National Standard for Radioactive Material Leakage Tests~~

7.4 REFERENCES

[7.0.1] 10CFR72, Licensing Requirements for the Independent Storage of Spent Nuclear Fuel and High-Level Radioactive Waste.

[7.0.2] NUREG-1536, "Standard Review Plan for Dry Cask Storage Systems", January, 1997.

[7.1.1] American Society of Mechanical Engineers (ASME), Boiler and Pressure Vessel Code, Section III, Division 1, Subsection NB, Class 1 Components, 1995 Edition.

[7.*12*.~~1~~2] *Interim Staff Guidance 18, "The Design/Qualification of Final Closure Welds on Austenitic Stainless Steel Canisters as Confinement Boundary for Spent Fuel Storage and Containment Boundary for Spent Fuel Transportation," May 2003.* ~~Interim Staff Guidance 11, Revision 1, "Transportation and Storage of Spent Fuel Having Burnups in Excess of 45GWD/MTU", May 16, 2000.~~

[7.2.2] ~~Interim Staff Guidance 5, Revision 1, "Normal, Off Normal, and Hypothetical Dose Estimate Calculations", June 18, 1999~~*Deleted.*

[7.3.1] Deleted.

[7.3.2] ~~Anderson, B.L. et al. *Containment Analysis for Type B Packages Used to Transport Various Contents.* NUREG/CR-6487, UCRL-ID-124822. Lawrence Livermore National Laboratory, November 1996~~*Deleted.*

[7.3.3] ~~Shleien, B, *The Health Physics and Radiological Health Handbook*, Scinta, Inc. Silver Spring, MD, 1992~~*Deleted.*

[7.3.4] ~~U.S. Nuclear Regulatory Commission, "Atmospheric Dispersement Models for Potential Accident Consequence Assessments at Nuclear Power Plants," Regulatory Guide 1.145, February 1989~~*Deleted.*

[7.3.5] ~~U.S. EPA, Federal Guidance Report No. 11, *Limiting Values of Radionuclide Intake and Air Concentration and Dose Conversion Factors for Inhalation, Submersion, and Ingestion*, DE89-011065, 1988~~*Deleted.*

[7.3.6] ~~U.S. EPA, Federal Guidance Report No. 12, *External Exposure to Radionuclides in Air, Water, and Soil*, EPA 402-R-93-081, 1993~~*Deleted.*

[7.3.7] ~~International Commission on Radiological Protection, Limits for Intakes of Radionuclides by Workers, ICRP Publication 30, Part 1; Pergamon Press; Oxford; 1978~~*Deleted.*

[7.3.8] ~~ANSI N14.5-1997. "American National Standard for Radioactive Material Leakage Tests~~

APPENDIX 7.A

DELETED

~~EXAMPLE DOSE CALCULATIONS FOR NORMAL, OFF-NORMAL, AND ACCIDENT CONDITIONS OF STORAGE~~

~~MPC-32, Normal Conditions of Storage, Dose from Inhalation: 7 pages~~
~~MPC-32, Off-Normal Conditions of Storage, Dose from Inhalation: 7 pages~~
~~MPC-32, Accident Conditions of Storage, Dose from Inhalation: 7 pages~~

~~MPC-32, Normal Conditions of Storage, Dose from Submersion: 8 pages~~
~~MPC-32, Off-Normal Conditions of Storage, Dose from Submersion: 8 pages~~
~~MPC-32, Accident Conditions of Storage, Dose from Submersion: 8 pages~~

~~MPC-68, Normal Conditions of Storage, Dose from Inhalation: 7 pages~~
~~MPC-68, Off-Normal Conditions of Storage, Dose from Inhalation: 7 pages~~
~~MPC-68, Accident Conditions of Storage, Dose from Inhalation: 7 pages~~

~~MPC-68, Normal Conditions of Storage, Dose from Submersion: 8 pages~~
~~MPC-68, Off-Normal Conditions of Storage, Dose from Submersion: 8 pages~~
~~MPC-68, Accident Conditions of Storage, Dose from Submersion: 8 pages~~

CHAPTER 8: OPERATING PROCEDURES†

8.0 INTRODUCTION:

This chapter outlines the loading, unloading, and recovery procedures for the HI-STORM 100 System for storage operations. The procedures provided in this chapter are prescriptive to the extent that they provide the basis and general guidance for plant personnel in preparing detailed, written, site-specific, loading, handling, storage and unloading procedures. Users may add, modify the sequence of, perform in parallel, or delete steps as necessary provided that the intent of this guidance is met and the requirements of the CoC are met. The information provided in this chapter meets all requirements of NUREG-1536 [8.0.1].

Section 8.1 provides the guidance for loading the HI-STORM 100 System in the spent fuel pool. Section 8.2 provides the procedures for ISFSI operations and general guidance for performing maintenance and responding to abnormal events. Responses to abnormal events that may occur during normal loading operations are provided with the procedure steps. Section 8.3 provides the procedure for unloading the HI-STORM 100 System in the spent fuel pool. Section 8.4 provides the guidance for MPC transfer to the HI-STAR 100 Overpack for transport or storage. Section 8.4 can also be used for recovery of a breached MPC for transport or storage. Section 8.5 provides the guidance for transfer of the MPC into HI-STORM from the HI-STAR 100 transport overpack. ~~The Technical Specifications in Appendix A to CoC 72-1014 provide Limiting Conditions of Operation (LCO), Surveillance Requirements (SR's), as well as administrative information, such as Use and Application. Appendix B to COC 72-1014 provides the approved contents and design features applicable to the HI-STORM 100 System. FSAR Appendix 12.A includes the Bases for the LCOs. The Technical Specifications impose restrictions and requirements that must be applied throughout the loading and unloading process.~~ Equipment specific operating details such as Vacuum Drying System, valve manipulation and Transporter operation are not within the scope of this FSAR and will be provided to users based on the specific equipment selected by the users and the configuration of the site.

The procedures contained herein describe acceptable methods for performing HI-STORM 100 loading and unloading operations. Unless otherwise stated, references to the HI-STORM 100 apply equally to the HI-STORM 100 and the HI-STORM 100S. Users may alter these procedures to allow alternate methods and operations to be performed in parallel or out of sequence as long as the general intent of the procedure is met. In the figures following each section, acceptable configurations of rigging, piping, and instrumentation are shown. In some cases, the figures are artists renditions. Users may select alternate configurations, equipment and methodology to accommodate their specific needs provided that the intent of this guidance is met and the requirements of the CoC are met. All rigging should be approved by the user's load handling authority prior to use. User-developed procedures and the design and operation of any alternate equipment must be reviewed by the Certificate holder prior to implementation.

---

† This chapter has been prepared in the format and section organization set forth in Regulatory Guide 3.61. However, the material content of this chapter also fulfills the requirements of NUREG 1536. Pagination and numbering of sections, figures, and tables are consistent with the convention set down in Chapter 1, Section 1.0, herein. Finally, all terms-of-art used in this chapter are consistent with the terminology of the glossary (Table 1.0.1) and component nomenclature of the Bill-of-Materials (Section 1.5).

Licensees (Users) will utilize the procedures provided in this chapter, ~~the Technical Specifications in Appendix A to CoC 72-1014, the conditions of the Certificate of Compliance,~~ equipment-specific operating instructions, and plant working procedures and apply them to develop the site specific written, loading and unloading procedures.

The loading and unloading procedures in Section 8.1 and 8.3 can also be appropriately revised into written site-specific procedures to allow dry loading and unloading of the system in a hot cell or other remote handling facility. The Dry Transfer Facility (DTF) loading and unloading procedures are essentially the same with respect to loading~~, and vacuum drying~~*removing moisture*, inerting, and leakage testing of the MPC. The dry transfer facility shall develop the appropriate site-specific procedures as part of the DTF facility license.

Tables 8.1.1 through 8.1.4 provide the handling weights for each of the HI-STORM 100 System major components and the loads to be lifted during various phases of the operation of the HI-STORM 100 System. Users shall take appropriate actions to ensure that the lift weights do not exceed user-supplied lifting equipment rated loads. Table 8.1.5 provides the HI-STORM 100 System bolt torque and sequencing requirements. Table 8.1.6 provides an operational description of the HI-STORM 100 System ancillary equipment along with its safety designation, where applicable. Fuel assembly selection and verification shall be performed by the licensee in accordance with written, approved procedures which ensure that only SNF assemblies authorized in the Certificate of Compliance and as defined in ~~the Appendix B to CoC 72-1014~~*Section 2.1.9* are loaded into the HI-STORM 100 System.

In addition to the requirements set forth in the CoC, users will be required to develop or modify existing programs and procedures to account for the operation of an ISFSI. Written procedures will be required to be developed or modified to account for such things as nondestructive examination (NDE) of the MPC welds, handling and storage of items and components identified as Important to Safety, 10CFR72.48 [8.1.1] programs, specialized instrument calibration, special nuclear material accountability at the ISFSI, security modifications, fuel handling procedures, training and emergency response, equipment and process qualifications. Users are required to take necessary actions to prevent boiling of the water in the MPC. This may be accomplished by performing a site-specific analysis to identify a time limitation to ensure that water boiling will not occur in the MPC prior to the initiation of draining operations. Chapter 4 of the FSAR provides some sample time limits for the time to initiation of draining for various spent fuel pool water temperatures using design basis heat loads. *Users are also required to take necessary actions to prevent the fuel cladding from exceeding temperature limits during vacuum drying operations and during handling of the MPC in the HI-TRAC transfer cask. Chapter 4.5 of the FSAR provides requirements on the necessary actions, if any, based on the heat load of the MPC.*

Table 8.1.7 summarizes some of the instrumentation used to load and unload the HI-STORM 100 System. ~~Other instrumentation that meets the requirements of the Technical Specifications is also acceptable.~~ Tables 8.1.8, 8.1.9, and 8.1.10 provide sample receipt inspection checklists for the HI-STORM 100 overpack, the MPC, and the HI-TRAC Transfer Cask, respectively. Users may develop site-specific receipt inspection checklists, as required for their equipment. Fuel handling, including the handling of fuel assemblies in the Damaged Fuel Container (DFC) shall

be performed in accordance with written site-specific procedures. DFCs shall be loaded in the spent fuel pool racks prior to placement into the MPC.

Technical and Safety Basis for Loading and Unloading Procedures

The procedures herein (Sections 8.1.2 through 8.1.5) are developed for the loading, storage, unloading, and recovery of spent fuel in the HI-STORM 100 System. The activities involved in loading of spent fuel in a canister system, if not carefully performed, may present risks. The design of the HI-STORM 100 System, including these procedures, the ancillary equipment and the Technical Specifications, serve to minimize risks and mitigate consequences of potential events. To summarize, consideration is given in the loading and unloading systems and procedures to the potential events listed in Table 8.0.1.

The primary objective is to reduce the risk of occurrence and/or to mitigate the consequences of the event. The procedures contain Notes, Warnings, and Cautions to notify the operators to upcoming situations and provide additional information as needed. The Notes, Warnings and Cautions are purposely bolded and boxed and immediately precede the applicable steps.

In the event of an extreme abnormal condition (e.g., cask drop or tip-over event) the user shall have appropriate procedural guidance to respond to the situation. As a minimum, the procedures shall address establishing emergency action levels, implementation of emergency action program, establishment of personnel exclusions zones, monitoring of radiological conditions, actions to mitigate or prevent the release of radioactive materials, and recovery planning and execution and reporting to the appropriate regulatory agencies, as required.

Table 8.0.1
OPERATIONAL CONSIDERATIONS

POTENTIAL EVENTS	METHODS USED TO ADDRESS EVENT	COMMENTS/ REFERENCES
Cask Drop During Handling Operations	Cask lifting and handling equipment is designed to ANSI N14.6. Procedural guidance is given for cask handling, inspection of lifting equipment, and proper engagement to the trunnions. ~~Technical Specifications limit the cask and overpack lift height outside the fuel building.~~	See Section 8.1.2. ~~See Technical Specifications in Appendix A to CoC 72-1014 for HI-TRAC and HI-STORM lift height limitations.~~
Cask Tip-Over Prior to welding of the MPC lid	The Lid Retention System is available to secure the MPC lid during movement between the spent fuel pool and the cask preparation area.	See Section 8.1.5. See Figure 8.1.15.
Contamination of the MPC external shell	The annulus seal, pool lid, and Annulus Overpressure System minimize the potential for the MPC external shell to become contaminated from contact with the spent fuel pool water. ~~Technical Specifications require surveys of certain components of the HI-STORM 100 System to monitor for removable contamination.~~	See Figures 8.1.13 and 8.1.14. ~~See Technical Specifications in Appendix A to CoC 72-1014.~~
Contamination spread from cask process system exhausts	Processing systems are equipped with exhausts that can be directed to the plant's processing systems.	See Figures 8.1.19-8.1.22.
Damage to fuel assembly cladding from oxidation/thermal shock	Fuel assemblies are never subjected to air or oxygen during loading and unloading operations. Cool-Down System brings fuel assembly bulk temperatures to below water boiling temperature prior to flooding.	See Section 8.1.5, *and* Section 8.3.3 ~~and LCO 3.1.3~~.
Damage to Vacuum Drying System vacuum gauges from positive pressure	Vacuum Drying System is separate from pressurized gas and water systems.	See Figure 8.1.22 and 8.1.23.
Ignition of combustible mixtures of gas (e.g., hydrogen) during MPC lid welding or cutting	The area around MPC lid shall be appropriately monitored for combustible gases prior to, and during welding or cutting activities. It is recommended for defense-in-depth that the space below the MPC lid be evacuated or purged prior to, and during these activities.	See Section 8.1.5 and Section 8.3.3.

Table 8.0.1
OPERATIONAL CONSIDERATIONS
(CONTINUED)

POTENTIAL EVENTS	METHODS USED TO ADDRESS EVENT	COMMENTS/ REFERENCES
Excess dose from failed fuel assemblies	MPC gas sampling allows operators to determine the integrity of the fuel cladding prior to opening the MPC. This allows preparation and planning for failed fuel. The RVOAs allow the vent and drain ports to be operated like valves and prevent the need to hot tap into the penetrations during unloading operation.	See Figure 8.1.16 and Section 8.3.3.
Excess dose to operators	The procedures provide ALARA Notes and Warnings when radiological conditions may change.	See ALARA Notes and Warnings throughout the procedures.
Excess generation of radioactive waste	The HI-STORM system uses process systems that minimize the amount of radioactive waste generated. Such features include smooth surfaces for ease of decontamination efforts, prevention of avoidable contamination, and procedural guidance to reduce decontamination requirements. Where possible, items are installed by hand and require no tools.	Examples: HI-TRAC bottom protective cover, bolt plugs in empty holes, pre-wetting of components.
Fuel assembly misloading event	Procedural guidance is given to perform assembly selection verification and a post-loading visual verification of assembly identification prior to installation of the MPC lid.	See Section 8.1.4.
Incomplete moisture removal from MPC	The vacuum drying process reduces the MPC pressure in stages to prevent the formation of ice. Vacuum is held below 3 torr for 30 minutes with the vacuum pump isolated to assure dryness. *If the forced helium dehydration process is used, the temperature of the gas exiting the demoisturizer is held below 21 °F for a minimum of 30 minutes. The TS require the surveillance requirement for moisture removal to be met before entering transport operations.*	See Section 8.1.5, ~~and Technical Specification LCO 3.1.1, and Appendix 12.A, Bases B 3.1.1.~~
Incorrect MPC lid installation	Procedural guidance is given to visually verify correct MPC lid installation prior to HI-TRAC removal from the spent fuel pool.	See Section 8.1.5.

Table 8.0.1
OPERATIONAL CONSIDERATIONS
(CONTINUED)

POTENTIAL EVENTS	METHODS USED TO ADDRESS EVENT	COMMENTS/ REFERENCES
Load Drop	Rigging diagrams and procedural guidance are provided for all lifts. Component weights are provided in Tables 8.1.1 through 8.1.4.	See Figures 8.1.6, 8.1.7, 8.1.9, 8.1.25 and 8.1.27. See Tables 8.1.1 through 8.1.4.
Over-pressurization of MPC during loading and unloading	Pressure relief valves in the water and gas processing systems limit the MPC pressure to acceptable levels.	See Figures 8.1.20, 8.1.21, 8.1.23 and 8.3.4.
Overstressing MPC lift lugs from side loading	The MPC is upended using the upending frame. The lift lugs are never side loaded.	See Figure 8.1.6 and Section 8.1.2.
Overweight cask lift	Procedural guidance is given to alert operators to potential overweight lifts.	See Section 8.1.7 for example. See Tables 8.1.1 through 8.1.4.
Personnel contamination by cutting/grinding activities	Procedural guidance is given to warn operators prior to cutting or grinding activities.	See Section 8.1.5 and Section 8.3.3.
Transfer cask carrying hot particles out of the spent fuel pool	Procedural guidance is given to scan the transfer cask prior to removal from the spent fuel pool.	See Section 8.1.3 and Section 8.1.5.
Unplanned or uncontrolled release of radioactive materials	The MPC vent and drain ports are equipped with metal-to-metal seals to minimize the leakage during *moisture removal*~~vacuum drying~~ and *helium* backfill operations. Unlike elastomer seals, the metal seals resist degradation due to temperature and radiation and allow future access to the MPC ports without hot tapping. The RVOAs allow the port to be opened and closed like a valve so gas sampling may be performed.	See Figure 8.1.11 and 8.1.16. See Section 8.3.3.

8.1 PROCEDURE FOR LOADING THE HI-STORM 100 SYSTEM IN THE SPENT FUEL POOL

8.1.1 Overview of Loading Operations:

The HI-STORM 100 System is used to load, transfer and store spent fuel. Specific steps are performed to prepare the HI-STORM 100 System for fuel loading, to load the fuel, to prepare the system for storage and to place it in storage at an ISFSI. The MPC transfer may be performed in the cask receiving area, at the ISFSI, or any other location deemed appropriate by the user. HI-TRAC and/or HI-STORM may be transferred between the ISFSI and the fuel loading facility using a specially designed transporter, heavy haul transfer trailer, or any other load handling equipment designed for such applications as long as the ~~Technical Specification~~ lift height restrictions are met (lift height restrictions apply only to suspended forms of transport). Users shall develop detailed written procedures to control on-site transport operations. Section 8.1.2 provides the general procedures for rigging and handling of the HI-STORM overpack and HI-TRAC transfer cask. Figure 8.1.1 shows a general flow diagram of the HI-STORM loading operations.

Refer to the boxes of Figure 8.1.2 for the following description. At the start of loading operations, an empty MPC is upended (Box 1). The empty MPC is raised and inserted into HI-TRAC (Box 2). The annulus is filled with plant demineralized water† and the MPC is filled with either spent fuel pool water or plant demineralized water *(borated as required)* (Box 3). An inflatable seal is installed in the upper end of the annulus between the MPC and HI-TRAC to prevent spent fuel pool water from contaminating the exterior surface of the MPC. HI-TRAC and the MPC are then raised and lowered into the spent fuel pool for fuel loading using the lift yoke (Box 4). Pre-selected assemblies are loaded into the MPC and a visual verification of the assembly identification is performed (Box 5).

While still underwater, a thick shielded lid (the MPC lid) is installed using either slings attached to the lift yoke or the optional Lid Retention System (Box 6). The lift yoke remotely engages to the HI-TRAC lifting trunnions to lift the HI-TRAC and loaded MPC close to the spent fuel pool surface (Box 7). When radiation dose rate measurements confirm that it is safe to remove the HI-TRAC from the spent fuel pool, the cask is removed from the spent fuel pool. If the Lid Retention System is being used, the HI-TRAC top lid bolts are installed to secure the MPC lid for the transfer to the cask preparation area. The lift yoke and HI-TRAC are sprayed with demineralized water to help remove contamination as they are removed from the spent fuel pool.

HI-TRAC is placed in the designated preparation area and the Lift Yoke and Lid Retention System (if utilized) are removed. The next phase of decontamination is then performed. The top surfaces of the MPC lid and the upper flange of HI-TRAC are decontaminated. The Temporary Shield Ring (if utilized) is installed and filled with water and the neutron shield jacket is filled with water (if drained). The inflatable annulus seal is removed, and the annulus shield (if utilized) is installed. The Temporary Shield Ring provides additional personnel shielding around the top of the HI-TRAC during MPC closure operations. The annulus shield provides additional personnel shielding at the top of the annulus and also prevents small items from being dropped

† Users may substitute domestic water in each step where demineralized water is specified.

into the annulus. Dose rates are measured at the MPC lid to ensure that the dose rates are within expected values.

The MPC water level is lowered slightly, the MPC is vented, and the MPC lid is seal welded using the automated welding system (Box 8). Visual examinations are performed on the tack welds. Liquid penetrant (PT) examinations are performed on the root and final passes. An ultrasonic or multi-layer PT examination is performed on the MPC Lid-to-Shell weld to ensure that the weld is satisfactory. As an alternative to volumetric examination of the MPC lid-to-shell weld, a multi-layer PT is performed including one intermediate examination after approximately every three-eighth inch of weld depth. The water level is raised to the top of the MPC and a hydrostatic test followed by an additional liquid penetrant examination is performed on the MPC Lid-to-Shell weld to verify structural integrity. ~~A small amount of water is displaced with helium gas for leakage testing. A leakage rate test is performed on the MPC lid-to-shell weld to verify weld integrity and to ensure that leakage rates are within acceptance criteria (See Technical Specification LCO 3.1.1).~~

To calculate the helium backfill requirements for the MPC, the free volume inside the MPC must first be determined. This free volume may be determined by measuring the volume of water displaced or any other suitable means.

Depending upon the heat load of the fuel, moisture is removed from the MPC using either a vacuum drying system or forced helium dehydration system. Section 4.5 of the FSAR has guidance on moisture removal requirements for the various heat loads. For lower heat loads, t~~T~~he vacuum drying system ~~is~~*may be* connected to the MPC and is used to remove all liquid water from the MPC in a stepped evacuation process (Box 9). A stepped evacuation process is used to preclude the formation of ice in the MPC and vacuum drying system lines. The internal pressure is reduced to below 3 torr and held for 30 minutes to ensure that all liquid water is removed ~~(See Technical Specification LCO 3.1.1)~~.

~~Alternatively f~~*F*or high*er* ~~burn-up fuel~~ *heat loads, or as an alternative for lower heat loads*, a *forced helium dehydration*~~moisture removal~~ system is utilized to remove residual moisture from the MPC. Gas is circulated through the MPC to evaporate and remove moisture. The residual moisture is condensed until no additional moisture remains in the MPC. *The temperature of the gas exiting the system demoisturizer is maintained below 21 °F for a minimum of 30 minutes to ensure that all liquid water is removed*~~Gas exiting the MPC is monitored for entrained moisture until no discernable moisture is present in the MPC.~~

Limitations for the at-vacuum duration are evaluated and established on a canister basis to ensure that acceptable cladding temperatures are not exceeded. Refer to FSAR Section 4.5 for requirements on moisture removal based on the heat load in the MPC. Following MPC ~~drying~~*moisture removal*, the MPC is ~~evacuated and~~ backfilled with a predetermined ~~pressure~~ *amount* of helium gas ~~(See Technical Specification LCO 3.1.1)~~. ~~Limitations for the at-vacuum duration are evaluated and established on a canister basis to ensure that acceptable cladding temperatures are not exceeded although a time limit of less than 2 hours at vacuum will bound any MPC.~~ The helium backfill ensures adequate heat transfer during storage, provides an inert atmosphere for long-term fuel integrity, and provides ~~the~~ *a* means ~~of~~ *for* future leakage rate testing of the MPC confinement boundary welds. Cover plates are installed and seal welded over

the MPC vent and drain ports with liquid penetrant examinations performed on the root and final passes (for multi-pass welds) (Box 10). ~~The cover plates are helium leakage tested to confirm that they meet the established leakage rate criteria.~~

The MPC closure ring is then placed on the MPC and dose rates are measured at the MPC lid to ensure that the dose rates are within expected values. The closure ring is aligned, tacked in place and seal welded providing redundant closure of the MPC confinement boundary closure welds. Tack welds are visually examined, and the root and final welds are inspected using the liquid penetrant examination technique to ensure weld integrity.

The annulus shield (if utilized) is removed and the remaining water in the annulus is drained. The Temporary Shield Ring (if utilized) is drained and removed. The MPC lid and accessible areas of the top of the MPC shell are smeared for removable contamination ~~(See Technical Specification LCO 3.2.2)~~ and HI-TRAC dose rates are measured. HI-TRAC top lid[3] is installed and the bolts are torqued (Box 11). The MPC lift cleats are installed on the MPC lid. The MPC lift cleats are the primary lifting point on the MPC. MPC slings are installed between the MPC lift cleats and the lift yoke (Box 12).

If the HI-TRAC 125 is not being used, the transfer lid is attached to the HI-TRAC as follows. The HI-TRAC is positioned above the transfer slide to prepare for bottom lid replacement. The transfer slide consists of an adjustable-height rolling carriage and a pair of channel tracks. The transfer slide supports the transfer step which is used to position the two lids at the same elevation and creates a tight seam between the two lids to eliminate radiation streaming. The overhead crane is shut down to prevent inadvertent operation. The transfer slide carriage is raised to support the pool lid while the bottom lid bolts are removed. The transfer slide then lowers the pool lid and replaces the pool lid with the transfer lid. The carriage is raised and the bottom lid bolts are replaced. The MPC lift cleats and slings support the MPC during the transfer operations. Following the transfer, the MPC slings are disconnected and HI-TRAC is positioned for MPC transfer into HI-STORM.

MPC transfer may be performed inside or outside the fuel building (Box 13). Similarly, HI-TRAC and HI-STORM may be transferred to the ISFSI in several different ways (Box 14 and 15). The empty HI-STORM overpack is inspected and positioned with the lid removed. Vent duct shield inserts[1] are installed in the HI-STORM exit vent ducts. The vent duct shield inserts prevent radiation streaming from the HI-STORM Overpack as the MPC is lowered past the exit vents. If the HI-TRAC 125D is used, the mating device is positioned on top of the HI-STORM. The HI-TRAC is placed on top of HI-STORM. An alignment device (or mating device in the case of HI-TRAC 125D) helps guide HI-TRAC during this operation[2]. The MPC may be lowered using the MPC downloader, the main crane hook or other similar devices. The MPC downloader (if used) may be attached to the HI-TRAC lid or mounted to the overhead lifting device. The MPC slings are attached to the MPC lift cleats.
If the transfer doors are used (i.e. not the HI-TRAC 125D), the MPC is raised slightly, the

---

1 Vent duct shield inserts are only used on the HI-STORM 100.

2 The alignment guide may be configured in many different ways to accommodate the specific sites. See Table 8.1.6.

3 Users with the optional HI-TRAC Lid Spacer shall modify steps in their procedures to install and remove the spacer together with top lid

transfer lid door locking pins are removed and the doors are opened. If the HI-TRAC 125D is used, the pool lid is removed and the mating device drawer is opened. Optional trim plates may be installed on the top and bottom of both doors (or drawer for HI-TRAC 125D) and secured using hand clamps. The trim plates eliminate radiation streaming above and below the doors (drawer). The MPC is lowered into HI-STORM. Following verification that the MPC is fully lowered, the MPC slings are disconnected from the lifting device and lowered onto the MPC lid. The trim plates are removed, the doors (or drawer) are closed. The empty HI-TRAC must be removed with the doors open when the HI-STORM 100S is used to prevent interference with the lift cleats and slings. HI-TRAC is removed from on top of HI-STORM. The MPC slings and MPC lift cleats are removed. Hole plugs are installed in the empty MPC lifting holes to fill the voids left by the lift cleat bolts. The alignment device (or mating device with pool lid for HI-TRAC 125D) and vent duct shield inserts (if used) are removed, and the HI-STORM lid is installed. The exit vent gamma shield cross plates temperature elements (if used) and vent screens are installed.. The HI-STORM lid studs and nuts are installed. The HI-STORM is secured to the transporter (as applicable) and moved to the ISFSI pad. The HI-STORM Overpack and HI-TRAC transfer cask may be moved using a number of methods as long as the lifting equipment requirements ~~in the Technical Specification~~ are met. For sites with high seismic conditions, the HI-STORM 100A is anchored to the ISFSI. Once located at the storage pad, the inlet vent gamma shield cross plates are installed and the shielding effectiveness test is performed. Finally, the temperature elements and their instrument connections are installed (if used), and the air temperature rise testing (if required~~ by the Technical Specifications~~) is performed to ensure that the system is functioning within its design parameters.

8.1.2 HI-TRAC and HI-STORM Receiving and Handling Operations

Note:
HI-TRAC may be received and handled in several different configurations and may be transported on-site in a horizontal or vertical orientation. This section provides general guidance for HI-TRAC and HI-STORM handling. Site-specific procedures shall specify the required operational sequences based on the handling configuration at the sites. ~~Refer to the Technical Specifications for loaded HI-TRAC and HI-STORM 100 Overpack handling limitations.~~

1. Vertical Handling of HI-TRAC:
   a. Verify that the lift yoke load test certifications are current.
   b. Visually inspect the lifting device (lift yoke or lift links) and the lifting trunnions for gouges, cracks, deformation or other indications of damage. Replace or repair damaged components as necessary.
   c. Engage the lift yoke to the lifting trunnions. See Figure 8.1.3.
   d. Apply lifting tension to the lift yoke and verify proper engagement of the lift yoke.

Note:

Refer to the site's heavy load handling procedures for lift height, load path, floor loading and other applicable load handling requirements. ~~Refer to Technical Specification 4.9 for additional equipment handling requirements.~~

Warning:

When lifting the loaded HI-TRAC with only the pool lid, the HI-TRAC should be carried as low as practicable. This minimizes the dose rates due to radiation scattering from the floor. Personnel should remain clear of the area and the HI-TRAC should be placed in position as soon as practicable.

e. Raise HI-TRAC and position it accordingly.

2. Upending of HI-TRAC in the Transfer Frame:

a. Position HI-TRAC under the lifting device. Refer to Step 1, above.

b. If necessary, remove the missile shield from the HI-TRAC Transfer Frame. See Figure 8.1.4.

c. Verify that the lift yoke load test certifications are current.

d. Visually inspect the lift yoke and the lifting trunnions for gouges, cracks, deformation or other indications of damage. Repair or replace damaged components as necessary.

e. Deleted.

f. Engage the lift yoke to the lifting trunnions. See Figure 8.1.3.

g. Apply lifting tension to the lift yoke and verify proper engagement of the lift yoke.

h. Slowly rotate HI-TRAC to the vertical position keeping all rigging as close to vertical as practicable. See Figure 8.1.4.

i. If used, lift the pocket trunnions clear of the Transfer Frame rotation trunnions.

3. Downending of HI-TRAC in the Transfer Frame:

ALARA Warning:

A loaded HI-TRAC should only be downended with the transfer lid or other auxiliary shielding installed.

a. Position the Transfer Frame under the lifting device.

b. Verify that the lift yoke load test certifications are current.

c. Visually inspect the lift yoke and the lifting trunnions for gouges, cracks, deformation or other indications of damage. Repair or replace damaged components as necessary.

d. Deleted.

e. Deleted.

f. Engage the lift yoke to the lifting trunnions. See Figure 8.1.3.

g. Apply lifting tension to the lift yoke and verify proper lift yoke engagement.

h. Position the pocket trunnions to receive the Transfer Frame rotation trunnions. See Figure 8.1.4 (Not used for HI-TRAC 125D).

i. Slowly rotate HI-TRAC to the horizontal position keeping all rigging as close to vertical as practicable.

j. Disengage the lift yoke.

4. Horizontal Handling of HI-TRAC in the Transfer Frame:

a. Verify that the Transfer Frame is secured to the transport vehicle as necessary.

b. Downend HI-TRAC on the Transfer Frame per Step 3, if necessary.

c. If necessary, install the HI-TRAC missile Shield on the HI-STAR 100 Transfer Frame (See Figure 8.1.4).

5. Vertical Handling of HI-STORM:

Note:
The HI-STORM 100 Overpack may be lifted with a special lifting device that engages the overpack anchor blocks with threaded studs and connects to a cask transporter, crane, or similar equipment. The device is designed in accordance with ANSI N14.6.

a. Visually inspect the HI-STORM lifting device for gouges, cracks, deformation or other indications of damage.

b. Visually inspect the transporter lifting attachments for gouges, cracks, deformation or other indications of damage..

c. If necessary, attach the transporter's lifting device to the transporter and HI-STORM..

d. Raise and position HI-STORM accordingly. See Figure 8.1.5.

6. Empty MPC Installation in HI-TRAC:

Note:
To avoid side loading the MPC lift lugs, the MPC must be upended in the MPC Upending Frame (or equivalent). See Figure 8.1.6.

a. If necessary, rinse off any road dirt with water. Remove any foreign objects from the MPC internals.

b. If necessary, upend the MPC as follows:

1. Visually inspect the MPC Upending Frame for gouges, cracks, deformation or other indications of damage. Repair or replace damaged components as necessary.

2. Install the MPC on the Upending Frame. Make sure that the banding straps are secure around the MPC shell. See Figure 8.1.6.

3. Inspect the Upending Frame slings in accordance with the site's lifting equipment inspection procedures. Rig the slings around the bar in a choker configuration to the outside of the cleats. See Figure 8.1.6.

4. Attach the MPC upper end slings of the Upending Frame to the main overhead lifting device. Attach the bottom-end slings to a secondary lifting device (or a chain fall attached to the primary lifting device) (See Figure 8.1.6).

5. Raise the MPC in the Upending Frame.

Warning:
The Upending Frame corner should be kept close to the ground during the upending process.

6. Slowly lift the upper end of the Upending Frame while lowering the bottom end of the Upending Frame.

7. When the MPC approaches the vertical orientation, tension on the lower slings may be released.

8. Place the MPC in a vertical orientation.

9. Disconnect the MPC straps and disconnect the rigging.

c. Install the MPC in HI-TRAC as follows:

1. Install the four point lift sling to the lift lugs inside the MPC. See Figure 8.1.7.

2. Raise and place the MPC inside HI-TRAC.

Note:
An alignment punch mark is provided on HI-TRAC and the top edge of the MPC. Similar marks are provided on the MPC lid and closure ring. See Figure 8.1.8.

3. Rotate the MPC so the alignment marks agree and seat the MPC inside HI-TRAC. Disconnect the MPC rigging or the MPC lift rig.

8.1.3 HI-TRAC and MPC Receipt Inspection and Loading Preparation

Note:

Receipt inspection, installation of the empty MPC in the HI-TRAC, and lower fuel spacer installation may occur at any location or be performed at any time prior to complete submersion in the spent fuel pool as long as appropriate steps are taken to prevent contaminating the exterior of the MPC or interior of the HI-TRAC.

ALARA Note:
A bottom protective cover may be attached to HI-TRAC pool lid bottom. This will help prevent imbedding contaminated particles in HI-TRAC bottom surface and ease the decontamination effort.

1. Place HI-TRAC in the cask receiving area. Perform appropriate contamination and security surveillances, as required.

2. If necessary, remove HI-TRAC Top Lid by removing the top lid bolts and using the lift sling. See Figure 8.1.9 for rigging.

   a. Rinse off any road dirt with water. Inspect all cavity locations for foreign objects. Remove any foreign objects.

   b. Perform a radiological survey of the inside of HI-TRAC to verify there is no residual contamination from previous uses of the cask.

3. Disconnect the rigging.

4. Store the Top Lid and bolts in a site-approved location.

5. If necessary, configure HI-TRAC with the pool lid as follows:

ALARA Warning:
The bottom lid replacement as described below may be performed only on an empty HI-TRAC.

   a. Inspect the seal on the pool lid for cuts, cracks, gaps and general condition. Replace the seal if necessary.

   b. Remove the bottom lid bolts and store them temporarily.

   c. Raise the empty HI-TRAC and position it on top of the pool lid.

   d. Inspect the pool lid bolts for general condition. Replace worn or damaged bolts with new bolts.

   e. Install the pool lid bolts. See Table 8.1.5 for torque requirements.

   f. If necessary, thread the drain connector pipe to the pool lid.

   g. Store the HI-TRAC Transfer Lid in a site-approved location.

6. At the site's discretion, perform an MPC receipt inspection and cleanliness inspection in accordance with a site-specific inspection checklist.

7. Install the MPC inside HI-TRAC and place HI-TRAC in the designated preparation area. See Section 8.1.2.

Note:
Upper fuel spacers are fuel-type specific. Not all fuel types require fuel spacers. Upper fuel spacer installation may occur any time prior to MPC lid installation.

8. Install the upper fuel spacers in the MPC lid as follows:

Warning:
Never work under a suspended load.

   a. Position the MPC lid on supports to allow access to the underside of the MPC lid.

b. Thread the fuel spacers into the holes provided on the underside of the MPC lid. See Figure 8.1.10 and Table 8.1.5 for torque requirements.

c. Install threaded plugs in the MPC lid where and when spacers will not be installed, if necessary. See Table 8.1.5 for torque requirements.

9. At the user's discretion perform an MPC lid and closure ring fit test:

Note:
It may be necessary to perform the MPC installation and inspection in a location that has sufficient crane clearance to perform the operation.

a. Visually inspect the MPC lid rigging (See Figure 8.1.9).

b. At the user's discretion, raise the MPC lid such that the drain line can be installed. Install the drain line to the underside of the MPC lid. Ensure that the reducer is fully seated against the bottom of the MPC lid. See Figure 8.1.11.

c. Align the MPC lid and lift yoke so the drain line will be positioned in the MPC drain location. See Figure 8.1.12. Install the MPC lid. Verify that the MPC lid fit and weld prep are in accordance with the design drawings.

ALARA Note:
The closure ring is installed by hand. Some grinding may be required on the closure ring to adjust the fit.

d. Install, align and fit-up the closure ring.

e. Verify that closure ring fit and weld prep are in accordance with the fabrication drawings or the approved design drawings.

f. Remove the closure ring, vent and drain port cover plates and the MPC lid. Disconnect the drain line. Store these components in an approved plant storage location.

10. At the user's discretion, perform an MPC vent and drain port cover plate fit test and verify that the weld prep is in accordance with the approved fabrication drawings.

Note:
Fuel spacers are fuel-type specific. Not all fuel types require fuel spacers. Lower fuel spacers are set in the MPC cells manually. No restraining devices are used.

11. Install lower fuel spacers in the MPC (if necessary). See Figure 8.1.10.

12. Fill the MPC and annulus as follows:

a. Fill the annulus with plant demineralized water to just below the inflatable seal seating surface.

Caution:
Do not use any sharp tools or instruments to install the inflatable seal. Some air in the inflatable seal helps in the installation.

b. Manually insert the inflatable annulus seal around the MPC. See Figure 8.1.13.

c. Ensure that the seal is uniformly positioned in the annulus area.

d. Inflate the seal.

e. Visually inspect the seal to ensure that it is properly seated in the annulus. Deflate, adjust and inflate the seal as necessary. Replace the seal as necessary.

ALARA Note:
Bolt plugs, placed in, or waterproof tape over empty bolt holes, reduce the time required for decontamination.

13. At the user's discretion, install HI-TRAC top lid bolt plugs and/or apply waterproof tape over any empty bolt holes.

ALARA Note:
Keeping the water level below the top of the MPC prevents splashing during handling.

14. Fill the MPC with either demineralized water or spent fuel pool water to approximately 12 inches below the top of the MPC shell. Refer to ~~LCO 3.3.1~~*Tables 2.1.14 and 2.1.16* for boron concentration requirements.

15. If necessary for plant crane capacity limitations, drain the water from the neutron shield jacket. See Tables 8.1.1 through 8.1.4 as applicable.

16. Place HI-TRAC in the spent fuel pool as follows:

ALARA Note:
The term "Spent Fuel Pool" is used generically to refer to the users designated cask loading location. The optional Annulus Overpressure System is used to provide further protection against MPC external shell contamination during in-pool operations.

a. If used, fill the Annulus Overpressure System lines and reservoir with demineralized water and close the reservoir valve. Attach the Annulus Overpressure System to the HI-TRAC. See Figure 8.1.14.

b. Verify spent fuel pool for boron concentration requirements in accordance with ~~LCO 3.3.1~~*Tables 2.1.14 and 2.1.16.*

c. Engage the lift yoke to HI-TRAC lifting trunnions and position HI-TRAC over the cask loading area with the basket aligned to the orientation of the spent fuel racks.

ALARA Note:
Wetting the components that enter the spent fuel pool may reduce the amount of decontamination work to be performed later.

d. Wet the surfaces of HI-TRAC and lift yoke with plant demineralized water while slowly lowering HI-TRAC into the spent fuel pool.

e. When the top of the HI-TRAC reaches the elevation of the reservoir, open the Annulus Overpressure System reservoir valve. Maintain the reservoir water level at approximately 3/4 full the entire time the cask is in the spent fuel pool.

f. Place HI-TRAC on the floor of the cask loading area and disengage the lift yoke. Visually verify that the lift yoke is fully disengaged. Remove the lift yoke from the spent fuel pool while spraying the crane cables and yoke with plant demineralized water.

g. Observe the annulus seal for signs of air leakage. If leakage is observed (by the steady flow of bubbles emanating from one or more discrete locations) then immediately remove the HI-TRAC from the spent fuel pool and repair or replace the seal.

8.1.4 MPC Fuel Loading

Note:
An underwater camera or other suitable viewing device may be used for monitoring underwater operations.

Note:
When loading MPCs requiring soluble boron, the boron concentration of the water shall be checked in accordance with Tables 2.1.14 and 2.1.16 before and during operations with fuel and water in the MPC.

1. Perform a fuel assembly selection verification using plant fuel records to ensure that only fuel assemblies that meet all the conditions for loading as specified in ~~Appendix B to CoC 72-1014~~*Section 2.1.9* have been selected for loading into the MPC.
2. Load the pre-selected fuel assemblies into the MPC in accordance with the approved fuel loading pattern.
3. Perform a post-loading visual verification of the assembly identification to confirm that the serial numbers match the approved fuel loading pattern.

8.1.5 MPC Closure

Note:
The user may elect to use the Lid Retention System (See Figure 8.1.15) to assist in the installation of the MPC lid and lift yoke, and to provide the means to secure the MPC lid in the event of a drop accident during loaded cask handling operations outside of the spent fuel pool. The user is responsible for evaluating the additional weight imposed on the cask, lift yoke, crane and floor prior to use. See Tables 8.1.1 through 8.1.4 as applicable. The following guidance describes installation of the MPC lid using the lift yoke. The MPC lid may also be installed separately.

Depending on facility configuration, users may elect to perform MPC closure operations with the HI-TRAC partially submerged in the spent fuel pool. If opted, operations involving removal of the HI-TRAC from the spent fuel pool shall be sequenced accordingly.

1. Remove the HI-TRAC from the spent fuel pool as follows:

a. Visually inspect the MPC lid rigging or Lid Retention System in accordance with site-approved rigging procedures. Attach the MPC lid to the lift yoke so that MPC lid, drain line and trunnions will be in relative alignment. Raise the MPC lid and adjust the rigging so the MPC lid hangs level as necessary.

b. Install the drain line to the underside of the MPC lid. Ensure that the reducer is fully seated against the bottom of the MPC lid. See Figure 8.1.17.

c. Align the MPC lid and lift yoke so the drain line will be positioned in the MPC drain location and the cask trunnions will also engage. See Figure 8.1.11 and 8.1.17.

ALARA Note:
Pre-wetting the components that enter the spent fuel pool may reduce the amount of decontamination work to be performed later.

d. Slowly lower the MPC lid into the pool and insert the drain line into the drain access location and visually verify that the drain line is correctly oriented. See Figure 8.1.12.

e. Lower the MPC lid while monitoring for any hang-up of the drain line. If the drain line becomes kinked or disfigured for any reason, remove the MPC lid and replace the drain line.

Note:
The outer diameter of the MPC lid will seat flush with the top edge of the MPC shell when properly installed.

f. Seat the MPC lid in the MPC and visually verify that the lid is properly installed.

g. Engage the lift yoke to HI-TRAC lifting trunnions.

h. Apply a slight tension to the lift yoke and visually verify proper engagement of the lift yoke to the lifting trunnions.

ALARA Note:
Activated debris may have settled on the top face of HI-TRAC and MPC during fuel loading. The cask top surface should be kept under water until a preliminary dose rate scan clears the cask for removal. Users are responsible for any water dilution considerations.

i. Raise HI-TRAC until the MPC lid is just below the surface of the spent fuel pool. Survey the area above the cask lid to check for hot particles. Remove any activated or highly radioactive particles from HI-TRAC or MPC.

j. Visually verify that the MPC lid is properly seated. Lower HI-TRAC, reinstall the lid, and repeat as necessary.

k. Install the Lid Retention System bolts if the lid retention system is used.

l. Continue to raise the HI-TRAC under the direction of the plant's radiological control personnel. Continue rinsing the surfaces with demineralized water. When the top of the HI-TRAC reaches the same elevation as the reservoir, close the Annulus Overpressure System reservoir valve (if used). See Figure 8.1.14.

Caution:

Users are required to take necessary actions to prevent boiling of the water in the MPC. This may be accomplished by performing a site-specific analysis to identify a time limitation to ensure that water boiling will not occur in the MPC prior to the initiation of draining operations. Chapter 4 of the FSAR provides some sample time limits for the time to initiation of draining for various spent fuel pool water temperatures using design basis heat loads. These time limits may be adopted if the user chooses not to perform a site-specific analysis. If time limitations are imposed, users shall have appropriate procedures and equipment to take action. One course of action involves initiating an MPC water flush for a certain duration and flow rate. Any site-specific analysis shall identify the methods to respond should it become likely that the imposed time limit could be exceeded. Refer to ~~LCO 3.3.1~~*Tables 2.1.14 and 2.1.16* for boron concentration requirements whenever water is added to the loaded MPC.

m. Remove HI-TRAC from the spent fuel pool while spraying the surfaces with plant demineralized water. Record the time.

ALARA Note:

Decontamination of HI-TRAC bottom should be performed using remote cleaning methods, covering or other methods to minimize personnel exposure. The bottom lid decontamination may be deferred to a convenient and practical time and location. Any initial decontamination should only be sufficient to preclude spread of contamination within the fuel building.

n. Decontaminate HI-TRAC bottom and HI-TRAC exterior surfaces including the pool lid bottom. Remove the bottom protective cover, if used.

o. If used, disconnect the Annulus Overpressure System from the HI-TRAC See Figure 8.1.14.

p. Set HI-TRAC in the designated cask preparation area.

Note:

If the transfer cask is expected to be operated in an environment below 32 °F, the water jacket shall be filled with an ethylene glycol solution (25% ethylene glycol). Otherwise, the jacket shall be filled with demineralized water. Depending on weight limitations, the neutron shield jacket may remain filled (with pure water or 25% ethylene glycol solution, as required). Users shall evaluate the cask weights to ensure that cask trunnion, lifting devices and equipment load limitations are not exceeded.

q. If previously drained, fill the neutron shield jacket with plant demineralized water or an ethylene glycol solution (25% ethylene glycol) as necessary.

r. Disconnect the lifting slings or Lid Retention System (if used) from the MPC lid and disengage the lift yoke. Decontaminate and store these items in an approved storage location.

Warning:
MPC lid dose rates are measured to ensure that dose rates are within expected values. Dose rates exceeding the expected values could indicate that fuel assemblies not meeting the CoC may have been loaded.

s. Measure the dose rates at the MPC lid and verify that the combined gamma and neutron dose is below expected values.

t. Perform decontamination and a dose rate/contamination survey of HI-TRAC.

u. Prepare the MPC annulus for MPC lid welding as follows:

ALARA Note:
If the Temporary Shield Ring is not used, some form of gamma shielding (e.g., lead bricks or blankets) should be placed in the trunnion recess areas of the HI-TRAC water jacket to eliminate the localized hot spot.

v. Decontaminate the area around the HI-TRAC top flange and install the Temporary Shield Ring, (if used). See Figure 8.1.18.

ALARA Note:
The water in the HI-TRAC-to-MPC annulus provides personnel shielding. The level should be checked periodically and refilled accordingly.

~~x.~~ *w.* Attach the drain line to the HI-TRAC drain port and lower the annulus water level approximately 6 inches.

2. Prepare for MPC lid welding as follows:

Note:
The following steps use two identical Removable Valve Operating Assemblies (RVOAs) (See Figure 8.1.16) to engage the MPC vent and drain ports. The MPC vent and drain ports are equipped with metal-to-metal seals to minimize leakage during drying, and to withstand the long-term effects of temperature and radiation. The RVOAs allow the vent and drain ports to be operated like valves and prevent the need to hot tap into the penetrations during unloading operations. The RVOAs are purposely not installed until the cask is removed from the spent fuel pool to reduce the amount of decontamination.

Note:
The vent and drain ports are opened by pushing the RVOA handle down to engage the square nut on the cap and turning the handle fully in the counter-clockwise direction. The handle will not turn once the port is fully open. Similarly, the vent and drain ports are closed by turning the handle fully in the clockwise direction. The ports are closed when the handle cannot be turned further.

Note:

Steps involving preparation for welding may occur in parallel as long as precautions are taken to prevent contamination of the annulus.

a. Clean the vent and drain ports to remove any dirt. Install the RVOAs (See Figure 8.1.16) to the vent and drain ports leaving caps open.

ALARA Warning:

Personnel should remain clear of the drain hoses any time water is being pumped or purged from the MPC. Assembly crud, suspended in the water, may create a radiation hazard to workers. Controlling the amount of water pumped from the MPC prior to welding keeps the fuel assembly cladding covered with water yet still allows room for thermal expansion.

b. Attach the water pump to the drain port (See Figure 8.1.19) and lower the water level to keep moisture away from the weld region.

c. Disconnect the water pump.

d. Carefully decontaminate the MPC lid top surface and the shell area above the inflatable seal

e. Deflate and remove the inflatable annulus seal.

ALARA Note:

The MPC exterior shell survey is performed to evaluate the performance of the inflatable annulus seal. Indications of contamination could require the MPC to be unloaded. In the event that the MPC shell is contaminated, users must decontaminate the annulus. If the contamination cannot be reduced to acceptable levels, the MPC must be returned to the spent fuel pool and unloaded. The MPC may then be removed and the external shell decontaminated.

f. Survey the MPC lid top surfaces and the accessible areas of the top three inches of the MPC ~~shell in accordance with the requirements of Technical Specification LCO 3.2.2~~.

ALARA Note:

The annulus shield is used to prevent objects from being dropped into the annulus and helps reduce dose rates directly above the annulus region. The annulus shield is hand installed and requires no tools.

g. Install the annulus shield. See Figure 8.1.13.

3. Weld the MPC lid as follows:

ALARA Warning:

Grinding of MPC welds may create the potential for contamination. All grinding activities shall be performed under the direction of radiation protection personnel.

ALARA Warning:

It may be necessary to rotate or reposition the MPC lid slightly to achieve uniform weld gap and lid alignment. A punch mark is located on the outer edge of the MPC lid and shell. These marks are aligned with the alignment mark on the top edge of the HI-TRAC Transfer Cask (See Figure 8.1.8). If necessary, the MPC lid lift should be performed using a hand operated chain fall to closely control the lift to allow rotation and repositioning by hand. If the chain fall is hung from the crane hook, the crane should be tagged out of service to prevent inadvertent use during this operation. Continuous radiation monitoring is recommended.

a. If necessary center the lid in the MPC shell using a hand-operated chain fall.

Note:

The MPC is equipped with lid shims that serve to close the gap in the joint for MPC lid closure weld.

b. As necessary, install the MPC lid shims around the MPC lid to make the weld gap uniform.

ALARA Note:

The AWS Baseplate shield is used to further reduce the dose rates to the operators working around the top cask surfaces.

c. Install the Automated Welding System baseplate shield. See Figure 8.1.9 for rigging.

d. If used, install the Automated Welding System Robot.

Note:

It may be necessary to remove the RVOAs to allow access for the automated welding system. In this event, the vent and drain port caps should be opened to allow for thermal expansion of the MPC water.

Caution:

Oxidation of Boral panels and aluminum components contained in the MPC may create hydrogen gas while the MPC is filled with water. Appropriate monitoring for combustible gas concentrations shall be performed prior to, and during MPC lid welding operations. *The space below the MPC lid shall be exhausted or purged with inert gas prior to, and during MPC lid welding operations to provide additional assurance that flammable gas concentrations will not develop in this space*~~It is also recommended for defense in depth that the space below the MPC lid be exhausted or purged with inert gas prior to, and during MPC lid welding operations to provide additional assurance that explosive gas mixtures will not develop in this space~~.

e. Perform combustible gas monitoring and~~, if desired,~~ exhaust or purge the space under the MPC lid with an inert gas to ensure that there is no combustible mixture present in the welding area.

f. Perform the MPC lid-to-shell weld and NDE with approved procedures (See 9.1 and Table 2.2.15).

g. Deleted.

h. Deleted.

i. Deleted.

j. Deleted.

4. Perform hydrostatic ~~and MPC leakage rate~~ testing as follows:

ALARA Note:
The leakage rates are determined before the MPC is drained for ALARA reasons. A weld repair is a lower dose activity if water remains inside the MPC.

a. Attach the drain line to the vent port and route the drain line to the spent fuel pool or the plant liquid radwaste system. See Figure 8.1.20 for the hydrostatic test arrangement.

ALARA Warning:
Water flowing from the MPC may carry activated particles and fuel particles. Apply appropriate ALARA practices around the drain line.

b. Fill the MPC with either spent fuel pool water or plant demineralized water until water is observed flowing out of the vent port drain hose. Refer to ~~LCO 3.3.1~~*Tables 2.1.14 and 2.1.16* for boron concentration requirements.

c. Perform a hydrostatic test of the MPC as follows:

1. Close the drain valve and pressurize the MPC to 125 +5/-0 psig.

2. Close the inlet valve and monitor the pressure for a minimum of 10 minutes. The pressure shall not drop during the performance of the test.

3. Following the 10-minute hold period, visually examine the MPC lid-to-shell weld for leakage of water. The acceptance criteria is no observable water leakage.

d. Release the MPC internal pressure, disconnect the water fill line and drain line from the vent and drain port RVOAs leaving the vent and drain port caps open.

1. Repeat the liquid penetrant examination on the MPC lid final pass.

~~e. Attach a regulated helium supply to the vent port and attach the drain line to the drain port as shown on Figure 8.1.21.~~

~~f. Verify the correct pressure on the helium supply and open the helium supply valve. Drain approximately twenty gallons.~~

~~g. Close the drain port valve and pressurize the MPC.~~

~~h. Close the vent port.~~

~~i. Perform a helium sniffer probe leakage rate test of the MPC lid-to-shell weld in accordance with the Mass Spectrometer Leak Detector (MSLD) manufacturer's instructions and ANSI N14.5 [8.1.2]. The MPC Helium Leak Rate shall be < 5.0E-6 atm cc/sec (He) based on a 1 atmosphere pressure differential across the weld joint.~~

~~j.~~*e.* Repair any weld defects in accordance with the site's approved weld repair procedures. Reperform the Ultrasonic (if necessary), PT, *and* Hydrostatic ~~and Helium Leakage~~ tests if weld repair is performed.

5. Drain the MPC as follows:

   a. Attach the drain line to the vent port and route the drain line to the spent fuel pool or the plant liquid radwaste system. See Figure 8.1.20.

ALARA Warning:
Water flowing from the MPC may carry activated particles and fuel particles. Apply appropriate ALARA practices around the drain line.

   b. Attach the water fill line to the drain port and fill the MPC with either spent fuel pool water or plant demineralized water until water is observed flowing out of the drain line.

   c. Disconnect the water fill and drain lines from the MPC leaving the vent port valve open to allow for thermal expansion of the MPC water.

ALARA Warning:
Dose rates will rise as water is drained from the MPC. Continuous dose rate monitoring is recommended.

   d. Attach a regulated helium or nitrogen supply to the vent port.

   e. Attach a drain line to the drain port shown on Figure 8.1.21.

   f. Deleted

   g. Verify the correct pressure on the gas supply.

   h. Open the gas supply valve and record the time at the start of MPC draining.

Note:
An optional warming pad may be placed under the HI-TRAC Transfer Cask to replace the heat lost during the evaporation process of MPC drying. This may be used at the user's discretion for older and colder fuel assemblies to reduce vacuum drying times.

   i. Start the warming pad, if used.

Note:
Users may continue to purge the MPC to remove as much water as possible.

   j. Drain the water out of the MPC until water ceases to flow out of the drain line. Shut the gas supply valve. See Figure 8.1.21.

   k. Deleted.

l. Disconnect the gas supply line from the MPC.

m. Disconnect the drain line from the MPC.

6. Dry the MPC as follows:

> ***Caution:***
> *Limitations for the at-vacuum duration are evaluated and established on a canister-specific basis to ensure that acceptable cladding temperatures are not exceeded. Refer to FSAR Section 4.5 for guidance.*

> **Note:**
> Vacuum drying or *forced helium dehydration* ~~moisture removal~~ (for higher ~~burn-up~~*heat load* fuel) is performed to remove moisture and oxidizing gasses from the MPC. This ensures a suitable environment for long-term storage of spent fuel assemblies and ensures that the MPC pressure remains within design limits. The vacuum drying process described herein reduces the MPC internal pressure in stages. Dropping the internal pressure too quickly may cause the formation of ice in the fittings. Ice formation could result in incomplete removal of moisture from the MPC. The moisture removal process limits bulk MPC temperatures by continuously circulating gas through the MPC. Steps *6.b*~~8.1.22a~~ through h *in Section 8.1.5* are used for vacuum drying. Steps ~~8.1.22~~*6*i through k *in Section 8.1.5* are used for moisture removal.

a. *If using the vacuum drying system, go to Section 8.1.5 Step 6.b. If using the forced helium dehydration system, go to Section 8.1.5 Step 6.i.*~~Attach the drying system (VDS) to the vent and drain port RVOAs. See Figure 8.1.22a for the vacuum drying system and 8.1.22b for the moisture removal system.~~ Other equipment configurations that achieve the same results may also be used.

> **Note:**
> The vacuum drying system may be configured with an optional fore-line condenser. Other equipment configurations that achieve the same results may be used.

b. *Attach the vacuum drying system (VDS) to the vent and drain port RVOAs. See Figure 8.1.22a*~~Deleted~~.

c. Deleted.

d. Deleted.

> ***Note:***
> *To prevent freezing of water, the MPC internal pressure should be lowered in incremental steps. The vacuum drying system pressure will remain at about 30 torr until most of the liquid water has been removed from the MPC.*

e. *Open the VDS suction valve and reduce the MPC pressure to below 3 torr*~~Deleted~~.

f. *Shut the VDS valves and verify a stable MPC pressure on the vacuum gauge*~~Open the VDS suction valve and reduce the MPC pressure to below 3 torr~~.

> ***Note:***
> *The MPC pressure may rise due to the presence of water in the MPC. The dryness test may need to be repeated several times until all the water has been removed. Leaks in the vacuum drying system, damage to the vacuum pump, and improper vacuum gauge calibration may cause repeated failure of the dryness verification test. These conditions should be checked as part of the corrective actions if repeated failure of the dryness verification test is occurring.*

g. *Perform the MPC drying pressure test*~~Shut the VDS valves and verify a stable MPC pressure on the vacuum gage~~.

h. ~~Perform the MPC drying pressure test in accordance with the technical specifications.~~*Proceed to Step 6.l of Section 8.1.5 if not using the forced helium dehydration system.*

i. Attach the ~~moisture removal~~*forced helium dehydration* system to the vent and drain port RVOAs. See Figure 8.1.22b.

j. Circulate the drying gas though the MPC while monitoring the circulating gas for moisture. Collect and remove the moisture from the system as necessary.

k. Continue the monitoring and moisture removal until ~~LCO 3.1.1 is~~ *the acceptance criteria are* met for MPC dryness. Discontinue MPC drying operations.

l. If necessary, attach the vacuum pump to the MPC.

m. Evacuate the MPC to below 10 torr.

n. Close the vent and drain port valves.

o. Disconnect the VDS from the MPC.

p. Stop the warming pad, if used.

q. Close the drain port RVOA cap and remove the drain port RVOA.

7. Backfill the MPC as follows:

> ***Caution:***
> *Limitations for the handling of the loaded MPC in HI-TRAC are evaluated and established on a canister basis to ensure that acceptable cladding temperatures are not exceeded. Refer to FSAR Section 4.5 for specific time limits for transporting a loaded MPC in the HI-TRAC transfer CASK based on MPC heat loads.*

> **Note:**
> Helium backfill shall be ~~in accordance with the Technical Specification at~~*performed using helium with* 99.995% (minimum) purity. Other equipment configurations that achieve the same results may be used.

a. Set the helium bottle regulator pressure to the appropriate pressure.

b. Purge the Helium Backfill System to remove oxygen from the lines.

c. Attach the Helium Backfill System to the vent port as shown on Figure 8.1.23 and open the vent port.

d. Slowly open the helium supply valve while monitoring the pressure rise in the MPC.

e. Deleted

f. Deleted

g. Deleted

Note:
If helium bottles need to be replaced, the bottle valve needs to be closed and the entire regulator assembly transferred to the new bottle.

h. Carefully backfill the MPC ~~in accordance with the technical specifications~~*in accordance with Table 4.4.37.*

i. Disconnect the helium backfill system from the MPC.

j. Close the vent port RVOA and disconnect the vent port RVOA.

8. Weld the vent and drain port cover plates as follows:

Note:
The process provided herein may be modified to perform actions in parallel. ~~Users may perform the final PT on the circumferential and plug welds at the same time.~~

a. Wipe the inside area of the vent and drain port recesses to dry and clean the surfaces.

b. Place the cover plate over the vent port recess.

c. Weld the cover plate.

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

Note:
ASME Boiler and Pressure Vessel Code [8.1.3], Section V, Article 6 provides the liquid penetrant inspection methods. The acceptance standards for liquid penetrant examination shall be in accordance with ASME Boiler and Pressure Vessel Code, Section III, Subsection NB, Article NB-5350 as specified on the Design Drawings. ASME Code, Section III, Subsection NB, Article NB-4450 provides acceptable requirements for weld repair. NDE personnel shall be qualified per the requirements of Section V of the Code or site-specific program.

~~f.~~*d. Perform NDE on the cover plate with approved procedures (See 9.1 and Table 2.2.15)*~~Deleted.~~

~~g.~~*e.* *Repair any weld defects in accordance with the site's approved code weld repair procedures.*~~Deleted~~.

~~h.~~*f.* Deleted.

~~i.~~*g.* Deleted.

~~j.~~*h.* Deleted.

~~k.~~*i.* Repeat for the drain port cover plate.

~~9.Perform a leakage test of the MPC vent and drain port cover plates as follows:~~

~~a.~~

~~b.Repair any weld defects in accordance with the site's approved code weld repair procedures. Re-perform the leakage test as required.~~

~~10.~~*9.* Weld the MPC closure ring as follows:

ALARA Note:
The closure ring is installed by hand. No tools are required. Localized grinding to achieve the desired fit and weld prep are allowed.

a. Install and align the closure ring. See Figure 8.1.8.

b. Weld the closure ring to the MPC shell and the MPC lid, and perform NDE with approved procedures (See 9.1 and Table 2.2.15).

c. Deleted.

d. Deleted.

e. Deleted.

f. Deleted.

g. Deleted.

h. Deleted.

i. Deleted.

j. If necessary, remove the AWS. See Figure 8.1.7 for rigging.

8.1.6 Preparation for Storage

ALARA Warning:
Dose rates will rise around the top of the annulus as water is drained from the annulus. Apply appropriate ALARA practices.
Caution:
Limitations for the handling of the loaded MPC in HI-TRAC are evaluated and established on a canister basis to ensure that acceptable cladding temperatures are not exceeded. Refer to FSAR Section 4.5 for more detailed guidance on time limits based on MPC heat loads.

1. Remove the annulus shield (if used) and store it in an approved plant storage location

2. Attach a drain line to the HI-TRAC and drain the remaining water from the annulus to the spent fuel pool or the plant liquid radwaste system.

3. Install HI-TRAC top lid as follows:

Warning:
When traversing the MPC with the HI-TRAC top lid using non-single-failure proof (or equivalent safety factors), the lid shall be kept less than 2 feet above the top surface of the MPC. This is performed to protect the MPC lid from a potential lid drop.

a. Install HI-TRAC top lid. Inspect the bolts for general condition. Replace worn or damaged bolts with new bolts.

b. Install and torque the top lid bolts. See Table 8.1.5 for torque requirements.

c. Inspect the lift cleat bolts for general condition. Replace worn or damaged bolts with new bolts.

d. Install the MPC lift cleats and MPC slings. See Figure 8.1.24 and 8.1.25. See Table 8.1.5 for torque requirements.

e. Drain and remove the Temporary Shield Ring, if used.

4. Replace the pool lid with the transfer lid as follows (Not required for HI-TRAC 125D):

ALARA Note:
The transfer slide is used to perform the bottom lid replacement and eliminate the possibility of directly exposing the bottom of the MPC. The transfer slide consists of the guide rails, rollers, transfer step and carriage. The transfer slide carriage and jacks are powered and operated by remote control. The carriage consists of short-stroke hydraulic jacks that raise the carriage to support the weight of the bottom lid. The transfer step produces a tight level seam between the transfer lid and the pool lid to minimize radiation streaming. The transfer slide jacks do not have sufficient lift capability to support the entire weight of the HI-TRAC. This was selected specifically to limit floor loads. Users should designate a specific area that has sufficient room and support for performing this operation.

Note:
The following steps are performed to pretension the MPC slings.

a. Lower the lift yoke and attach the MPC slings to the lift yoke. See Figure 8.1.25.

b. Raise the lift yoke and engage the lift yoke to the HI-TRAC lifting trunnions.

c. If necessary, position the transfer step and transfer lid adjacent to one another on the transfer slide carriage. See Figure 8.1.26. See Figure 8.1.9 for transfer step rigging.

d. Deleted.

e. Position HI-TRAC with the pool lid centered over the transfer step approximately one inch above the transfer step.

f. Raise the transfer slide carriage so the transfer step is supporting the pool lid bottom. Remove the bottom lid bolts and store them temporarily.

ALARA Warning:
Clear all personnel away from the immediate operations area. The transfer slide carriage and jacks are remotely operated. The carriage has fine adjustment features to allow precise positioning of the lids.

g. Lower the transfer carriage and position the transfer lid under HI-TRAC.

h. Raise the transfer slide carriage to place the transfer lid against the HI-TRAC bottom lid bolting flange.

i. Inspect the transfer lid bolts for general condition. Replace worn or damaged bolts with new bolts.

j. Install the transfer lid bolts. See Table 8.1.5 for torque requirements.

k. Raise and remove the HI-TRAC from the transfer slide.

l. Disconnect the MPC slings and store them in an approved plant storage location.

Note:
HI-STORM receipt inspection and preparation may be performed independent of procedural sequence.

5. Perform a HI-STORM receipt inspection and cleanliness inspection in accordance with a site-approved inspection checklist, if required. See Figure 8.1.27 for HI-STORM lid rigging.

Note:
MPC transfer may be performed in the truck bay area, at the ISFSI, or any other location deemed appropriate by the licensee. The following steps describe the general transfer operations (See Figure 8.1.28). The HI-STORM may be positioned on an air pad, roller skid in the cask receiving area or at the ISFSI. The HI-STORM or HI-TRAC may be transferred to the ISFSI using a heavy haul transfer trailer, special transporter or other equipment specifically designed for such a function (See Figure 8.1.29) as long as the HI-TRAC and HI-STORM lifting requirements ~~as described in the Technical Specifications~~ are not exceeded. The licensee is responsible for assessing and controlling floor loading conditions during the MPC transfer operations. Installation of the lid, vent screen, and other components may vary according to the cask movement methods and location of MPC transfer.

8.1.7 Placement of HI-STORM into Storage

1. Position an empty HI-STORM module at the designated MPC transfer location. The HI-STORM may be positioned on the ground, on a deenergized air pad, on a roller skid, on a flatbed trailer or other special device designed for such purposes. If necessary, remove the exit vent screens and gamma shield cross plates ,temperature elements and the HI-STORM lid. See Figure 8.1.28 for some of the various MPC transfer options.

a. Rinse off any road dirt with water. Inspect all cavity locations for foreign objects. Remove any foreign objects.

b. Transfer the HI-TRAC to the MPC transfer location.

2. De-energize the air pad or chock the vehicle wheels to prevent movement of the HI-STORM during MPC transfer and to maintain level, as required.

ALARA Note:
The HI-STORM vent duct shield inserts eliminate the streaming path created when the MPC is transferred past the exit vent ducts. Vent duct shield inserts are not used with the HI-STORM 100S.

3. Install the alignment device (or mating device for HI-TRAC 125D) and if necessary, install the HI-STORM vent duct shield inserts. See Figure 8.1.30.

4. Deleted.

5. Position HI-TRAC above HI-STORM. See Figure 8.1.28.

6. Align HI-TRAC over HI-STORM (See Figure 8.1.31) and mate the overpacks.

7. If necessary, attach the MPC Downloader. See Figure 8.1.32.

8. Attach the MPC slings to the MPC lift cleats.

9. Raise the MPC slightly to remove the weight of the MPC from the transfer lid doors (or pool lid for HI-TRAC 125D and mating device)

10. If using the HI-TRAC 125D, unbolt the pool lid from the HI-TRAC..

11. Remove the transfer lid door (or mating device drawer) locking pins and open the doors (or drawer).

ALARA Warning:
MPC trim plates are used to eliminate the streaming path above and below the doors (or drawer). If trim plates are not used, personnel should remain clear of the immediate door area during MPC downloading since there may be some radiation streaming during MPC raising and lowering operations.

12. At the user's discretion, install trim plates to cover the gap above and below the door/drawer. The trim plates may be secured using hand clamps or any other method deemed suitable by the user. See Figure 8.1.33.

13. Lower the MPC into HI-STORM.

14. Disconnect the slings from the MPC lifting device and lower them onto the MPC lid.

15. Remove the trim plates (if used), and close the doors (or mating device drawer)

ALARA Warning:
Personnel should remain clear (to the maximum extent practicable) of the HI-STORM annulus when HI-TRAC is removed due to radiation streaming.

Note:
It may be necessary, due to site-specific circumstances, to move HI-STORM from under the empty HI-TRAC to install the HI-STORM lid, while inside the Part 50 facility. In these cases. users shall evaluate the specifics of their movements within the requirements of their Part 50 license.

16. Remove HI-TRAC from on top of HI-STORM.

17. Remove the MPC lift cleats and MPC slings and install hole plugs in the empty MPC bolt holes. See Table 8.1.5 for torque requirements.

18. Place HI-STORM in storage as follows:

    a. Remove the alignment device (mating device with HI-TRAC pool lid for HI-TRAC 125D)and vent duct shield inserts (if used). See Figure 8.1.30.

    b. Inspect the HI-STORM lid studs and nuts for general condition. Replace worn or damaged components with new ones.

    c. If used, inspect the HI-STORM 100A anchor components for general condition. Replace worn or damaged components with new ones.

    d. Deleted.

Warning:
Unless the lift is single failure proof (or equivalent safety factor) for the HI-STORM Lid, the lid shall be kept less than 2 feet above the top surface of the overpack. This is performed to protect the MPC lid from a potential HI-STORM 100 lid drop.

Note:
Shims may be used on the HI-STORM 100 lid studs. If used, the shims shall be positioned to ensure a radial gap of less than 1/8 inch around each stud. The method of cask movement will determine the most effective sequence for vent screen, lid, temperature element, and vent gamma shield cross plate installation.

    e. Install the HI-STORM lid and the lid studs and nuts.. See Table 8.1.5 for bolting requirements. Install the HI-STORM 100 lid stud shims if necessary. See Figure 8.1.27 for rigging.

    f. Install the HI-STORM exit vent gamma shield cross plates, temperature elements (if used) and vent screens. See Table 8.1.5 for torque requirements. See Figure 8.1.34a and 8.1.34b.

    g. Remove the HI-STORM lid lifting device and install the hole plugs in the empty holes. Store the lifting device in an approved plant storage location. See Table 8.1.5 for torque requirements.

    h. Secure HI-STORM to the transporter device as necessary.

19. Perform a transport route walkdown to ensure that the cask transport conditions are met. ~~See Technical Specification for the on-site cask handling limitations.~~

20. Transfer the HI-STORM to its designated storage location at the appropriate pitch. See Figure 8.1.35.

Note:
Any jacking system shall have the provisions to ensure uniform loading of all four jacks during the lifting operation.

a. If air pads were used, insert the HI-STORM lifting jacks and raise HI-STORM. See Figure 8.1.36. Remove the air pad.

b. Lower and remove the HI-STORM lifting jacks, if used.

c. For HI-STORM 100A overpack (anchored), perform the following:

1. Inspect the anchor stud receptacles and verify that they are clean and ready for receipt of the anchor hardware.

2. Align the overpack over the anchor location.

3. Lower the overpack to the ground while adjusting for alignment.

4. Install the anchor connecting hardware (See Table 8.1.5 for torque requirements).

21. Install the HI-STORM inlet vent gamma shield cross plates and vent screens. See Table 8.1.5 for torque requirements. See Figure 8.1.34.

22. Perform shielding effectiveness testing ~~per Technical Specification LCO 3.2.3~~.

23. Perform an air temperature rise test as follows for the first HI-STORM 100 System placed in service:

Note:
The air temperature rise test shall be performed between 5 and 7 days after installation of the HI-STORM 100 lid to allow thermal conditions to stabilize. The purpose of this test is to confirm the initial performance of the HI-STORM 100 ventilation system.

a. Measure the inlet air (or screen surface) temperature at the center of each of the four vent screens. Determine the average inlet air (or surface screen) temperature.

b. Measure the outlet air (or screen surface) temperature at the center of each of the four vent screens. Determine the average outlet air (or surface screen) temperature.

c. Determine the average air temperature rise by subtracting the results of the average inlet screen temperature from the average outlet screen temperature.

d. Report the results to the certificate holder.

Table 8.1.1
ESTIMATED HANDLING WEIGHTS OF HI-STORM 100 SYSTEM COMPONENTS
125-TON HI-TRAC**

Component	MPC-24 (Lbs.)	MPC-32 (Lbs.)	MPC-68 (Lbs.)	Case† Applicability					
				1	2	3	4	5	6
Empty HI-STORM 100 overpack (without lid)††	245,040	245,040	245,040					1	
HI-STORM 100 lid (without rigging)	23,963	23,963	23,963					1	
Empty HI-STORM 100S (232) overpack (without lid)††	230,000	230,000	230,000					1	
Empty HI-STORM 100S (243) overpack (without lid) ††	239,000	239,000	239,000					1	
HI-STORM 100S lid (without rigging)	25,500	25,500	25,500					1	
Empty MPC (without lid or closure ring including drain line)	29,845	24,503	29,302	1	1	1	1	1	1
MPC lid (without fuel spacers or drain line)	9,677	9,677	10,194	1	1	1	1	1	1
MPC Closure Ring	145	145	145			1	1	1	1
Fuel (design basis)	40,320	53,760	47,600	1	1	1	1	1	1
Damaged Fuel Container (Dresden 1)	0	0	150						
Damaged Fuel Container (Humboldt Bay)	0	0	120						
MPC water (with fuel in MPC)	17,630	17,630	16,957	1	1				
Annulus Water	256	256	256	1	1				
HI-TRAC Lift Yoke (with slings)	3,600	3,600	3,600	1	1	1			
Annulus Seal	50	50	50	1	1				
Lid Retention System	2,300	2,300	2,300						
Transfer frame	6,700	6,700	6,700						1
Mating Device	15,000	15,000	15,000						
Empty HI-TRAC 125 (without Top Lid, neutron shield jacket water, or bottom lids)	117,803	117,803	117,803	1	1	1			1
Empty HI-TRAC 125D (without Top Lid, neutron shield jacket water, or bottom lids)	119,400	119,400	119,400	1	1	1			1
HI-TRAC 125 Top Lid	2,745	2,745	2,745			1			1
HI-TRAC 125D Top Lid	2,645	2,645	2,645			1			1
Optional HI-TRAC Lid Spacer (weight lbs/in thickness)	400	400	400						
HI-TRAC 125/125D Pool Lid(with bolts)	11,900	11,900	11,900	1	1				
HI-TRAC Transfer Lid (with bolts) (125 Only)	23,437	23,437	23,437			1			1
HI-TRAC 125 Neutron Shield Jacket Water	8,281	8,281	8,281		1	1			1
HI-TRAC 125 D Neutron Shield Jacket Water	9,000	9,000	9,000		1	1			1
MPC Stays (total of 2)	200	200	200						
MPC Lift Cleat	480	230	480			1	1		1

** Actual component weights are dependant upon as-built dimensions. The values provided herein are estimated. FSAR analyses use bounding values provided elsewhere. Users are responsible for ensuring lifted loads meet site capabilities and requirements.

† See Table 8.1.2 for a description of each load handling case.
†† Add an additional 1955 lbs. for the HI-STORM 100A overpack.

TABLE 8.1.2
ESTIMATED HANDLING WEIGHTS
125-TON HI-TRAC**

Caution:
The maximum weight supported by the 125-Ton HI-TRAC lifting trunnions cannot exceed 250,000 lbs. Users must take actions to ensure that this limit is not exceeded.

Note:
The weight of the fuel spacers and the damaged fuel container are less than the weight of the design basis fuel assembly for each MPC and are therefore not included in the maximum handling weight calculations. Fuel spacers are determined to be the maximum combination weight of fuel + spacer. Users should determine their specific handling weights based on the MPC contents and the expected handling modes.

Case No.	Load Handling Evolution	Weight (lbs)		
		MPC-24	MPC-32	MPC-68
1	Loaded HI-TRAC 125 removal from spent fuel pool (neutron tank empty)	231,300	239,300	237,800
2	Loaded HI-TRAC 125 removal from spent fuel pool (neutron tank full)	239,500	247,600	246,100
3	Loaded HI-TRAC 125 During Movement through Hatchway	236,500	244,300	243,700
1A	Loaded HI-TRAC 125D removal from spent fuel pool (neutron tank empty)	232,800	240,900	239,400
2A	Loaded HI-TRAC 125D removal from spent fuel pool (neutron tank full)	241,800	249,900	248,400
3A	Loaded HI-TRAC 125D During Movement through Hatchway	227,300	235,100	234,500
4	MPC during transfer operations	80,467	88,315	87,721
5A	Loaded HI-STORM 100 in storage (See Note 5 to Table 8.1.1)	348,990	357,088	356,244
5B	Loaded HI-STORM 100S (232) in storage (See Note 5 to Table 8.1.1)	335,500	343,600	342,800
5C	Loaded HI-STORM 100S (243) in storage (See Note 5 to Table 8.1.1)	344,500	352,600	351,800
6	Loaded HI-TRAC and transfer frame during on site handling	239,434	247,282	246,688

** Actual component weights are dependant upon as-built dimensions. The values provided herein are estimated. FSAR analyses use bounding values provided elsewhere. Users are responsible for ensuring lifted loads meet site capabilities and requirements.

Table 8.1.3
ESTIMATED HANDLING WEIGHTS OF HI-STORM 100 SYSTEM COMPONENTS100-TON HI-TRAC**

Component	MPC-24 (Lbs.)	MPC-32 (Lbs.)	MPC-68 (Lbs.)	Case† Applicability					
				1	2	3	4	5	6
Empty HI-STORM 100 overpack (without lid)††	245,040	245,040	245,040					1	
HI-STORM 100 lid (without rigging)	23,963	23,963	23,963					1	
Empty HI-STORM 100S (232) overpack (without lid)††	230,000	230,000	230,000					1	
Empty HI-STORM 100S (243) overpack (without lid)††	239,000	239,000	239,000					1	
HI-STORM 100S lid (without rigging)	25,500	25,500	25,500						
Empty MPC (without lid or closure ring including drain line)	29,845	24,503	29,302	1	1	1	1	1	1
MPC lid (without fuel spacers or drain line)	9,677	9,677	10,194	1	1	1	1	1	1
MPC Closure Ring	145	145	145			1	1	1	1
Fuel (design basis)	40,320	53,760	47,600	1	1	1	1	1	1
Damaged Fuel Container (Dresden 1)	0	0	150						
Damaged Fuel Container (Humboldt Bay)	0	0	120						
MPC water (with fuel in MPC)	17,630	17,630	16,957	1	1				
Annulus Water	256	256	256	1	1				
HI-TRAC Lift Yoke (with slings)	3,200	3,200	3,200	1	1	1			
Annulus Seal	50	50	50	1	1				
Lid Retention System	2,300	2,300	2,300						
Transfer frame	6,700	6,700	6,700						1
Empty HI-TRAC (without Top Lid, neutron shield jacket water, or bottom lids)	84,003	84,003	84,003	1	1	1			1
HI-TRAC Top Lid	1,189	1,189	1,189			1			1
HI-TRAC Pool Lid	7,863	7,863	7,863	1	1				
HI-TRAC Transfer Lid	16,686	16,686	16,686			1			1
HI-TRAC Neutron Shield Jacket Water	7,583	7,583	7,583		1	1			1
MPC Stays (total of 2)	200	200	200						
MPC Lift Cleat	480	480	480				1		1

** Actual component weights are dependant upon as-built dimensions. The values provided herein are estimated. FSAR analyses use bounding values provided elsewhere. Users are responsible for ensuring lifted loads meet site capabilities and requirements.

† See Table 8.1.4 for a description of each load handling case.
†† Add an additional 1955 lbs. for the HI-STORM 100A overpack.

Table 8.1.4
ESTIMATED HANDLING WEIGHTS
100-TON HI-TRAC**

Caution:
The maximum weight supported by the 100-Ton HI-TRAC lifting trunnions cannot exceed 200,000 lbs. Users must take actions to ensure that this limit is not exceeded.

Note:
The weight of the fuel spacers and the damaged fuel container are less than the weight of the design basis fuel assembly and therefore not included in the maximum handling weight calculations. Fuel spacers are determined to be the maximum combination weight of fuel + spacer. Users should determine the handling weights based on the contents to be loaded and the expected mode of operations.

Case No.	Load Handling Evolution	Weight (lbs)		
		MPC-24	MPC-32	MPC-68
1	Loaded HI-TRAC removal from spent fuel pool (neutron tank empty)	192,844	200,942	199,425
2	Loaded HI-TRAC removal from spent fuel pool (neutron tank full)	200,427	208,525	207,008
3	Loaded HI-TRAC During Movement through Hatchway	192,647	200,745	199,901
4	MPC during transfer operations	80,467	88,565	87,721
5A	Loaded HI-STORM 100 in storage (See Note 5 to Table 8.1.1)	348,990	357,088	356,244
5B	Loaded HI-STORM 100S (232) in storage (See Note 5 to Table 8.1.1)	335,500	343,600	342,700
5C	Loaded HI-STORM 100S (243) in storage (See Note 5 to Table 8.1.1)	344,500	352,600	351,700
6	Loaded HI-TRAC and transfer frame during on site handling	196,627	204,725	203,881

** Actual component weights are dependant upon as-built dimensions. The values provided herein are estimated. FSAR analyses use bounding values provided elsewhere. Users are responsible for ensuring lifted loads meet site capabilities and requirements.

Table 8.1.5
HI-STORM 100 SYSTEM TORQUE REQUIREMENTS

Fastener†	Torque (ft-lbs) ††	Pattern†††
HI-TRAC Top Lid Bolts†	Hand tight	None
HI-TRAC Pool Lid Bolts (36 Bolt Lid)†	58 ft-lbs	Figure 8.1.37
HI-TRAC Pool Lid Bolts (16 Bolt Lid)†	110 ft-lbs	Figure 8.1.37
100-Ton HI-TRAC Transfer Lid Bolts†	203 ft-lbs	Figure 8.1.37
125-Ton HI-TRAC Transfer Lid Bolts†	270 ft-lbs	Figure 8.1.37
MPC Lift Cleats Stud Nuts†	793 ft-lbs	None
MPC Lift Hole Plugs†	Hand tight	None
Threaded Fuel Spacers	Hand Tight	None
HI-STORM Lid Nuts †	100 ft-lbs	None
HI-STORM 100S Lid Nuts† (Temporary and Permanent Lids)	Hand Tight +1/8 to ½ turn	None
Door Locking Pins	Hand Tight + 1/8 to 1/2 turn	None
HI-STORM 100 Vent Screen/Temperature Element Screws	Hand Tight	None
HI-STORM 100A Anchor Studs	55- 65 ksi tension applied by bolt tensioner (no initial torque)	None

† Studs and nuts shall be cleaned and inspected for damage or excessive thread wear (replace if necessary) and coated with a light layer of Fel-Pro Chemical Products, N-5000, Nuclear Grade Lubricant (or equivalent).

†† Unless specifically specified, torques have a +/- 5% tolerance.

††† No detorquing pattern is needed.

Table 8.1.6
HI-STORM 100 SYSTEM ANCILLARY EQUIPMENT OPERATIONAL DESCRIPTION

Equipment	Important To Safety Classification	Reference Figure†	Description
Air Pads/Rollers	Not Important To Safety	8.1.29	Used for HI-STORM or HI-TRAC cask positioning. May be used in conjunction with the cask transporter or other HI-STORM 100 or HI-TRAC lifting device.
Annulus Overpressure System	Not Important To Safety	8.1.14	The Annulus Overpressure System is used for protection against spent fuel pool water contamination of the external MPC shell and baseplate surfaces by providing a slight annulus overpressure during in-pool operations.
Annulus Shield	Not Important To Safety	8.1.13	A shield that is placed at the top of the HI-TRAC annulus to provide supplemental shielding to the operators performing cask loading and closure operations.
Automated Welding System	Not Important To Safety	8.1.2b	Used for remote field welding of the MPC.
AWS Baseplate Shield	Not Important To Safety	8.1.2b	Provides supplemental shielding to the operators during the cask closure operations.
Bottom Lid Transfer Slide (Not used with HI-TRAC 125D)	Not Important To Safety	8.1.26	Used to simultaneously replace the pool lid with the transfer lid under the suspended HI-TRAC and MPC. Used in conjunction with the bottom lid transfer step.
Cask Transporter	Not Important to Safety unless site-specific conditions require transfer cask or overpack handling outside drop analysis basis.	8.1.29a and 8.1.29b	Used for handling of the HI-STORM 100 Overpack and/or the HI-TRAC Transfer Cask around the site. The cask transporter may take the form of heavy haul transfer trailer, special transporter or other equipment specifically designed for such a function.
Cool-Down System	Not Important To Safety	8.3.4	A closed-loop forced ventilation cooling system used to gas-cool the MPC fuel assemblies down to a temperature at which water can be introduced without the risk of uncontrolled pressure transients in the MPC due to flashing or thermally shocking the fuel assemblies. The cool-down system is attached between the MPC drain and vent ports. The cool-down system is used only for unloading operations.

† Figures are representative and may not depict all configurations for all users.

Table 8.1.6
HI-STORM 100 SYSTEM ANCILLARY EQUIPMENT OPERATIONAL DESCRIPTION
(Continued)

Equipment	Important To Safety Classification	Reference Figure[†]	Description
Lid and empty component lifting rigging	Not Important To Safety, Rigging shall be provided in accordance with NUREG 0612	8.1.9	Used for rigging such components such as the HI-TRAC top lid, pool lid, MPC lid, transfer lid, AWS, HI-STORM Lid and auxiliary shielding and the empty MPC.
Helium Backfill System	Not Important To Safety	8.1.23	Used for controlled insertion of helium into the MPC for leakage testing, blowdown and placement into storage.
HI-STORM 100 Lifting Jacks	Not Important To Safety	8.1.36	Jack system used for lifting the HI-STORM overpack to provide clearance for inserting or removing a device for transportation. .
Alignment Device	Not Important To Safety	8.1.31	Guides HI-TRAC into place on top of HI-STORM for MPC transfers. (Not used for HI-TRAC 125D)
HI-STORM Lifting Devices	Determined site-specifically based on type, location, and height of lift being performed. Lifting devices shall be provided in accordance with ANSI N14.6.	Not shown.	A special lifting device used for connecting the crane (or other primary lifting device) to the HI-STORM 100 for cask handling. Does not include the crane hook (or other primary lifting device) device.
HI-STORM Vent Duct Shield Inserts	Important to Safety Category C .	8.1.30	Used for prevention of radiation streaming from the HI-STORM 100 exit vents during MPC transfers to and from HI-STORM. Not used with the HI-STORM 100S.
HI-TRAC Lid Spacer	Spacer Ring is Not-Important-To-Safety, Studs or bolts are I Important to Safety Category B	Not Shown	Optional ancillary which is used during MPC transfer operations to increase the clearance between the top of the MPC and the underside of the HI-TRAC top lid. Longer threaded studs (or bolts), supplied with the lid spacer, replace the standard threaded studs (or bolts) supplied with the HI-TRAC. The HI-TRAC lid spacer may ONLY be used when the HI-TRAC is handled in the vertical orientation or if HI-TRAC transfer lid is NOT used. The height of the spacer shall be limited to ensure that the weights and C.G. heights in a loaded HI-TRAC with the spacer do not exceed the bounding values found in Section 3.2 of the FSAR.
HI-TRAC Lift Yoke/Lifting Links	Determined site-specifically based on type and location, and height of lift being performed. Lift yoke and lifting devices for loaded HI-TRAC handling shall be provided in accordance with ANSI N14.6.	8.1.3	Used for connecting the crane (or other primary lifting device) to the HI-TRAC for cask handling. Does not include the crane hook (or other primary lifting device).

[†] Figures are representative and may not depict all configurations for all users.

Table 8.1.6
HI-STORM 100 SYSTEM ANCILLARY EQUIPMENT OPERATIONAL DESCRIPTION
(Continued)

Equipment	Important To Safety Classification	Reference Figure†	Description
HI-TRAC transfer frame	Not Important To Safety	8.1.4	A steel frame used to support HI-TRAC during delivery, on-site movement and upending/downending operations.
Cask Primary Lifting Device (Cask Transfer Facility)	Important to Safety. Quality classification of subcomponents determined site-specifically.	8.1.28 and 8.1.32	Optional auxiliary (Non-Part 50) cask lifting device(s) used for cask upending and downending and HI-TRAC raising for positioning on top of HI-STORM to allow MPC transfer. The device may consist of a crane, lifting platform, gantry system or any other suitable device used for such purpose.
Inflatable Annulus Seal	Not Important To Safety	8.1.13	Used to prevent spent fuel pool water from contaminating the external MPC shell and baseplate surfaces during in-pool operations.
Lid Retention System	Important to Safety Status determined by each licensee. MPC lid lifting portions of the Lid Retention System shall meet the requirements of ANSI N14.6.	8.1.15, 8.1.17	Optional. The Lid Retention System secures the MPC lid in place during cask handling operations between the pool and decontamination pad.
MPC Lift Cleats	Important To Safety – Category A. MPC Lift Cleats shall be provided in accordance with of ANSI N14.6.	8.1.24	MPC lift cleats consist of the cleats and attachment hardware. The cleats are supplied as solid steel components that contain no weds. The MPC lift cleats are used to secure the MPC inside HI-TRAC during bottom lid replacement and support the MPC during MPC transfer from HI-TRAC into HI-STORM and vice versa. The ITS classification of the lifting device attached to the cleats may be lower than the cleat itself, as determined site-specifically.
Hydrostatic Test System	Not Important to Safety	8.1.20	Used to pressure test the MPC lid-to-shell weld.
MPC Downloader	Important To Safety status determined site-specifically. MPC Downloader Shall meet the requirements of CoC, Appendix B, Section 3.5 .	8.1.28 and 8.1.32	A lifting device used to help raise and lower the MPC during MPC transfer operations to limit the lift force of the MPC against the top lid of HI-TRAC. The MPC downloader may take several forms depending on the location of MPC transfer and may be used in conjunction with other lifting devices.

† Figures are representative and may not depict all configurations for all users.

Table 8.1.6
HI-STORM 100 SYSTEM ANCILLARY EQUIPMENT OPERATIONAL DESCRIPTION
(Continued)

Equipment	Important To Safety Classification	Reference Figure[†]	Description
Deleted			
Deleted			
Mating Device	Important-To-Safety – Category B	8.1.31	Used to mate HI-TRAC 125D to HI-STORM during transfer operations. Includes sliding drawer for use in removing HI-TRAC pool lid.
MPC Support Slings	Important To Safety – Category A – Rigging shall be provided in accordance with NUREG 0612.	8.1.25	Used to secure the MPC to the lift yoke during HI-TRAC bottom lid replacement operations. Attaches between the MPC lift cleats and the lift yoke. Can be configured for different crane hook configuration.
MPC Upending Frame	Not Important to Safety	8.1.6	A steel frame used to evenly support the MPC during upending operations. and control the upending process.
Deleted~~MSLD (Helium Leakage Detector)~~	~~Not Important To Safety~~	~~Not shown~~	~~Used for helium leakage testing of the MPC closure welds.~~
Deleted			
Deleted			
Temporary Shield Ring	Not Important To Safety	8.1.18	A water-filled tank that fits on the cask neutron shield around the upper forging and provides supplemental shielding to personnel performing cask loading and closure operations.
Vacuum Drying (Moisture Removal) System	Not Important To Safety	8.1.22a	Used for removal of residual moisture from the MPC following water draining.
Forced Helium Dehydration System	Not Important To Safety	8.1.22b	Used for removal of residual moisture from the MPC following water draining.
Vent and Drain RVOAs	Not Important To Safety	8.1.16	Used to access the vent and drain ports. The vent and drain RVOAs allow the vent and drain ports to be operated like valves and prevent the need to hot tap into the penetrations during unloading operation.
Deleted			
Weld Removal System	Not Important To Safety	8.3.2b	Semi-automated weld removal system used for removal of the MPC field weld to support unloading operations.

[†] Figures are representative and may not depict all configurations for all users.

Table 8.1.7
HI-STORM 100 SYSTEM INSTRUMENTATION SUMMARY FOR LOADING AND UNLOADING OPERATIONS†

Instrument	Function
Contamination Survey Instruments	Monitors fixed and non-fixed contamination levels.
Dose Rate Monitors/Survey Equipment	Monitors dose rate and contamination levels and ensures proper function of shielding. Ensures assembly debris is not inadvertently removed from the spent fuel pool during overpack removal.
Flow Rate Monitor	Monitors fluid flow rate during various loading and unloading operations.
Deleted	
~~Helium Mass Spectrometer Leak Detector (MSLD)~~*Deleted*	~~Ensures leakage rates of welds are within acceptance criteria.~~
Deleted	
Volumetric Examination Testing Rig	Used to assess the integrity of the MPC lid-to-shell weld.
Pressure Gauges	Ensures correct pressure during loading and unloading operations.
Temperature Gauges	Monitors the state of gas and water temperatures during closure and unloading operations.
Deleted	
Temperature Surface Pyrometer	For HI-STORM vent operability testing.
Vacuum Gages	Used for vacuum drying operations and to prepare an MPC evacuated sample bottle for MPC gas sampling for unloading operations.
Deleted	
Deleted	
Moisture Monitoring Instruments	Used to monitor the MPC moisture levels as part of the moisture removal system.

† All instruments require calibration. See figures at the end of this section for additional instruments, controllers and piping diagrams.

Table 8.1.8
HI-STORM 100 SYSTEM OVERPACK INSPECTION CHECKLIST

Note:
This checklist provides the basis for establishing a site-specific inspection checklist for the HI-STORM 100 overpack. Specific findings shall be brought to the attention of the appropriate site organizations for assessment, evaluation and potential corrective action prior to use.

HI-STORM 100 Overpack Lid:

1. Lid studs and nuts shall be inspected for general condition.
2. The painted surfaces shall be inspected for corrosion and chipped, cracked or blistered paint.
3. All lid surfaces shall be relatively free of dents, scratches, gouges or other damage.
4. The lid shall be inspected for the presence or availability of studs and nuts and hole plugs.
5. Lid lifting device/ holes shall be inspected for dirt and debris and thread condition.
6. Lid bolt holes shall be inspected for general condition.

HI-STORM 100 Main Body:

1. Lid bolt holes shall be inspected for dirt, debris, and thread condition.
2. Vents shall be free from obstructions.
3. Vent screens shall be available, intact, and free of holes and tears in the fabric.
4. The interior cavity shall be free of debris, litter, tools, and equipment.
5. Painted surfaces shall be inspected for corrosion, and chipped, cracked or blistered paint.
6. The nameplate shall be inspected for presence, legibility, and general condition and conformance to Quality Assurance records package.
7. Anchor hardware, if used, shall be checked for general condition.

Table 8.1.9
MPC INSPECTION CHECKLIST

Note:
This checklist provides the basis for establishing a site-specific inspection checklist for MPC. Specific findings shall be brought to the attention of the appropriate site organizations for assessment, evaluation and potential corrective action prior to use.

MPC Lid and Closure Ring:

1. The MPC lid and closure ring surfaces shall be relatively free of dents, gouges or other shipping damage.
2. The drain line shall be inspected for straightness, thread condition, and blockage.
3. Vent and Drain attachments shall be inspected for availability, thread condition operability and general condition.
4. Upper fuel spacers (if used) shall be inspected for availability and general condition. Plugs shall be available for non-used spacer locations.
5. Lower fuel spacers (if used) shall be inspected for availability and general condition.
6. Drain and vent port cover plates shall be inspected for availability and general condition.
7. Serial numbers shall be inspected for readability.

MPC Main Body:

1. All visible MPC body surfaces shall be inspected for dents, gouges or other shipping damage.
2. Fuel cell openings shall be inspected for debris, dents and general condition.
3. Lift lugs shall be inspected for general condition.
4. Verify proper MPC basket type for contents.

Table 8.1.10
HI-TRAC TRANSFER CASK INSPECTION CHECKLIST

Note:
This checklist provides the basis for establishing a site-specific inspection checklist for the HI-TRAC Transfer Cask. Specific findings shall be brought to the attention of the appropriate site organizations for assessment, evaluation and potential corrective action prior to use.

HI-TRAC Top Lid:

1. The painted surfaces shall be inspected for corrosion and chipped, cracked or blistered paint.
2. All Top Lid surfaces shall be relatively free of dents, scratches, gouges or other damage.

HI-TRAC Main Body:

1. The painted surfaces shall be inspected for corrosion, chipped, cracked or blistered paint.
2. The Top Lid bolt holes shall be inspected for dirt, debris and thread damage.
3. The Top Lid lift holes shall be inspected for thread condition.
4. Lifting trunnions shall be inspected for deformation, cracks, end plate damage, corrosion, excessive galling, damage to the locking plate, presence or availability of locking plate and end plate retention bolts.
5. Pocket trunnion, if used, recesses shall be inspected for indications of overstressing (i.e., cracks, deformation, and excessive wear).
6. Annulus inflatable seal groove shall be inspected for cleanliness, scratches, dents, gouges, sharp corners, burrs or any other condition that may damage the inflatable seal.
7. The nameplate shall be inspected for presence and general condition.
8. The neutron shield jacket shall be inspected for leaks.
9. Neutron shield jacket pressure relief valve shall be inspected for presence, and general condition.
10. The neutron shield jacket fill and drain plugs shall be inspected for presence, leaks, and general condition.
11. Bottom lid flange surface shall be clean and free of large scratches and gouges.

Table 8.1.10 (Continued)
HI-TRAC OVERPACK INSPECTION CHECKLIST

HI-TRAC Transfer Lid (Not used with HI-TRAC 125D):

1. The doors shall be inspected for smooth actuation.
2. The threads shall be inspected for general condition.
3. The bolts shall be inspected for indications of overstressing (i.e., cracks, deformation, thread damage, excessive wear) and replaced as necessary.
4. Door locking pins shall be inspected for indications of overstressing (i.e., cracks, and deformation, thread damage, excessive wear) and replaced as necessary.
5. Painted surfaces shall be inspected for corrosion and chipped, cracked or blistered paint.
6. Lifting holes shall be inspected for thread damage.

HI-TRAC Pool Lid:

1. Seal shall be inspected for cracks, breaks, cuts, excessive wear, flattening, and general condition.
2. Drain line shall be inspected for blockage and thread condition.
3. The lifting holes shall be inspected for thread damage.
4. The bolts shall be inspected for indications of overstressing (i.e., cracks and deformation, thread damage, and excessive wear).
5. Painted surfaces shall be inspected for corrosion and chipped, cracked or blistered paint.
6. Threads shall be inspected for indications of damage.

8.3 PROCEDURE FOR UNLOADING THE HI-STORM 100 SYSTEM IN THE SPENT FUEL POOL

8.3.1 Overview of HI-STORM 100 System Unloading Operations

ALARA Note:
The procedure described below uses the weld removal system to remove the welds necessary to enable the MPC lid to be removed. Users may opt to remove some or all of the welds using hand operated equipment. The decision should be based on dose rates, accessibility, degree of weld removal, and available tooling and equipment.

The HI-STORM 100 System unloading procedures describe the general actions necessary to prepare the MPC for unloading, cool the stored fuel assemblies in the MPC, flood the MPC cavity, remove the lid welds, unload the spent fuel assemblies, and recover the HI-TRAC and empty MPC. Special precautions are outlined to ensure personnel safety during the unloading operations, and to prevent the risk of MPC over pressurization and thermal shock to the stored spent fuel assemblies. Figure 8.3.1 shows a flow diagram of the HI-STORM unloading operations. Figure 8.3.2 illustrates the major HI-STORM unloading operations.

Refer to the boxes of Figure 8.3.2 for the following description. The MPC is recovered from HI-STORM either at the ISFSI or the fuel building using the same methodologies as described in Section 8.1 (Box 1). The HI-STORM lid is removed, the vent duct shield inserts are installed, the alignment device (or mating device with pool lid for HI-TRAC 125D) is positioned, and the MPC lift cleats are attached to the MPC. The exit vent screens and gamma shield cross plates are removed as necessary. MPC slings are attached to the MPC lift cleat and positioned on the MPC lid. HI-TRAC is positioned on top of HI-STORM (Box 2) and the slings are brought through the HI-TRAC top lid. The MPC is raised into HI-TRAC, the HI-TRAC doors (or mating device drawer) are closed and the locking pins are installed. If the mating device and HI-TRAC 125D are used, the pool lid is bolted to the HI-TRAC. The HI-TRAC is removed from on top of HI-STORM. If the HI-TRAC 125D is not used, the HI-TRAC is positioned in the transfer slide and the transfer lid is replaced with the pool lid (Box 3) using the same methodology as with the loading operations.

HI-TRAC and its enclosed MPC are returned to the designated preparation area and the MPC slings, MPC lift cleats and top lid are removed[1] (Box 4). The temporary shield ring is installed on the HI-TRAC upper section and filled with plant demineralized water. The HI-TRAC top lid is removed and a water flush is performed on the annulus. Water is fed into the annulus through the drain port and allowed to cool the MPC shell. After a predetermined period (based on the fuel conditions), cover the annulus and HI-TRAC top surfaces to protect them from debris produced when removing the MPC lid. The weld removal system is installed (Box 7) and the MPC vent and drain ports are accessed (Box 5). The vent RVOA is attached to the vent port and an evacuated sample bottle is connected. The vent port is slightly opened to allow the sample bottle to obtain a gas sample from inside the MPC. A gas sample is performed to assess the condition of the fuel assembly cladding. A vent line is attached to the vent port and the MPC is vented to the fuel building ventilation system or spent fuel pool as determined by the site's

~~1.~~[1] Users with the optional HI-TRAC Lid Spacer shall modify steps in their procedures to install and remove the spacer together with top lid.

radiation protection personnel. The MPC is cooled using the cool-down system to reduce the MPC internal temperature to allow water flooding (Box 6). The cool-down process gradually reduces the cladding temperature to a point where the MPC may be flooded with water without thermally shocking the fuel assemblies or over-pressurizing the MPC from the formation of steam ~~(See Technical Specification LCO 3.1.3)~~. Following the fuel cool-down, the MPC is filled with water *(borated as required)*. The weld removal system then removes the MPC lid-to-shell weld. The weld removal system is removed with the MPC lid left in place (Box 7).

The top surfaces of the HI-TRAC and MPC are cleared of metal shavings. The inflatable annulus seal is installed and pressurized. The MPC lid is rigged to the lift yoke or lid retention system and the lift yoke is engaged to HI-TRAC lifting trunnions. If weight limitations require, the neutron shield jacket is drained of water. HI-TRAC is placed in the spent fuel pool and the MPC lid is removed (Boxes 8 and 9). All fuel assemblies are returned to the spent fuel storage racks and the MPC fuel cells are vacuumed to remove any assembly debris and crud (Box 10). HI-TRAC and MPC are returned to the designated preparation area (Box 11) where the MPC water is pumped back into the spent fuel pool or liquid radwaste facility. The annulus water is drained and the MPC and overpack are decontaminated (Box 12 and 13).

8.3.2 HI-STORM Recovery from Storage

Note:
The MPC transfer may be performed using the MPC downloader or the overhead crane.

1. Recover the MPC from HI-STORM as follows:

   a. If necessary, perform a transport route walkdown to ensure that the cask transport conditions are met. ~~See Technical Specifications for the on-site lifting requirements.~~

   b. Transfer HI-STORM to the fuel building or site designated location for the MPC transfer.

   c. Position HI-STORM under the lifting device.

   d. Remove the HI-STORM lid nuts, washers and studs.

   e. Remove the HI-STORM lid lifting hole plugs and install the lid lifting sling. See Figure 8.1.27.

Note:
The specific sequence for vent screen, temperature element, and gamma shield cross plate removal may vary based on the mode(s) or transport.

   f. Remove the HI-STORM exit vent screens, temperature elements and gamma shield cross plates. See Figure 8.1.34a and b.

Warning:
Unless the lift is single-failure proof (or equivalent safety factor) for the HI-STORM lid, the lid shall be kept less than 2 feet above the top surface of the overpack. This is performed to protect the MPC lid from a potential HI-STORM 100 lid drop.

   g. Remove the HI-STORM lid. See Figure 8.1.27.

h. Install the alignment device (or mating device with pool lid for HI-TRAC 125D) and vent duct shield inserts (HI-STORM 100 only). See Figure 8.1.30.

i. Deleted.

j. Remove the MPC lift cleat hole plugs and install the MPC lift cleats and MPC slings to the MPC lid. See Table 8.1.5 for torque requirements.

k. If necessary, install the top lid on HI-TRAC. See Figure 8.1.9 for rigging. See Table 8.1.5 for torque requirements.

l. Deleted.

2. If necessary, configure HI-TRAC with the transfer lid (Not required for HI-TRAC 125D):

ALARA Warning:
The bottom lid replacement as described below may only be performed on an empty (i.e., no MPC) HI-TRAC.

a. Position HI-TRAC vertically adjacent to the transfer lid. See Section 8.1.2.

b. Remove the bottom lid bolts and plates and store them temporarily.

c. Raise the empty HI-TRAC and position it on top of the transfer lid.

d. Inspect the pool lid bolts for general condition. Replace worn or damaged bolts with new bolts.

e. Install the transfer lid bolts. See Table 8.1.5 for torque requirements.

3. At the site's discretion, perform a HI-TRAC receipt inspection and cleanliness inspection in accordance with a site-specific inspection checklist.

Note:
If the HI-TRAC is expected to be operated in an environment below 32 °F, the water jacket shall be filled with an ethylene glycol solution (25% ethylene glycol). Otherwise, the jacket shall be filled with demineralized water.

4. If previously drained, fill the neutron shield jacket with plant demineralized water or an ethylene glycol solution (25% ethylene glycol) as necessary. Ensure that the fill and drain plugs are installed.

5. Engage the lift yoke to the HI-TRAC lifting trunnions.

6. Align HI-TRAC over HI-STORM and mate the overpacks. See Figure 8.1.31.

7. If necessary, install the MPC downloader.

8. Remove the transfer lid (or mating device) locking pins and open the doors (mating device drawer).

9. At the user's discretion, install trim plates to cover the gap above and below the door (drawer for 125D). The trim plates may be secured using hand clamps or any other method deemed suitable by the user. See Figure 8.1.33.

10. Attach the ends of the MPC sling to the lifting device or MPC downloader. See Figure 8.1.32.

ALARA Warning:
If trim plates are not used, personnel should remain clear of the immediate door area during MPC downloading since there may be some radiation streaming during MPC raising and lowering operations.
Caution:
Limitations for the handling of the loaded MPC in HI-TRAC are evaluated and established on a canister basis to ensure that acceptable cladding temperatures are not exceeded. Refer to FSAR Section 4.5 for specific time limits for transporting a loaded MPC in the HI-TRAC transfer CASK based on MPC heat loads.

11. Raise the MPC into HI-TRAC.

12. Verify the MPC is in the full-up position.

13. Close the HI-TRAC doors (or mating device drawer) and install the door locking pins.

14. For the HI-TRAC 125D, bolt the pool lid to the HI-TRAC. See Table 8.1.5 for torque requirements.

15. Lower the MPC onto the transfer lid doors (or pool lid for 125D).

16. Disconnect the slings from the MPC lift cleats.

17. If necessary, remove the MPC downloader from the top of HI-TRAC.

18. Remove HI-TRAC from the top of HI-STORM.

8.3.3 Preparation for Unloading:

1. Replace the pool lid with the transfer lid as follows (Not required for HI-TRAC 125D):

    a. Lower the lift yoke and attach the MPC slings between the lift cleats and the lift yoke. See Figure 8.1.25.

    b. Engage the lift yoke to the HI-TRAC lifting trunnions.

    c. Deleted.

    d. Raise HI-TRAC and position the transfer lid approximately one inch above the transfer step. See Figure 8.1.26.

    e. Raise the transfer slide carriage so the transfer carriage is supporting the transfer lid bottom. Remove the transfer lid bolts and store them temporarily.

ALARA Warning:
Clear all personnel away from the immediate operations area. The transfer slide carriage and jacks are remotely operated. The carriage has fine adjustment features to allow precise positioning of the lids.

f. Lower the transfer carriage and position the pool lid under HI-TRAC.

g. Raise the transfer slide carriage to place the pool lid against the HI-TRAC bottom lid bolting flange.

h. Inspect the bottom lid bolts for general condition. Replace worn or damaged bolts with new bolts.

i. Install the pool lid bolts. See Table 8.1.5 for torque requirements.

j. Raise and remove the HI-TRAC from the transfer slide.

k. Disconnect the MPC slings and lift cleats.

l. Deleted.

m. Deleted.

2. Place HI-TRAC in the designated preparation area.

Warning:
Unless the lift is single-failure proof (or equivalent safety factor) the HI-TRAC top lid, the top lid shall be kept less than 2 feet above the top surface of the MPC. This is performed to protect the MPC lid from a potential lid drop.

3. Prepare for MPC cool-down as follows:

a. Remove the top lid bolts and remove HI-TRAC top lid. See Figure 8.1.9 for rigging.

Warning:
At the start of annulus filling, the annulus fill water may flash to steam due to high MPC shell temperatures. Users may select the location and means of performing the annulus flush. Users may also elect the source of water and method for collecting the water flowing from the annulus. Water addition should be preformed in a slow and controlled manner until water steam generation has ceased. Water flush should be performed for a minimum of 33 hours at a flow rate of 10 GPM or as specified for the particular heat load of the MPC.

b. Perform annulus flush by injecting water into the HI-TRAC drain port and allowing the water to cool the MPC shell and lid.

4. Set the annulus water level to approximately 4 inches below the top of the MPC shell and install the annulus shield. Cover the annulus and HI-TRAC top surfaces to protect them from debris produced when removing the MPC lid.

5. Access the MPC as follows:

ALARA Note:
The following procedures describe weld removal using a machine tool head. Other methods may also be used. The metal shavings may need to be periodically vacuumed.

ALARA Warning:
Weld removal may create an airborne radiation condition. Weld removal must be performed under the direction of the user's Radiation Protection organization.

a. Install bolt plugs and/or waterproof tape from HI-TRAC top bolt holes.

b. Using the marked locations of the vent and drain ports, core drill the closure ring and vent and drain port cover plates.

6. Remove the closure ring section and the vent and drain port cover plates.

ALARA Note:
The MPC vent and drain ports are equipped with metal-to-metal seals to minimize leakage and withstand the long-term effects of temperature and radiation. The vent and drain port design prevents the need to hot tap into the penetrations during unloading operation and eliminate the risk of a pressurized release of gas from the MPC.

7. Take an MPC gas sample as follows:

Note:
Users may select alternate methods of obtaining a gas sample.

a. Attach the RVOAs (See Figure 8.1.16).

b. Attach a sample bottle to the vent port RVOA as shown on Figure 8.3.3.

c. Using the vacuum drying system, evacuated the RVOA and Sample Bottle.

d. Slowly open the vent port cap using the RVOA and gather a gas sample from the MPC internal atmosphere.

e. Close the vent port cap and disconnect the sample bottle.

ALARA Note:
The gas sample analysis is performed to determine the condition of the fuel cladding in the MPC. The gas sample may indicate that fuel with damaged cladding is present in the MPC. The results of the gas sample test may affect personnel protection and how the gas is processed during MPC depressurization.

f. Turn the sample bottle over to the site's Radiation Protection or Chemistry Department for analysis.

g. Remove the drain port cover plate weld and remove the cover plate.

8. Fill the MPC cavity with water as follows:

a. Configure the cool-down system as shown on Figure 8.3.4.

b. Verify that the helium gas pressure regulator is set to the appropriate pressure.

c. Open the helium gas supply valve to purge the gas lines of air.

d. Deleted.

e. If necessary, slowly open the helium supply valve and increase the Cool-Down System pressure. Close the helium supply valve.

f. Start the gas coolers.

g. Open the vent and drain port caps using the RVOAs.

h. Start the blower and monitor the gas exit temperature. Continue the fuel cool-down operations until the gas exit temperature meets the requirements~~ of the Technical Specification LCO 3.1.3~~.

Note:

Water filling should commence immediately at the completion of fuel cool-down operations to prevent fuel assembly heat-up. Prepare the water fill line and the vent line in advance of water filling.

i. Prepare the MPC fill and vent lines as shown on Figure 8.1.20. Route the vent port line several feet below the spent fuel pool surface or to the radwaste gas facility. Turn off the blower and disconnect the gas lines to the vent and drain port RVOAs. Attach the vent line to the MPC vent port and slowly open the vent line valve to depressurize the MPC.

Note:

When unloading MPCs requiring soluble boron, the boron concentration of the water shall be checked in accordance with LCO 3.3.1 before and during operations with fuel and water in the MPC.

j. Attach the water fill line to the MPC drain port and slowly open the water supply valve and establish a pressure less than 90 psi *(Refer to Tables 2.1.14 and 2.1.16 for boron concentration requirements)*. Fill the MPC until bubbling from the vent line has terminated. Close the water supply valve on completion.

Caution:

Oxidation of Boral panels ~~and aluminum components~~ contained in the MPC may create hydrogen gas while the MPC is filled with water. Appropriate monitoring for combustible gas concentrations shall be performed prior to, and during MPC lid cutting operations. ~~It is also recommended for defense-in-depth that the space below the MPC lid be exhausted prior to, and during MPC lid cutting operations to provide additional assurance that explosive gas mixtures will not develop in this space.~~ *The space below the MPC lid shall be exhausted or purged with inert gas prior to, and during MPC cutting operations to provide additional assurance that flammable gas concentrations will not develop in this space*

k. *Connect a combustible gas monitor to the MPC vent port and check for combustible gas concentrations prior to and periodically during weld removal activities. Purge or evacuate the gas space under the lid as necessary*~~Disconnect both lines from the drain and vent ports and, if desired, install an exhaust line to the vent port to evacuate the head space. Perform combustible gas monitoring to ensure there is no combustible mixture present in the exhaust gases~~.

l. Remove the MPC lid-to-shell weld using the weld removal system. See Figure 8.1.9 for rigging.

m. Vacuum the top surfaces of the MPC and HI-TRAC to remove any metal shavings.

9. Install the inflatable annulus seal as follows:

Caution:
Do not use any sharp tools or instruments to install the inflatable seal.

a. Remove the annulus shield.

b. Manually insert the inflatable seal around the MPC. See Figure 8.1.13.

c. Ensure that the seal is uniformly positioned in the annulus area.

d. Inflate the seal

e. Visually inspect the seal to ensure that it is properly seated in the annulus. Deflate, adjust and inflate the seal as necessary.

10. Place HI-TRAC in the spent fuel pool as follows:

a. If necessary for plant weight limitations, drain the water from the neutron shield jacket.

b. Engage the lift yoke to HI-TRAC lifting trunnions, remove the MPC lid lifting hole plugs and attach the MPC lid slings or lid retention system to the MPC lid.

c. If the lid retention system is used, inspect the lid bolts for general condition. Replace worn or damaged bolts with new bolts.

d. Install the lid retention system bolts if the lid retention system is used.

ALARA Note:
The optional Annulus Overpressure System is used to provide further protection against MPC external shell contamination during in-pool operations.

e. If used, fill the annulus overpressure system lines and reservoir with demineralized water and close the reservoir valve. Attach the annulus overpressure system to the HI-TRAC. See Figure 8.1.14.

f. Position HI-TRAC over the cask loading area with the basket aligned to the orientation of the spent fuel racks.

ALARA Note:
Wetting the components that enter the spent fuel pool may reduce the amount of decontamination work to be performed later.

g. Wet the surfaces of HI-TRAC and lift yoke with plant demineralized water while slowly lowering HI-TRAC into the spent fuel pool.

h. When the top of the HI-TRAC reaches the elevation of the reservoir, open the annulus overpressure system reservoir valve. Maintain the reservoir water level at approximately 3/4 full the entire time the cask is in the spent fuel pool.

i. If the lid retention system is used, remove the lid retention bolts when the top of HI-TRAC is accessible from the operating floor.

j. Place HI-TRAC on the floor of the cask loading area and disengage the lift yoke. Visually verify that the lift yoke is fully disengaged.

k. Apply slight tension to the lift yoke and visually verify proper disengagement of the lift yoke from the trunnions.

l. Remove the lift yoke, MPC lid and drain line from the pool in accordance with directions from the site's Radiation Protection personnel. Spray the equipment with demineralized water as they are removed from the pool.

m. Disconnect the drain line from the MPC lid.

n. Store the MPC lid components in an approved location. Disengage the lift yoke from MPC lid. Remove any upper fuel spacers using the same process as was used in the installation.

o. Disconnect the lid retention system if used.

8.3.4 MPC Unloading

1. Remove the spent fuel assemblies from the MPC using applicable site procedures.
2. Vacuum the cells of the MPC to remove any debris or corrosion products.
3. Inspect the open cells for presence of any remaining items. Remove them as appropriate.

8.3.5 Post-Unloading Operations

1. Remove HI-TRAC and the unloaded MPC from the spent fuel pool as follows:

a. Engage the lift yoke to the top trunnions.

b. Apply slight tension to the lift yoke and visually verify proper engagement of the lift yoke to the trunnions.

c. Raise HI-TRAC until HI-TRAC flange is at the surface of the spent fuel pool.

ALARA Warning:
Activated debris may have settled on the top face of HI-TRAC during fuel unloading.

d. Measure the dose rates at the top of HI-TRAC in accordance with plant radiological procedures and flush or wash the top surfaces to remove any highly-radioactive particles.

e. Raise the top of HI-TRAC and MPC to the level of the spent fuel pool deck.

f. Close the annulus overpressure system reservoir valve.

g. Using a water pump, lower the water level in the MPC approximately 12 inches to prevent splashing during cask movement.

ALARA Note:
To reduce contamination of HI-TRAC, the surfaces of HI-TRAC and lift yoke should be kept wet until decontamination can begin.

h. Remove HI-TRAC from the spent fuel pool while spraying the surfaces with plant demineralized water.

i. Disconnect the annulus overpressure system from the HI-TRAC via the quick disconnect.

j. Place HI-TRAC in the designated preparation area.

k. Disengage the lift yoke.

l. Perform decontamination on HI-TRAC and the lift yoke.

2. Carefully decontaminate the area above the inflatable seal. Deflate, remove, and store the seal in an approved plant storage location.

3. Using a water pump, pump the remaining water in the MPC to the spent fuel pool or liquid radwaste system.

4. Drain the water in the annulus area by connecting the drain line to the HI-TRAC drain connector.

5. Remove the MPC from HI-TRAC and decontaminate the MPC as necessary.

6. Decontaminate HI-TRAC.

7. Remove the bolt plugs and/or waterproof tape from HI-TRAC top bolt holes.

8. Return any HI-STORM 100 equipment to storage as necessary.

9.1 ACCEPTANCE CRITERIA

This section provides the workmanship inspections and acceptance tests to be performed on the HI-STORM 100 System prior to and during ~~first~~ loading of the system. These inspections and tests provide assurance that the HI-STORM 100 System has been fabricated, assembled, inspected, tested, and accepted for use under the conditions specified in this FSAR and the Certificate of Compliance issued by the NRC in accordance with the requirements of 10CFR72 [9.0.1].

~~These inspections and tests are also intended to demonstrate that the operation of the HI-STORM 100 System complies with the applicable regulatory requirements and the Technical Specifications contained in Appendix A to CoC 72-1014. Noncompliances encountered during the required inspections and tests shall be corrected or dispositioned to bring the item into compliance with this FSAR.~~ Identification and resolution of noncompliances shall be performed in accordance with the Holtec International Quality Assurance Program as described in Chapter 13 of this FSAR, or the licensee's NRC-approved Quality Assurance Program.

The testing and inspection acceptance criteria applicable to the MPCs, the HI-STORM 100 overpack, and the 100-ton HI-TRAC and 125-ton HI-TRAC transfer casks are listed in Tables 9.1.1, 9.1.2, and 9.1.3, respectively, and discussed in more detail in the sections that follow. Chapters 8 and 12 provide ~~details on~~ operating *guidance*~~procedures~~ and the bases for the Technical Specifications, respectively. These inspections and tests are intended to demonstrate that the HI-STORM 100 System has been fabricated, assembled, and examined in accordance with the design criteria contained in Chapter 2 of this FSAR.

This section summarizes the test program required for the HI-STORM 100 System.

9.1.1 Fabrication and Nondestructive Examination (NDE)

The design, fabrication, inspection, and testing of the HI-STORM 100 System is performed in accordance with the applicable codes and standards specified in Tables 2.2.6 and 2.2.7 and on the Design Drawings. Additional details on specific codes used are provided below.

The following fabrication controls and required inspections shall be performed on the HI-STORM 100 System, including the MPCs, overpacks, and HI-TRAC transfer casks, in order to assure compliance with this FSAR and the Certificate of Compliance.

1. Materials of construction specified for the HI-STORM 100 System are identified in the drawings in Chapter 1 and shall be procured with certification and supporting documentation as required by ASME Code [9.1.1] Section II (when applicable); the requirements of ASME Section III (when applicable); Holtec procurement specifications; and 10CFR72, Subpart G. Materials and components shall be receipt inspected for visual and dimensional acceptability, material conformance to specification requirements, and traceability markings, as applicable. Controls shall be in place to assure material traceability is maintained throughout fabrication. Materials

for the confinement boundary (MPC baseplate, lid, closure ring, port cover plates and shell) shall also be inspected per the requirements of ASME Section III, Article NB-2500.

2. The MPC confinement boundary shall be fabricated and inspected in accordance with ASME Code, Section III, Subsection NB, with ~~exceptions~~ *alternatives* as noted below. The MPC basket and basket supports shall be fabricated and inspected in accordance with ASME Code, Section III, Subsection NG, with ~~exceptions~~ *alternatives* as noted below. Metal components of the HI-TRAC transfer cask and the HI-STORM overpack, as applicable, shall be fabricated and inspected in accordance with ASME Code, Section III, Subsection NF, Class 3 or AWS D1.1, as shown on the design drawings, with ~~exceptions~~ *alternatives* as noted below.

   NOTE: ~~Exceptions~~—*NRC-approved alternatives* to these Code requirements are ~~provided~~—*discussed* in FSAR *Section 2.2.4.*~~Chapter 2 and in Table 3-1 of Appendix B to CoC 72-1014.~~

3. ASME Code welding shall be performed using welders and weld procedures that have been qualified in accordance with ASME Code Section IX and the applicable ASME Section III Subsections (e.g., NB, NG, or NF, as applicable to the SSC). AWS code welding may be performed using welders and weld procedures that have been qualified in accordance with applicable AWS requirements or in accordance with ASME Code Section IX

4. Welds shall be visually examined in accordance with ASME Code, Section V, Article 9 with acceptance criteria per ASME Code, Section III, Subsection NF, Article NF-5360, except the MPC fuel basket cell plate-to-cell plate welds and fuel basket support-to-canister welds which shall have acceptance criteria to ASME Code Section III, Subsection NG, Article NG-5360, (as modified by the design drawings). Table 9.1.4 identifies additional nondestructive examination (NDE) requirements to be performed on specific welds, and the applicable codes and acceptance criteria to be used in order to meet the inspection requirements of the applicable ASME Code, Section III. Acceptance criteria for NDE shall be in accordance with the applicable Code for which the item was fabricated. These additional NDE criteria are also specified on the design drawings for the specific welds. Weld inspections shall be detailed in a weld inspection plan which shall identify the weld and the examination requirements, the sequence of examination, and the acceptance criteria. The inspection plan shall be reviewed and approved by Holtec in accordance with its QA program. NDE inspections shall be performed in accordance with written and approved procedures by personnel qualified in accordance with SNT-TC-1A [9.1.2] or other site-specific, NRC-approved program for personnel qualification.

5. Machined surfaces of the metal components of the HI-STORM 100 System shall be visually examined in accordance with ASME Section V, Article 9, to verify they are free of cracks and pin holes.

6. ASME Code welds requiring weld repair shall be repaired in accordance with the requirements of the ASME Code, Section III, Article NB-4450, NG-4450, or NF-4450, as applicable to the SSC, and examined after repair in the same manner as the original weld.

7. Base metal repairs shall be performed and examined in accordance with the applicable fabrication Code.

8. Grinding and machining operations on the MPC confinement boundary shall be controlled through written and approved procedures and quality assurance oversight to ensure grinding and machining operations do not reduce base metal wall thicknesses of the confinement boundary beyond that allowed per the design drawings. The thicknesses of base metals shall be ultrasonically tested, as necessary, in accordance with written and approved procedures to verify base metal thickness meets Design Drawing requirements. A nonconformance shall be written for areas found to be below allowable base metal thickness and shall be evaluated and repaired per the applicable ASME Code, Subsection NB requirements.

9. Dimensional inspections of the HI-STORM 100 System shall be performed in accordance with written and approved procedures in order to verify compliance to design drawings and fit-up of individual components. All dimensional inspections and functional fit-up tests shall be documented.

10. Required inspections shall be documented. The inspection documentation shall become part of the final quality documentation package.

11. The HI-STORM 100 System shall be inspected for cleanliness and proper packaging for shipping in accordance with written and approved procedures.

12. Each cask shall be durably marked with the appropriate model number, a unique identification number, and its empty weight per 10CFR72.236(k) at the completion of the acceptance test program.

13. A documentation package shall be prepared and maintained during fabrication of each HI-STORM 100 System to include detailed records and evidence that the required inspections and tests have been performed. The completed documentation package shall be reviewed to verify that the HI-STORM 100 System or component has been properly fabricated and inspected in accordance with the design and Code construction requirements. The documentation package shall include, but not be limited to:

- Completed Shop Weld Records
- Inspection Records
- Nonconformance Reports
- Material Test Reports
- NDE Reports
- Dimensional Inspection Report

9.1.1.1 MPC Lid-to-Shell Weld Volumetric Inspection

1. The MPC lid-to-shell (LTS) weld shall be volumetrically or multi-layer liquid penetrant (PT) examined following completion of welding. If volumetric examination is used, the ultrasonic testing (UT) method shall be employed. Ultrasonic techniques (including, as appropriate, Time-of-Flight Diffraction, Focussed Phased Array, and conventional pulse-echo) shall be supplemented, as necessary, to ensure substantially complete coverage of the examination volume.

2. If volumetric examination is used, then a PT examination of the root and final pass of the LTS weld shall also be performed and unacceptable indications shall be documented, repaired and re-examined.

3. If volumetric examination is not used, a multi-layer PT examination shall be employed. The multi-layer PT must, at a minimum, include the root and final weld layers and one intermediate PT after each approximately 3/8 inch weld depth has been completed. The 3/8 inch weld depth corresponds to the maximum allowable flaw size determined in Holtec Position Paper DS-213 [9.1.6].

4. It is recognized that welding of the LTS weld may result in indications in the root pass that are not detected by the root pass PT. The overall minimum thickness of the LTS weld has been increased by 0.125 inch such that it is not necessary to take structural credit for the root pass of the weld (actual weld to be a minimum of 0.75 inch). A 0.625-inch J-groove weld was assumed in structural analyses in Chapter 3.

5. For either UT or PT, the maximum detectable flaw size must be demonstrated to be less than the critical flaw size. The critical flaw size must be determined in accordance with ASME Section XI methods. The critical flaw size shall not cause the primary stress limits of NB-3000 to be exceeded. The inspection ~~process~~*results*,

including *relevant* findings (indications) shall be made a permanent part of the cask user's records by video, photographic, or other means which provide an equivalent retrievable record of weld integrity. The video or photographic records should be taken during the final interpretation period described in ASME Section V, Article 6, T-676. The inspection of the weld shall be performed by qualified personnel and shall meet the acceptance requirements of ASME Section III, NB-5350 for PT and NB-5332 for UT.

6. Evaluation of any indications shall include consideration of any active flaw mechanisms. However, cyclic loading on the LTS weld is not significant, so fatigue is not a factor. The LTS weld is protected from the external environment by the closure ring and the root of the LTS weld is dry and inert (He atmosphere), so stress corrosion cracking is not a concern for the LTS weld.

7. The volumetric or multi-layer PT examination of the LTS weld, in conjunction with other examinations *and tests* performed on this weld (PT of root and final layer, *and* ~~hydrostatic~~ *pressure* test*)*~~,, and a helium leakage test,~~ the use of ASME Section III acceptance criteria, and the additional weld material added to account for potential defects in the root pass of the weld, in total, provide reasonable assurance that the LTS weld is sound and will perform its design function under all loading conditions. The volumetric (or multi-layer PT) examination and evaluation of indications provides reasonable assurance that leakage of the weld or structural failure under the design basis normal, off-normal, and accident loading conditions will not occur.

9.1.2 Structural and Pressure Tests

9.1.2.1 Lifting Trunnions

Two trunnions (located near the top of the HI-TRAC transfer cask) are provided for vertical lifting and handling. The trunnions are designed in accordance with ANSI N14.6 [9.1.3] using a high-strength and high-ductility material (see Chapter 1). The trunnions contain no welded components. The maximum design lifting load of 250,000 pounds for the HI-TRAC 125 and HI-TRAC 125D and 200,000 pounds for the HI-TRAC 100 will occur during the removal of the HI-TRAC from the spent fuel pool after the MPC has been loaded, flooded with water, and the MPC lid is installed. The high-material ductility, absence of materials vulnerable to brittle fracture, large stress margins, and a carefully engineered design to eliminate local stress risers in the highly-stressed regions (during the lift operations) ensure that the lifting trunnions will work reliably. However, pursuant to the defense-in-depth approach of NUREG-0612 [9.1.4], the acceptance criteria for the lifting trunnions must be established in conjunction with other considerations applicable to heavy load handling.

Section 5 of NUREG-0612 calls for measures to "provide an adequate defense-in-depth for handling of heavy loads...". The NUREG-0612 guidelines cite four major causes of load handling accidents, of which rigging failure (including trunnion failure) is one:

i. operator errors
ii. rigging failure
iii. lack of adequate inspection
iv. inadequate procedures

The cask loading and handling operations program shall ensure maximum emphasis to mitigate the potential load drop accidents by implementing measures to eliminate shortcomings in all aspects of the operation including the four aforementioned areas.

In order to ensure that the lifting trunnions do not have any hidden material flaws, the trunnions shall be tested at 300% of the maximum design (service) lifting load. The load (750,000 lbs for the HI-TRAC 125 and HI-TRAC 125D and 600,000 lbs for the HI-TRAC 100) shall be applied for a minimum of 10 minutes. The accessible parts of the trunnions (areas outside the HI-TRAC cask), and the adjacent HI-TRAC cask trunnion attachment area shall then be visually examined to verify no deformation, distortion, or cracking occurred. Any evidence of deformation, distortion or cracking of the trunnion or adjacent HI-TRAC cask trunnion attachment areas shall require replacement of the trunnion and/or repair of the HI-TRAC cask. Following any replacements and/or repair, the load testing shall be performed and the components re-examined in accordance with the original procedure and acceptance criteria. Testing shall be performed in accordance with written and approved procedures. Certified material test reports verifying trunnion material mechanical properties meet ASME Code Section II requirements will provide further verification of the trunnion load capabilities. Test results shall be documented. The documentation shall become part of the final quality documentation package.

The acceptance testing of the trunnions in the manner described above will provide adequate assurance against handling accidents.

9.1.2.2 ~~Hydrostatic~~ *Pressure* Testing

9.1.2.2.1 HI-TRAC Transfer Cask Water Jacket

The 125-ton (including HI-TRAC 125 and HI-TRAC 125D) and 100-ton HI-TRAC transfer cask water jackets shall be hydrostatically tested to 75 psig +3,-0 psig, and 71 psig +3, -0 psig, respectively, in accordance with written and approved procedures. The water jacket fill port will be used for filling the cavity with water and the vent port for venting the cavity. The approved test procedure shall clearly define the test equipment arrangement.

The hydrostatic test shall be performed after the water jacket has been welded together. The test pressure gage installed on the water jacket shall have an upper limit of approximately twice that of the test pressure. The hydrostatic test pressure shall be maintained for ten minutes. During this time

period, the pressure gage shall not fall below the applicable minimum test pressure. At the end of ten minutes, and while the pressure is being maintained at the minimum pressure, weld joints shall be visually examined for leakage. If a leak is discovered, the cavity shall be emptied and an examination to determine the cause of the leakage shall be made. Repairs and retest shall be performed until the hydrostatic test criteria are met.

After completion of the hydrostatic testing, the water jacket exterior surfaces shall be visually examined for cracking or deformation. Evidence of cracking or deformation shall be cause for rejection, or repair and retest, as applicable. Liquid penetrant (PT) or magnetic particle (MT) examination of accessible welds shall be performed in accordance with ASME Code, Section V, Articles 6 and 7, respectively, with acceptance criteria per ASME Code, Section III, Subsection NF, Articles NF-5350 and NF-5340, respectively. Unacceptable areas shall require repair and re-examination per the applicable ASME Code. The HI-TRAC water jacket hydrostatic test shall be repeated until all examinations are found to be acceptable.

If a hydrostatic retest is required and fails, a nonconformance report shall be issued and a root cause evaluation and appropriate corrective actions taken before further repairs and retests are performed.

Test results shall be documented. The documentation shall become part of the final quality documentation package.

9.1.2.2.2 MPC Confinement Boundary

~~Hydrostatic~~ *Pressure* testing *(hydrostatic or pneumatic)* of the MPC confinement boundary shall be performed in accordance with the requirements of the ASME Code Section III, Subsection NB, Article NB-6000 *and applicable sub-articles*, when field welding of the MPC lid-to-shell weld is completed. *If hydrostatic testing is used, the MPC shall be pressure tested to 125% of design pressure. If pneumatic testing is used, the MPC shall be pressure tested to 120% of design pressure.* ~~The hydrostatic pressure for the test is 125 +5, 0 psig, which is 125% of the design pressure of 100 psig.~~ The MPC vent and drain ports will be used for pressurizing the MPC cavity. The loading procedures in FSAR Chapter 8 define the test equipment arrangement. The calibrated test pressure gage installed on the MPC confinement boundary shall have an upper limit of approximately twice that of the test pressure. Following completion of the ~~10-minute~~*required* hold period at the ~~hydrostatic~~ test pressure, ~~and while maintaining a minimum test pressure of 125 psig,~~ the surface of the MPC lid-to-shell weld shall be ~~visually examined for leakage and then~~ re-examined by liquid penetrant examination in accordance with ASME Code, Section III, Subsection NB, Article NB-5350 acceptance criteria. Any evidence of cracking or deformation shall be cause for rejection, or repair and retest, as applicable. The performance and sequence of the test is described in FSAR Section 8.1 (loading procedures).

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

4450.~~Subsection NB, NB-4450.~~

The MPC confinement boundary ~~hydrostatic~~*pressure* test shall be repeated until all ~~visual and liquid penetrant~~*required* examinations are found to be acceptable. ~~in accordance with the acceptance criteria.~~ Test results shall be documented and maintained as part of the loaded MPC quality documentation package.

9.1.2.3 Materials Testing

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

Materials of the HI-TRAC transfer cask and HI-STORM overpack, as required, shall be Charpy V-notch tested in accordance with ASME Section IIA and/or ASME Section III, Subsection NF, Articles NF-2300, and NF-2430. The materials to be tested include the components identified in Table 3.1.18 and applicable weld materials. Table 3.1.18 provides the test temperatures and test acceptance criteria to be used when performing the material testing specified above.

The concrete utilized in the construction of the HI-STORM overpack shall be mixed, poured, and tested as described in FSAR Appendix 1.D in accordance with written and approved procedures. Testing shall verify the composition, compressive strength, and density meet design requirements.

Concrete testing shall be performed for each lot of concrete. Concrete testing shall comply with ACI 349, as described in Table 1.D.2. Test specimens shall be in accordance with ASTM C39.

Test results shall be documented and become part of the final quality documentation package.

9.1.3 Leakage Testing

Leakage testing shall be performed in accordance with the requirements of ANSI N14.5 [9.1.5]. Testing shall be performed in accordance with written and approved procedures.

At completion of welding the MPC shell to the baseplate, an MPC confinement boundary weld helium leakage test shall be performed using a helium mass spectrometer leak detector (MSLD). A temporary test closure lid is used in order to provide a sealed MPC. The confinement boundary welds shall have indicated helium leakage rates less than or equal to 5×10^{-6} atm cm^3/s (helium). If a leakage rate exceeding the acceptance criterion is detected, then the area of leakage shall be determined and the area repaired per ASME Code Section III, Subsection NB, Article NB-4450 requirements. Re-testing shall be performed until the leakage rate acceptance criteria is met.

If failure of the leakage rate retest occurs after initial repairs are completed, a nonconformance report shall be issued, and a root cause evaluation and appropriate corrective actions taken before further repairs and retest are performed.

Leakage testing of the field-welded MPC lid-to-shell weld, *vent and drain port cover plate welds, and closure ring welds is not required.* ~~shall be performed following the successful completion of the MPC hydrostatic test performed per Section 9.1.2.2.2. Leakage testing of the vent and drain port cover plate welds shall be performed after welding of the cover plates and subsequent NDE. The description and procedures for these field leakage tests are provided in FSAR Section 8.1, and the acceptance criteria are defined in the Technical Specifications in Appendix A to CoC 72-1014.~~

Leak testing results for the MPC shall be documented and shall become part of the quality record documentation package.

9.1.4 Component Tests

9.1.4.1 Valves, Rupture Discs, and Fluid Transport Devices

There are no fluid transport devices or rupture discs associated with the HI-STORM 100 System. The only valve-like components in the HI-STORM 100 System are the specially designed caps installed in the MPC lid for the drain and vent ports. These caps are recessed inside the MPC lid and covered by the fully-welded vent and drain port cover plates. No credit is taken for the caps' ability to confine helium or radioactivity. After completion of drying and backfill operations, the drain and vent port cover plates are welded in place on the MPC lid and are ~~leak-tested~~*liquid penetrant examined* to verify the MPC confinement boundary.

There are two pressure relief valves installed in the upper ledge surface of the HI-TRAC transfer cask water jacket. These pressure relief valves are ~~is~~ provided for venting of the neutron shield jacket fluid under hypothetical fire accident conditions in which the design pressure of the water jacket may be exceeded. The pressure relief valves shall relieve at 60 psig and 65 psig.

9.1.4.2 Seals and Gaskets

There are no confinement seals or gaskets included in the HI-STORM 100 System.

9.1.5 Shielding Integrity

The HI-STORM overpack and MPC have two designed shields for neutron and gamma ray attenuation. The HI-STORM overpack concrete provides both neutron and gamma shielding. Additional neutron shielding is provided by the encased ~~Boral~~ neutron absorber attached to the fuel basket cell surfaces inside the MPCs. The overpack's inner and outer steel shells, and the steel shield shell[†], provide radial gamma shielding. Concrete and steel plates provide axial neutron and gamma shielding. A concrete ring attached to the top of the overpack lid provides additional gamma and

[†] The shield shell design feature was deleted in June, 2001 after overpack serial number 7 was fabricated. Those overpacks without the shield shell are required to have a higher concrete density in the overpack body to provide compensatory shielding. See Table 1.D.1.

neutron shielding in the axial direction. Steel gamma shield cross plates, installed in the overpack air inlet and outlet vents, provide additional shielding for radiation through the vent openings.

The HI-TRAC transfer cask uses three different materials for primary shielding. All three HI-TRAC transfer cask designs include a radial steel-lead-steel shield and a steel-lead-steel pool lid design. The top lid in the HI-TRAC 125 and HI-TRAC 125D designs includes Holtite neutron shielding inside a steel enclosure. The HI-TRAC 100 top lid includes only steel shielding. The HI-TRAC 125 transfer lid includes steel, lead, and Holtite, while the HI-TRAC 100 includes only steel and lead. The HI-TRAC 125D design does not include a transfer lid. The water jacket, included in all transfer cask designs, provides radial neutron shielding. Testing requirements for the shielding items are described below.

9.1.5.1 Fabrication Testing and Control

Holtite-A:

Neutron shield properties of Holtite-A are provided in Chapter 1, Section 1.2.1.3.2. Each manufactured lot of neutron shield material shall be tested to verify the material composition (aluminum and hydrogen), boron concentration and neutron shield density (or specific gravity) meet the requirements specified in Chapter 1 and the Bill-of-Material. A manufactured lot is defined as the total amount of material used to make any number of mixed batches comprised of constituent ingredients from the same lot/batch identification numbers supplied by the constituent manufacturer. Testing shall be performed in accordance with written and approved procedures and/or standards. Material composition, boron concentration and density (or specific gravity) data for each manufactured lot of neutron shield material shall become part of the quality documentation package.

The installation of the neutron shielding material shall be performed in accordance with written and qualified procedures. The procedures shall ensure that mix ratios and mixing methods are controlled in order to achieve proper material composition, boron concentration and distribution, and that pours are controlled in order to prevent gaps from occurring in the material. Samples of each manufactured lot of neutron shield material shall be maintained by Holtec International as part of the quality record documentation package.

Concrete:

The dimensions of the HI-STORM overpack steel shells and the density of the concrete shall be verified to be in accordance with FSAR Appendix 1.D and the design drawings prior to concrete installation. The dimensional inspection and density measurements shall be documented. Also, see Subsection 9.1.2.3 for concrete material testing requirements.

Lead:

The installation of the lead in the HI-TRAC transfer cask shall be performed using written and qualified procedures in order to ensure voids are minimized. Stand pipes or similar devices (as applicable) shall be placed on the upper portion of the cask to ensure the presence of an excess head of liquid lead. Vent risers shall be provided to allow for the air to escape. The HI-TRAC cask components shall be uniformly preheated prior to a lead pour. The temperature of the lead shall be verified to be in the correct temperature range and the lead shall be poured or pumped into place in the annulus (as applicable). The lead pour shall be followed by a controlled cooldown to minimize the gap between the lead and steel shells. The lead shall be cooled from the bottom up and additional molten lead shall be added to the standpipes as necessary to account for lead shrinkage. Each lot of lead shall be tested for chemical composition.

As an alternative to pouring molten lead, the HI-TRAC lead shielding may be installed as pre-cast sections. If pre-cast sections are used, the design of the sections and the installations instructions shall minimize the gaps between adjacent lead sections and between the lead and the transfer cask walls to the extent practicable.

Steel:

Steel plates utilized in the construction of the HI-STORM 100 System shall be dimensionally inspected to assure compliance with the requirements specified on the Design Drawings.

General Requirements for Shield Materials:

1. Test results shall be documented and become part of the quality documentation package.

2. Dimensional inspections of the cavities containing the shielding materials shall assure that the design required amount of shielding material is being incorporated into the fabricated item.

Shielding effectiveness tests shall be performed during fabrication and again after initial loading operations in accordance with Section 9.1.5.2 below and the operating procedures in Chapter 8.

9.1.5.2 Shielding Effectiveness Tests

The effectiveness of the lead pours in the HI-TRAC transfer cask body shall be verified during fabrication by performing gamma scanning on all accessible surfaces of the cask in the lead pour region. The gamma scanning may be performed prior to, or after installation of the water jacket. The purpose of the gamma scanning test is to demonstrate that the gamma shielding of the transfer cask body is at least as effective as that of a lead and steel test block. For the test block, the steel thickness shall be equivalent to the minimum design thickness of steel in the transfer cask component and the lead thickness shall be 5 percent lower than the minimum design thickness of

lead in the transfer cask component (see the Design Drawings for the design values). Data shall be recorded on a 6-inch by 6-inch (nominal) grid pattern over the surfaces to be scanned. Should the measured gamma dose rates exceed those established with the test block, the shielding of that transfer cask component shall be deemed unacceptable. Corrective actions should be taken, if practicable, and the testing re-performed until successful results are achieved. If physical corrective actions are not practicable, the degraded condition may be dispositioned with a written evaluation in accordance with applicable procedures to determine the acceptability of the transfer cask for service. Gamma scanning shall be performed in accordance with written and approved procedures. Dose rate measurements shall be documented and shall become part of the quality documentation package.

The effectiveness of the lead plates in the HI-TRAC pool lid (all transfer cask designs) and transfer lid (HI-TRAC 125 and 100 only) shall be verified during fabrication by performing a UT test of the lead plates. The UT testing will take place before the installation of the plates. The UT testing ensures that the plates are uniform internally. This is an accepted industry procedure for locating voids within the lead plate in order to verify the shielding effectiveness of the plate.

Following the first fuel loading of each HI-STORM 100 System (HI-TRAC transfer cask and HI-STORM storage overpack), a shielding effectiveness test shall be performed at the loading facility site to verify the effectiveness of the radiation shield. This test shall be performed after the HI-STORM overpack and HI-TRAC transfer cask have been loaded with an MPC containing spent fuel assemblies and the MPC has been drained, moisture removed, and backfilled with helium.

Operational neutron and gamma shielding effectiveness tests shall be performed after fuel loading using written and approved procedures. Calibrated neutron and gamma dose rate meters shall be used to measure the actual neutron and gamma dose rates at the surface of the HI-STORM overpack and HI-TRAC. Measurements shall be taken at the locations specified in the ~~technical specifications in Appendix A to CoC 72-1014 and, if necessary, average dose rates computed~~*Radiation Protection Program* for comparison against the prescribed limits. ~~The results of the dose rate measurements shall be compared to the limits specified in the technical specifications.~~ The test is considered acceptable if the dose rate readings are less than or equal to *the calculated* limits. ~~in the technical specifications.~~ If dose rates are higher than the limits, the required actions provided in the ~~technical specifications~~*Radiation Protection Program* shall be completed. Dose rate measurements shall be documented and shall become part of the quality documentation package.

9.1.5.3 Neutron Absorber Tests

Each plate of ~~Boral~~ *neutron absorber* shall be visually inspected ~~by the manufacturer~~ for damage *such as* ~~(e.g.,~~ scratches, cracks, burrs, ~~and~~ peeled cladding~~)~~ ~~and~~ foreign material embedded in the surfaces~~. In addition, the MPC fabricator shall visually inspect the Boral plates on a lot sampling basis. The sample size shall be determined in accordance with MIL-STD-105D or equivalent. The selected~~ *neutron absorber*~~Boral plates shall be inspected for damage such as inclusions, cracks,~~ voids, delamination, and surface finish*, as applicable.*

9.1.5.3.1 Boral (75% Credit)

After manufacturing, a statistical sample of each lot of *neutron absorber*~~Boral~~ shall be tested using wet chemistry and/or neutron attenuation *testing* ~~techniques~~ to verify a minimum ^{10}B content (areal density)~~at~~ *in samples taken from* the ends of the panel. The minimum ^{10}B loading of the *neutron absorber*~~Boral~~ panels for each MPC model is provided in Table 2.1.15. Any panel in which ^{10}B loading is less than the minimum allowed shall be rejected. Testing shall be performed using written and approved procedures. Results shall be documented and become part of the cask quality records documentation package.

9.1.5.3.2 METAMIC® (90% Credit)

NUREG/CR-5661 identifies the main reason for a penalty in the neutron absorber B-10 density as the potential of neutron streaming due to non-uniformities in the neutron absorber, and recommends comprehensive acceptance tests to verify the presence and uniformity of the neutron absorber for credits more than 75%. Since a 90% credit is taken for METAMIC®, the following criteria must be satisfied:

- *The boron carbide powder used in the manufacturing of METAMIC® must have small particle sizes to preclude neutron streaming*
- *The ^{10}B areal density must comply with the limits of Table 2.1.15.*
- *The B_4C powder must be uniformly dispersed locally, i.e. must not show any particle agglomeration. This precludes neutron streaming.*
- *The B_4C powder must be uniformly dispersed macroscopically, i.e. must have a consistent concentration throughout the entire neutron absorber panel.*

To ensure that the above requirements are met the following tests shall be performed:

- *All lots of boron carbide powder are analyzed to meet particle size distribution requirements.*
- *The following qualification testing shall be performed on the first production run of METAMIC® panels for the MPCs in order to validate the acceptability and consistency of the manufacturing process and verify the acceptability of the METAMIC® panels for neutron absorbing capabilities:*

  1) *The boron carbide powder weight percent shall be verified by testing a sample from forty different mixed batches. (A mixed batch is defined as a single mixture of aluminum powder and boron carbide powder used to make one or more billets. Each billet will produce several panels.) The samples shall be drawn from the mixing containers after mixing operations have been completed. Testing shall be performed using the wet chemistry method.*

2) *The ^{10}B areal density shall be verified by testing a sample from one panel from each of forty different mixed batches. The samples shall be drawn from areas contiguous to the manufactured panels of METAMIC® and shall be tested using the wet chemistry method. Alternatively, or in addition to the wet chemistry tests, neutron attenuation tests on the samples may be performed to quantify the actual ^{10}B areal density.*

3) *To verify the local uniformity of the boron particle dispersal, neutron attenuation measurements of random test coupons shall be performed. These test coupons may come from the production run or from pre-production trial runs.*

4) *To verify the macroscopic uniformity of the boron particle distribution, test samples shall be taken from the sides of one panel from five different mixed batches before the panels are cut to their final sizes. The sample locations shall be chosen to be representative of the final product. Wet chemistry or neutron attenuation shall be performed on each of the samples.*

- *During production runs, testing of mixed batches shall be performed on a statistical basis to verify the correct boron carbide weight percent is being mixed.*

- *During production runs, samples from random METAMIC® panels taken from areas contiguous to the manufactured panels shall be tested via wet chemistry to verify the ^{10}B areal density. This test shall be performed to verify the continued acceptability of the manufacturing process.*

The measurements of B_4C particle size, ^{10}B isotopic assay, uniformity of B_4C distribution and ^{10}B areal density shall be made using written and approved procedures. Results shall be documented.

9.1.5.3.3 Installation of the Neutron Absorber Panels

Installation of *neutron absorber*~~Boral~~ panels into the fuel basket shall be performed in accordance with written and approved instructions. Travelers and quality control procedures shall be in place to assure each required cell wall of the MPC basket contains a *neutron absorber*~~Boral~~ panel in accordance with the drawings in Chapter 1. These quality control processes, in conjunction with ~~Boral~~ *in-process* manufacturing testing, provide the necessary assurances that the *neutron absorber*~~Boral~~ will perform its intended function. No additional testing or in-service monitoring of the *neutron absorber material* ~~Boral~~ will be required.

9.1.6 Thermal Acceptance Tests

The thermal performance of the HI-STORM 100 System, including the MPCs and HI-TRAC transfer casks, is demonstrated through analysis in Chapter 4 of the FSAR. Dimensional inspections to verify the item has been fabricated to the dimensions provided in the drawings shall be performed prior to system loading. Following the loading and placement on the storage pad of the first HI-STORM System placed in service, the operability of the natural convective cooling of the HI-STORM 100 System shall be verified by the performance of an air temperature rise test. A description of the test

is described in FSAR Chapter 8.

In addition, the technical specifications require periodic surveillance of the overpack air inlet and outlet vents or, optionally, implementation of an overpack air temperature monitoring program to provide continued assurance of the operability of the HI-STORM 100 heat removal system.

9.1.7 Cask Identification

Each MPC, HI-STORM overpack, and HI-TRAC transfer cask shall be marked with a model number, identification number (to provide traceability back to documentation), and the empty weight of the item in accordance with the marking requirements specified in the Design Drawings in Chapter 1.

Table 9.1.1
MPC INSPECTION AND TEST ACCEPTANCE CRITERIA

Function	Fabrication	Pre-operation	Maintenance and Operations
Visual Inspection and Nondestructive Examination (NDE)	a) Examination of MPC components per ASME Code Section III, Subsections NB and NG, as defined on design drawings, per NB-5300 and NG-5300, as applicable. b) A dimensional inspection of the internal basket assembly and canister shall be performed to verify compliance with design requirements. c) A dimensional inspection of the MPC lid and MPC closure ring shall be performed prior to inserting into the canister shell to verify compliance with design requirements. d) NDE of weldments are defined on the design drawings using standard American Welding Society NDE symbols and/or notations. e) Cleanliness of the MPC shall be verified upon completion of fabrication. f) The packaging of the MPC at the completion of fabrication shall be verified prior to shipment.	a) The MPC shall be visually inspected prior to placement in service at the licensee's facility. b) MPC protection at the licensee's facility shall be verified. c) MPC cleanliness and exclusion of foreign material shall be verified prior to placing in the spent fuel pool.	a) None.

Table 9.1.1 (continued)
MPC INSPECTION AND TEST ACCEPTANCE CRITERIA

Function	Fabrication	Pre-operation	Maintenance and Operations
Structural	a) Assembly and welding of MPC components shall be performed per ASME Code Section IX and III, Subsections NB and NG, as applicable. b) Materials analysis (steel, *neutron absorber*~~Boral~~, etc.), shall be performed and records shall be kept in a manner commensurate with "important to safety" classifications.	a) None.	a) An ultrasonic (UT) examination or multi-layer liquid penetrant (PT) examination of the MPC lid-to-shell weld shall be performed per ASME Section V, Article 5 (or ASME Section V, Article 2). Acceptance criteria for the examination are defined in Subsection 9.1.1.1 and in the Design Drawings. b) ASME Code NB-6000 ~~hydrostatic~~ *pressure* test shall be performed after MPC closure welding. Acceptance criteria are defined in ~~Subsection 9.1.2.2.2~~*the Code*.
Leak Tests	a) Helium leak rate testing shall be performed on all MPC pressure boundary shop welds.	a) None.	a) *None.*~~Helium leak rate testing shall be performed on MPC lid-to-shell, and vent and drain ports to MPC lid field welds after closure welding. Acceptance criteria are defined in the technical specifications.~~

Table 9.1.1 (continued) MPC INSPECTION AND TEST ACCEPTANCE CRITERIA			
Function	**Fabrication**	**Pre-operation**	**Maintenance and Operations**
Criticality Safety	a) The boron content shall be verified at the time of neutron absorber material manufacture. b) The installation of *neutron absorber*~~Boral~~ panels into MPC basket plates shall be verified by inspection.	a) None.	a) None.
Shielding Integrity	a) Material compliance shall be verified through CMTRs. b) Dimensional verification of MPC lid thickness shall be performed.	a) None.	a) None.
Thermal Acceptance	a) None.	a) None.	a) None.
Fit-Up Tests	a) Fit-up of the following components is to be tested during fabrication. - MPC lid - vent/drain port cover plates - MPC closure ring b) A gauge test of all basket fuel compartments.	a) Fit-up of the following components shall be verified during pre-operation. - MPC lid - MPC closure ring - vent/drain cover plates	a) None.
Canister Identification Inspections	a) Verification of identification marking applied at completion of fabrication.	a) Identification marking shall be checked for legibility during pre-operation.	a) None.

Table 9.1.2
HI-STORM STORAGE OVERPACK INSPECTION AND TEST ACCEPTANCE CRITERIA

Function	Fabrication	Pre-operation	Maintenance and Operations
Visual Inspection and Nondestructive Examination (NDE)	Structural Steel Components: a) All ASME and AWS welds shall be visually examined per ASME Section V, Article 9 with acceptance criteria per ASME Section III, Subsection NF, NF-5360. b) All welds requiring PT examination as shown on the Design Drawings shall be PT examined per ASME Section V, Article 6 with acceptance criteria per ASME Section III, Subsection NF, NF-5350. c) All welds requiring MT examination as shown on the drawings shall be MT examined per ASME Section V, Article 7 with acceptance criteria per ASME Section III, Subsection NF, NF-5340. d) NDE of weldments shall be defined on design drawings using standard AWS NDE symbols and/or notations. Concrete Components: The following processes related to concrete components shall be implemented per ACI 349 as clarified in FSAR Appendix 1.D. Concrete testing shall be in accordance with Table 1.D.2. Activities shall be conducted in accordance with written and approved procedures. a) Assembly and examination. b) Materials verification. c) Mixing, pouring, and testing.	a) The overpack shall be visually inspected prior to placement in service. b) Fit-up with mating components (e.g., lid) shall be performed directly whenever practical or using templates or other means. c) Overpack protection at the licensee's facility shall be verified. d) Exclusion of foreign material shall be verified prior to placing the overpack in service at the licensee's facility.	a) Indications identified during visual inspection shall be corrected, reconciled, or otherwise dispositioned. b) Exposed surfaces shall be monitored for coating deterioration and repair/recoat as necessary.

Table 9.1.2 (continued) HI-STORM STORAGE OVERPACK INSPECTION AND TEST ACCEPTANCE CRITERIA			
Function	**Fabrication**	**Pre-operation**	**Maintenance and Operations**
Visual Inspection and Nondestructive Examination (NDE) (continued)	General: a) Cleanliness of the overpack shall be verified upon completion of fabrication. b) Packaging of the overpack at the completion of shop fabrication shall be verified prior to shipment.		
Structural	a) No structural or pressure tests are required for the overpack during fabrication. b) Concrete compressive strength tests shall be performed per ASTM C39.	a) No structural or pressure tests are required for the overpack during pre-operation.	a) No structural or pressure tests are required for the overpack during operation.
Leak Tests	a) None.	a) None.	a) None.
Criticality Safety	a) No neutron absorber tests of the overpack are required for criticality safety during fabrication.	a) None.	a) None.
Shielding Integrity	a) Concrete density shall be verified per ACI-349 as clarified by FSAR Appendix 1.D, at time of placement. b) Shell thicknesses and dimensions between inner and outer shells shall be verified as conforming to design drawings prior to concrete placement. c) Verification of material composition shall be performed.	a) None	a) A shielding effectiveness *test* shall be performed after the initial fuel loading. ~~in accordance with the Technical Specifications.~~ Repeat shielding effectiveness test every five years as part of the Maintenance Program described in FSAR Section 9.2.

Table 9.1.2 (continued)
HI-STORM STORAGE OVERPACK INSPECTION AND TEST ACCEPTANCE CRITERIA

Function	Fabrication	Pre-operation	Maintenance and Operations
Thermal Acceptance	a) Inner shell I.D. and vent size, configuration and placement shall be verified.	a) No pre-operational testing related to the thermal characteristics of the overpack is required.	a) Air temperature rise test(s) shall be performed after initial loading of the first HI-STORM 100 System in accordance with the operating procedures in Chapter 8. b) Periodic surveillance shall be performed by either (1) or (2) below, at the licensee's discretion.~~in accordance with the technical specifications:~~ (1) Inspection of overpack inlet and outlet air vent openings for debris and other obstructions. (2) Temperature monitoring.
Cask Identification	a) Verification that the overpack identification is present in accordance with the drawings shall be performed upon completion of assembly.	a) The overpack identification shall be checked prior to loading..	a) The overpack identification shall be periodically inspected per licensee procedures and repaired or replaced if damaged.
Fit-up Tests	a) Lid fit-up with the overpack shall be verified following fabrication.	a) None.	a) None.

Table 9.1.3
HI-TRAC TRANSFER CASK INSPECTION AND TEST ACCEPTANCE CRITERIA

Function	Fabrication	Pre-operation	Maintenance and Operations
Visual Inspection and Nondestructive Examination (NDE)	a) All ASME and AWS welds shall be visually examined per ASME Section V, Article 9 with acceptance criteria per ASME Section III, Subsection NF, NF-5360. b) All welds requiring PT examination as shown on the Design Drawings shall be PT examined per ASME Section V, Article 6 with acceptance criteria per ASME Section III, Subsection NF, NF-5350. c) All welds requiring MT examination as shown on the Design Drawings shall be MT examined per ASME Section V, Article 7 with acceptance criteria per ASME Section III, Subsection NF, NF-5340. d) NDE of weldments shall be defined on design drawings using standard AWS NDE symbols and/or notations e) Cleanliness of the transfer cask shall be verified upon completion of fabrication. f) Packaging of the transfer cask at the completion of fabrication shall be verified prior to shipment.	a) The transfer cask shall be visually inspected prior to placement in service. b) Transfer cask protection at the licensee's facility shall be verified. c) Transfer cask cleanliness and exclusion of foreign material shall be verified prior to use.	a) Annual visual inspections of the transfer cask shall be performed to assure continued compliance with drawing requirements. (See footnote for Table 9.2.1).

Table 9.1.3 (continued)
HI-TRAC TRANSFER CASK INSPECTION AND TEST ACCEPTANCE CRITERIA

Function	Fabrication	Pre-operation	Maintenance and Operations
Structural	a) Verification of structural materials shall be performed through receipt inspection and review of certified material test reports (CMTRs) obtained in accordance with the item's quality category. a) A load test of the lifting trunnions shall be performed during fabrication per ANSI N14.6. b) A pressure test of the neutron shield water jacket shall be performed during fabrication.	a) None.	a) Annual load testing of the lifting trunnions shall be performed per ANSI N14.6. (See footnote to Table 9.2.1). b) The set pressure of the relief valve on the neutron shield water jacket shall be verified by calibration annually. (See footnote to Table 9.2.1)
Leak Tests	a) None.	a) None.	a) None.
Criticality Safety	a) None.	a) None.	a) None.

Table 9.1.3 (continued)
TRANSFER CASK INSPECTION AND TEST ACCEPTANCE CRITERIA

Function	Fabrication	Pre-operation	Maintenance and Operations
Thermal Acceptance	a) The thermal properties of the transfer cask are established by calculation and inspection, and are not tested during fabrication.	a) None.	a) None
Cask Identification	a) Verification that the transfer cask identification is present in accordance with the drawings shall be performed upon completion of assembly.	a) The transfer cask identification shall be checked prior to loading..	a) The transfer cask identification shall be periodically inspected per licensee procedures and repaired or replaced if damaged.
Fit-up Tests	a) Fit-up tests of the transfer cask components (top, in-pool, and transfer lids) shall be performed during fabrication.	a) Fit-up test of the transfer cask lifting trunnions with the transfer cask lifting yoke shall be performed. b) Fit-up test of the transfer cask pocket trunnions with the horizontal transfer skid shall be performed.	a) Fit-up of the top, in-pool, and transfer lids shall be verified prior to use.

Table 9.1.4 HI-STORM 100 NDE REQUIREMENTS			
MPC			
Weld Location	**NDE Requirement**	**Applicable Code**	**Acceptance Criteria (Applicable Code)**
Shell longitudinal seam	RT	ASME Section V, Article 2 (RT)	RT: ASME Section III, Subsection NB, Article NB-5320
	PT (surface)	ASME Section V, Article 6 (PT)	PT: ASME Section III, Subsection NB, Article NB-5350
Shell circumferential seam	RT	ASME Section V, Article 2 (RT)	RT: ASME Section III, Subsection NB, Article NB-5320
	PT (surface)	ASME Section V, Article 6 (PT)	PT: ASME Section III, Subsection NB, Article NB-5350
Baseplate-to-shell	RT or UT	ASME Section V, Article 2 (RT) ASME Section V, Article 5 (UT)	RT: ASME Section III, Subsection NB, Article NB-5320 UT: ASME Section III, Subsection NB, Article NB-5330
	PT (surface)	ASME Section V, Article 6 (PT)	PT: ASME Section III, Subsection NB, Article NB-5350

Table 9.1.4 (continued) HI-STORM 100 NDE REQUIREMENTS			
MPC			
Weld Location	**NDE Requirement**	**Applicable Code**	**Acceptance Criteria (Applicable Code)**
Lid-to-shell	PT (root and final pass) and multi-layer PT (if UT is not performed). PT (surface following ~~hydrostatic~~*pressure* test)	ASME Section V, Article 6 (PT)	PT: ASME Section III, Subsection NB, Article NB-5350
	UT (if multi-layer PT is not performed)	ASME Section V, Article 5 (UT)	UT: ASME Section III, Subsection NB, Article NB-5332
Closure ring-to-shell	PT (final pass)	ASME Section V, Article 6 (PT)	PT: ASME Section III, Subsection NB, Article NB-5350
Closure ring-to-lid	PT (final pass)	ASME Section V, Article 6 (PT)	PT: ASME Section III, Subsection NB, Article NB-5350
Closure ring radial welds	PT (final pass)	ASME Section V, Article 6 (PT)	PT: ASME Section III, Subsection NB, Article NB-5350
Port cover plates-to-lid	PT (root and final pass)	ASME Section V, Article 6 (PT)	PT: ASME Section III, Subsection NB, Article NB-5350
Lift lug, lift lug baseplate, and fuel spacers	PT (surface)	ASME Section V, Article 6 (PT)	PT: ASME Section III, Subsection NG, Article NG-5350
Vent and drain port cover plate plug welds	PT (surface)	ASME Section V, Article 6 (PT)	PT: ASME Section III, Subsection NG, Article NG-5350

Table 9.1.4 (continued) HI-STORM 100 NDE REQUIREMENTS			
HI-STORM OVERPACK			
Weld Location	**NDE Requirement**	**Applicable Code**	**Acceptance Criteria (Applicable Code)**
N/A	N/A	N/A	N/A
HI-TRAC TRANSFER CASK			
Weld Location	**NDE Requirement**	**Applicable Code**	**Acceptance Criteria (Applicable Code)**
HI-TRAC Body: Radial ribs and short ribs to outer shell	PT (surface) or MT	ASME Section V, Article 6 (PT) ASME Section V, Article 7 (MT)	PT: ASME Section III, Subsection NF, Article NF-5350 MT: ASME Section III, Subsection NF, Article NF-5340
HI-TRAC Body: Water jacket end plate-to-radial channel or enclosure shell panel	PT (surface) or MT	ASME Section V, Article 6 (PT) ASME Section V, Article 7 (MT)	PT: ASME Section III, Subsection NF, Article NF-5350 MT: ASME Section III, Subsection NF, Article NF-5340
Pool Lid: Pool lid top plate-to-pool lid outer ring [HI-TRAC 125 and 125D only]	PT (surface) or MT	ASME Section V, Article 6 (PT) ASME Section V, Article 7 (MT)	PT: ASME Section III, Subsection NF, Article NF-5350 MT: ASME Section III, Subsection NF, Article NF-5340
Pool Lid: Pool lid bottom plate-to-pool lid outer ring [HI-TRAC 125 and 125D only]	PT (surface) or MT	ASME Section V, Article 6 (PT) ASME Section V, Article 7 (MT)	PT: ASME Section III, Subsection NF, Article NF-5350 MT: ASME Section III, Subsection NF, Article NF-5340

Table 9.1.4 (continued) HI-STORM 100 NDE REQUIREMENTS HI-TRAC TRANSFER CASK			
Weld Location	**NDE Requirement**	**Applicable Code**	**Acceptance Criteria (Applicable Code)**
HI-TRAC Body: Water jacket end plate-to-outer shell	PT (surface) or MT	ASME Section V, Article 6 (PT) ASME Section V, Article 7 (MT)	PT: ASME Section III, Subsection NF, Article NF-5350 MT: ASME Section III, Subsection NF, Article NF-5340
HI-TRAC Body: Outer shell-to-outer shell longitudinal and circumferential welds	PT (surface) or MT	ASME Section V, Article 6 (PT) ASME Section V, Article 7 (MT)	PT: ASME Section III, Subsection NF, Article NF-5350 MT: ASME Section III, Subsection NF, Article NF-5340
HI-TRAC Body: Radial ribs and short ribs –to-enclosure shell panel	PT (surface) or MT	ASME Section V, Article 6 (PT) ASME Section V, Article 7 (MT)	PT: ASME Section III, Subsection NF, Article NF-5350 MT: ASME Section III, Subsection NF, Article NF-5340
HI-TRAC Body: Jacket drain pipe and couplings	PT (surface) or MT	ASME Section V, Article 6 (PT) ASME Section V, Article 7 (MT)	PT: ASME Section III, Subsection NF, Article NF-5350 MT: ASME Section III, Subsection NF, Article NF-5340
HI-TRAC Body: Outer shell-to-bottom flange	PT (surface) or MT	ASME Section V, Article 6 (PT) ASME Section V, Article 7 (MT)	PT: ASME Section III, Subsection NF, Article NF-5350 MT: ASME Section III, Subsection NF, Article NF-5340

Table 9.1.4 (continued) HI-STORM 100 NDE REQUIREMENTS HI-TRAC TRANSFER CASK			
Weld Location	**NDE Requirement**	**Applicable Code**	**Acceptance Criteria (Applicable Code)**
HI-TRAC Body: Outer shell-to-top flange	PT (surface) or MT	ASME Section V, Article 6 (PT) ASME Section V, Article 7 (MT)	PT: ASME Section III, Subsection NF, Article NF-5350 MT: ASME Section III, Subsection NF, Article NF-5340
HI-TRAC Body: Lifting trunnion block-to-top flange	PT (surface) or MT	ASME Section V, Article 6 (PT) ASME Section V, Article 7 (MT)	PT: ASME Section III, Subsection NF, Article NF-5350 MT: ASME Section III, Subsection NF, Article NF-5340
HI-TRAC Body: Lifting trunnion block-to-outer and inner shells	PT (surface) or MT	ASME Section V, Article 6 (PT) ASME Section V, Article 7 (MT)	PT: ASME Section III, Subsection NF, Article NF-5350 MT: ASME Section III, Subsection NF, Article NF-5340
HI-TRAC Body: Pocket trunnion-to-outer shell (HI-TRAC 125 and 100 only)	PT (surface) or MT	ASME Section V, Article 6 (PT) ASME Section V, Article 7 (MT)	PT: ASME Section III, Subsection NF, Article NF-5350 MT: ASME Section III, Subsection NF, Article NF-5340
HI-TRAC Body: Top lid welds except as noted on applicable drawings	PT (surface) or MT	ASME Section V, Article 6 (PT) ASME Section V, Article 7 (MT)	PT: ASME Section III, Subsection NF, Article NF-5350 MT: ASME Section III, Subsection NF, Article NF-5340
HI-TRAC Body: Pocket trunnion-to-enclosure shell panel and radial rib (HI-TRAC 125 and 100 only)	PT (surface) or MT	ASME Section V, Article 6 (PT) ASME Section V, Article 7 (MT)	PT: ASME Section III, Subsection NF, Article NF-5350 MT: ASME Section III, Subsection NF, Article NF-5340

Table 9.1.4 (continued) HI-STORM 100 NDE REQUIREMENTS HI-TRAC TRANSFER CASK			
Weld Location	**NDE Requirement**	**Applicable Code**	**Acceptance Criteria (Applicable Code)**
HI-TRAC Body: Lower water jacket welds	PT (surface) or MT	ASME Section V, Article 6 (PT) ASME Section V, Article 7 (MT)	PT: ASME Section III, Subsection NF, Article NF-5350 MT: ASME Section III, Subsection NF, Article NF-5340
HI-TRAC Body: Gusset-to-baseplate, outer shell and water jacket bottom plate (HI-TRAC 125D only)	PT (surface) or MT	ASME Section V, Article 6 (PT) ASME Section V, Article 7 (MT)	PT: ASME Section III, Subsection NF, Article NF-5350 MT: ASME Section III, Subsection NF, Article NF-5340
Transfer Lid: Lid intermediate plate and lead cover plate-to-lid top plate & lid bottom plate	PT (surface) or MT	ASME Section V, Article 6 (PT) ASME Section V, Article 7 (MT)	PT: ASME Section III, Subsection NF, Article NF-5350 MT: ASME Section III, Subsection NF, Article NF-5340
Transfer Lid: Door top plate-to-door wheel housing	PT (surface) or MT	ASME Section V, Article 6 (PT) ASME Section V, Article 7 (MT)	PT: ASME Section III, Subsection NF, Article NF-5350 MT: ASME Section III, Subsection NF, Article NF-5340
Transfer Lid: Door side plate-to-door wheel housing	PT (surface) or MT	ASME Section V, Article 6 (PT) ASME Section V, Article 7 (MT)	PT: ASME Section III, Subsection NF, Article NF-5350 MT: ASME Section III, Subsection NF, Article NF-5340
Transfer Lid: Door side plate-to-door end plate	PT (surface) or MT	ASME Section V, Article 6 (PT) ASME Section V, Article 7 (MT)	PT: ASME Section III, Subsection NF, Article NF-5350 MT: ASME Section III, Subsection NF, Article NF-5340
Transfer Lid: Lead cover plate-to-lead cover side plate	PT (surface) or MT	ASME Section V, Article 6 (PT) ASME Section V, Article 7 (MT)	PT: ASME Section III, Subsection NF, Article NF-5350 MT: ASME Section III, Subsection NF, Article NF-5340

9.2 MAINTENANCE PROGRAM

An ongoing maintenance program shall be defined and incorporated into the HI-STORM 100 System Operations Manual which shall be prepared and issued prior to the delivery and first use of the system to each user. This document shall delineate the detailed inspections, testing, and parts replacement necessary to ensure continued structural, thermal, and confinement performance; radiological safety, and proper handling of the system in accordance with 10CFR72 regulations, the conditions in the Certificate of Compliance, and the design requirements and criteria contained in this FSAR.

The HI-STORM 100 System is totally passive by design. There are no active components or monitoring systems required to assure the performance of its safety functions. As a result, only minimal maintenance will be required over its lifetime, and this maintenance would primarily result from weathering effects in storage. Typical of such maintenance would be the reapplication of corrosion inhibiting materials on accessible external surfaces. Visual inspection of the vent screens is required to ensure the air inlets and outlets are free from obstruction (or alternatively, temperature monitoring may be utilized). Such maintenance requires methods and procedures no more demanding than those currently in use at power plants.

Maintenance activities shall be performed under the licensee's NRC-approved quality assurance program. Maintenance activities shall be administratively controlled and the results documented. The maintenance program schedule for the HI-STORM 100 System is provided in Table 9.2.1.

9.2.1 Structural and Pressure Parts

Prior to each fuel loading, a visual examination in accordance with a written procedure shall be required of the HI-TRAC lifting trunnions and pocket trunnion recesses. The examination shall inspect for indications of overstress such as cracking, deformation, or wear marks. Repairs or replacement in accordance with written and approved procedures shall be required if unacceptable conditions are identified.

A load test on the transfer cask trunnions shall be performed annually or prior to the next HI-TRAC use if the period the HI-TRAC is out of use exceeds one year. The requirements are specified in Section 9.1.2.1.

As described in FSAR Chapters 7 and 11, there are no credible normal, off-normal, or accident events which can cause the structural failure of the MPC. Therefore, periodic structural or pressure tests on the MPCs following the initial acceptance tests are not required as part of the storage maintenance program.

9.2.2 Leakage Tests

There are no seals or gaskets used on the fully-welded MPC confinement system. As described in Chapters 7 and 11, there are no credible normal, off-normal, or accident events which can cause

the failure of the MPC confinement boundary welds. Therefore, leakage tests are not required as part of the storage maintenance program.

9.2.3 Subsystem Maintenance

The HI-STORM 100 System does not include any subsystems which provide auxiliary cooling. Normal maintenance and calibration testing will be required on the vacuum drying, helium backfill, and leakage testing systems. Rigging, remote welders, cranes, and lifting beams shall also be inspected prior to each loading campaign to ensure proper maintenance and continued performance is achieved. Auxiliary shielding provided during on-site transfer operations with the HI-STORM 100 require no maintenance. If the cask user chooses to use an air temperature monitoring system in lieu of visual inspection of the air inlet and outlet vents, the thermocouples and associated temperature monitoring instrumentation shall be maintained and calibrated in accordance with the user's QA program commensurate with the equipment's safety classification and designated QA category. See also FSAR Section 9.2.6.

9.2.4 Pressure Relief Valves

The pressure relief valves used on the water jackets for the HI-TRAC transfer cask shall be calibrated on an annual basis (or prior to the next HI-TRAC use if the period the HI-TRAC is out of use exceeds one year) to ensure pressure relief setting is 60 +2/-0 psig or replaced with factory-set relief valves.

9.2.5 Shielding

The gamma and neutron shielding materials in the HI-STORM overpack, HI-TRAC, and MPC degrade negligibly over time or as a result of usage. To ensure continuing compliance of the HI-STORM 100 System to the design basis dose rate values, a shielding effectiveness test shall be performed every five years after placement into service.

Radiation monitoring of the ISFSI by the licensee in accordance with 10CFR72.106(b) provides ongoing evidence and confirmation of shielding integrity and performance. If increased radiation doses are indicated by the facility monitoring program, additional surveys of overpacks shall be performed to determine the cause of the increased dose rates.

The water level in the HI-TRAC water jacket shall be verified prior to each loading campaign in accordance with the licensee's approved operations procedures.

The ~~Boral~~ *neutron absorber* panels installed in the MPC baskets are not expected to degrade under normal long-term storage conditions. The use of Boral in similar nuclear applications is discussed in Chapter 1, and the long-term performance in a dry, inert gas atmosphere is evaluated in Chapter 3. *A similar discussion is provided for METAMIC® neutron absorber material.* Therefore, no periodic verification testing of neutron poison material is required on the HI-STORM 100 System.

9.2.6 Thermal

In order to assure that the HI-STORM 100 System continues to provide effective thermal performance during storage operations, surveillance of the air vents (or alternatively, by temperature monitoring) shall be performed in accordance with ~~the Technical Specifications and~~ written procedures.

For those licensees choosing to implement temperature monitoring as the means to verify overpack heat transfer system operability, a maintenance and calibration program shall be established in accordance with the plant-specific Quality Assurance Program, the equipment's quality category, and manufacturer's recommendations.

Table 9.2.1

HI-STORM SYSTEM MAINTENANCE PROGRAM SCHEDULE

Task	Frequency
Overpack cavity visual inspection	Prior to fuel loading
Overpack bolt visual inspection	Prior to installation during each use
Overpack external surface (accessible) visual examination	Annually, during storage operation
Overpack vent screen visual inspection for damage, holes, etc.	Monthly
HI-STORM 100 Shielding Effectiveness Test	In accordance with Technical Specifications after initial fuel loading, and every five years thereafter under the Maintenance Program
HI-TRAC cavity visual inspection	Prior to each handling campaign
HI-TRAC lifting trunnion and pocket trunnion recess visual inspection	Prior to each handling campaign
Load Testing of HI-TRAC Lifting Trunnions	Annually†
HI-TRAC pressure relief valve calibration	Annually†
HI-TRAC internal and external visual inspection for compliance to design drawings	Annually†
HI-TRAC water jacket water level visual examination	Prior to each handling campaign
HI-TRAC and Overpack visual inspection of identification markings	Annually
Overpack Air Temperature Monitoring System	Per licensee's QA program and manufacturer's recommendations

† Or prior to next HI-TRAC use if the period the HI-TRAC is out of use exceeds one year.

10.2 RADIATION PROTECTION DESIGN FEATURES

The development of the HI-STORM 100 System has focused on design provisions to address the considerations summarized in Sections 10.1.2 and 10.1.3. The intent has been to improve on past concrete-based dry storage system designs by developing HI-STORM 100 as a hybrid of current metal and concrete storage system technologies. The design is, therefore, an evolution in storage systems, which incorporates preferred features from concrete storage, canister-based systems while retaining several of the advantages of metal casks as well. This approach results in a reduction in the need for maintenance, in overall radiation levels, and in the time spent on maintenance, when compared with current concrete-based dry storage systems. The following specific design features ensure a high degree of confinement integrity and radiation protection:

- HI-STORM 100 has been designed to meet storage condition dose rates required by 10CFR72 [10.0.1] for ~~five~~*three*-year cooled fuel;

- HI-STORM 100 has been designed to accommodate a maximum number of PWR or BWR fuel assemblies to minimize the number of cask systems that must be handled and stored at the storage facility and later transported off-site;

- HI-STORM 100 overpack structure is virtually maintenance free, especially over the years following its initial loading, because of the outer metal shell. The metal shell and its protective coating provide a high level of resistance to corrosion and other forms of degradation (e.g., erosion);

- HI-STORM 100 has been designed for redundant, multi-pass welded closures on the MPC; consequently, no monitoring of the confinement boundary is necessary and no gaseous or particulate releases occur for normal, off-normal or credible accident conditions;

- HI-TRAC transfer cask has a transfer step and other auxiliary shielding devices which eliminates streaming paths and simplify operations;

- The pool lid maximizes available fuel assembly water coverage in the spent fuel pool.

- The transfer lid is designed for quick alignment with HI-STORM; and

- HI-STORM 100 has been designed to allow close positioning (pitch) on the ISFSI storage pad, thereby increasing the ISFSI self-shielding by decreasing the view factors and reducing exposures to on-site and off-site personnel.

10.3 ESTIMATED ON-SITE COLLECTIVE DOSE ASSESSMENT

This section provides the estimates of the cumulative exposure to personnel performing loading, unloading and transfer operations using the HI-STORM system. This section uses the shielding analysis provided in Chapter 5 and the operations procedures provided in Chapter 8 to develop a dose assessment. The dose assessment is provided in Tables 10.3.1, 10.3.2, and 10.3.3.

The dose rates from the HI-STORM 100 overpack, MPC lid, HI-TRAC transfer cask, and HI-STAR 100 overpack are calculated to determine the dose to personnel during the various loading and unloading operations. The dose rates are also calculated for the various conditions of the cask that may affect the dose rates to the operators (e.g., MPC water level, HI-TRAC annulus water level, neutron shield water level, presence of temporary shielding). The dose rates around the 100-Ton HI-TRAC transfer cask are based on 24 PWR fuel assemblies with a burnup of ~~42,500~~*46,000* MWD/MTU and cooling of ~~5~~*3* years including BPRAs. The dose rates around the 125-Ton HI-TRAC transfer cask are based on 24 PWR fuel assemblies with a burnup of ~~57,500~~*75,000* MWD/MTU and cooling of ~~12~~*5* years including BPRAs. The dose rates around the HI-STORM 100 overpack are based on 24 PWR fuel assemblies with a burnup of ~~52~~*47*,500 MWD/MTU and cooling of ~~5~~*3* years. The selection of these fuel assembly types in all fuel cell locations bound all possible PWR and BWR loading scenarios for the HI-STORM System from a dose-rate perspective. No assessment is made with respect to background radiation since background radiation can vary significantly by site. In addition, exposures are based on work being performed with the temporary shielding described in Table 10.1.2.

The choice of burnup and cooling times used in this chapter is extremely conservative. The bounding burnup and cooling time that resulted in the highest dose rates around the 100-ton and 125-ton HI-TRACs were used in conjunction with the very conservative burnup and cooling time for the HI-STORM 100 overpack (as discussed in Section 5.1). In addition, including the source term from BPRAs increases the level of conservatism. The maximum dose rate due to BPRAs was used in this analysis. As stated in Chapter 5, using the maximum source for the BPRAs in conjunction with the bounding burnup and cooling time for fuel assemblies is very conservative as it is not expected that burnup and cooling times of the BPRAs and fuel assemblies would be such that they are both at the maximum design basis values. This combined with the already conservative dose rates for the HI-TRACs and HI-STORMs results in an upper bound estimate of the occupational exposure. Users' radiation protection programs will assure appropriate temporary shielding is used based on actual fuel to be loaded and resulting dose rates in the field.

For each step in Tables 10.3.1 through 10.3.3, the operator work location is identified. These correspond to the locations identified in Figure 10.3.1. The relative locations refer to both the HI-STORM 100 Overpack and the HI-STORM 100s Overpack. The dose rate location points around the transfer cask and overpack were selected to model actual worker locations and cask conditions during the operation. Cask operators typically work at an arms-reach distance from the cask. To account for this, an 18-inch distance was used to estimate the dose rate for the

worker. This assessment addresses only the operators that perform work on or immediately adjacent to the cask.

Justification for the duration of operations along with the corresponding procedure steps from Chapter 8 are also provided in the tables. The assumptions used in developing time durations are based on mockups of the MPC, review of design drawings, walk-downs using other equipment to represent the HI-TRAC transfer cask and HI-STORM 100 overpack the HI-STAR 100 overpack and MPC-68 prototype, consultation with UST&D (weld examination) and consultation with cask operations personnel from Calvert Cliffs Nuclear Power Plant (for items such as lid installation and decontamination). In addition, for the shielding calculations, only the Temporary Shield Ring was assumed to be in place for applicable portions of the operations.

Tables 10.3.1a, 10.3.1b, and 10.3.1c provide a summary of the dose assessment for a HI-STORM 100 System loading operation using the 125-ton HI-TRAC, the 100-ton HI-TRAC, and the 125-ton HI-TRAC 125D respectively. Tables 10.3.2a, 2b, and 2c provide a summary of the dose assessment for HI-STORM 100 System unloading operations using the 125-ton HI-TRAC, the 100-ton HI-TRAC, and 125-ton HI-TRAC 125D respectively. Tables 10.3.3a, 3b, and 3c provide a summary of the dose assessment for transferring the MPC to a HI-STAR 100 overpack as described in Section 8.5 of the operating procedures using the 125-ton HI-TRAC and the 100-ton HI-TRAC transfer cask, respectively.

10.3.1 Estimated Exposures for Loading and Unloading Operations

The assumptions used to estimate personnel exposures are conservative by design. The main factors attributed to actual personnel exposures are the age and burnup of the spent fuel assemblies and good ALARA practices. To estimate the dose received by a single worker, it should be understood that a canister-based system requires a diverse range of disciplines to perform all the necessary functions. The high visibility and often critical path nature of fuel movement activities have prompted utilities to load canister systems in a round-the-clock mode in most cases. This results in the exposure being spread out over several shifts of operators and technicians with no single shift receiving a majority of the exposure.

The total person-rem exposure from operation of the HI-STORM 100 System is proportional to the number of systems loaded. A typical utility will load approximately four MPCs per reactor cycle to maintain the current available spent fuel pool capacity. Utilities requiring dry storage of spent fuel assemblies typically have a large inventory of spent fuel assemblies that date back to the reactor's first cycle. The older fuel assemblies will have a significantly lower dose rate than the design basis fuel assemblies due to the extended cooling time (i.e., much greater than the values used to compute the dose rates). Users shall assess the cask loading for their particular fuel types (burnup, cooling time) to satisfy the requirements of 10CFR20 [10.1.1].

For licensees using the 100-Ton HI-TRAC transfer cask, design basis dose rates will be higher (than a corresponding 125-Ton HI-TRAC) due to the decreased mass of shielding. Due to the higher expected dose rates from the 100-Ton HI-TRAC, users may need to use the auxiliary shielding (See Table 10.1.2), and should consider preferential loading, and increased precautions

(e.g., additional temporary or auxiliary shielding, remotely operated equipment, additional contamination prevention measures). Actual use of optional dose reduction measures must be decided by each user based on the fuel to be loaded.

10.3.2 Estimated Exposures for Surveillance and Maintenance

Table 10.3.4 provides the maximum occupational exposure required for security surveillance and maintenance of an ISFSI. Although the HI-STORM 100 System requires only minimal maintenance during storage, maintenance will be required around the ISFSI for items such as security equipment maintenance, grass cutting, snow removal, vent system surveillance, drainage system maintenance, and lighting, telephone, and intercom repair. Security surveillance time is based on a daily security patrol around the perimeter of the ISFSI security fence. The estimated dose rates described below are based on a sample array of HI-STORM 100 overpacks fully loaded with design basis fuel assemblies, placed at their minimum required pitch, in a 2 x 6 HI-STORM array. The maintenance worker is assumed to be at a distance of 5 meters from the center of the long edge of the array. The security worker is assumed to be at a distance of 15 meters from the center of the long edge of the array. Users may opt to utilize electronic temperature monitoring of the HI-STORM modules or remote viewing methods instead of performing direct visual observation of the modules. Since security surveillances can be performed from outside the ISFSI, a dose rate of 3 mrem/hour is estimated. For maintenance of the casks and the ISFSI, a dose rate of 10 mrem/hour is estimated

Table 10.3.1a
HI-STORM 100 SYSTEM LOADING OPERATIONS USING THE 125-TON HI-TRAC TRANSFER CASK
ESTIMATED OPERATIONAL EXPOSURES† (~~57,500~~*75,000* MWD/MTU, ~~125~~-YEAR COOLED PWR FUEL)

ACTION	DURATION (MINUTES)	OPERATOR LOCATION (FIGURE 10.3.1)	NUMBER OF OPERATORS	DOSE RATE AT OPERATOR LOCATION (MREM/HR)	DOSE TO INDIVIDUAL (MREM)	TOTAL DOSE (PERSON-MREM)	ASSUMPTIONS
Section 8.1.4							
LOAD PRE-SELECTED FUEL ASSEMBLIES INTO MPC	1020	1	2	1	17.0	34.0	15 MINUTES PER ASSEMBLY/68 ASSY
PERFORM POST-LOADING VISUAL VERIFICATION OF ASSEMBLY IDENTIFICATION	68	1	2	1	1.1	2.3	1 MINUTES PER ASSY/68 ASSY
Section 8.1.5							
INSTALL MPC LID AND ATTACH LIFT YOKE	45	2	2	*2.0*	*1.5*	*3.0*	CONSULTATION WITH CALVERT CLIFFS
RAISE HI-TRAC TO SURFACE OF SPENT FUEL POOL	20	2	2	*2.0*	*0.7*	*1.3*	40 FEET @ 2 FT/MINUTE (CRANE SPEED)
SURVEY MPC LID FOR HOT PARTICLES	3	3A	1	*31.1*	*1.6*	*1.6*	TELESCOPING DETECTOR USED
VERIFY MPC LID IS SEATED	0.5	3A	1	*31.1*	*0.3*	*0.3*	VISUAL VERIFICATION FROM 3 METERS
INSTALL LID RETENTION SYSTEM BOLTS	6	3B	2	*46.4*	*4.6*	*9.3*	24 BOLTS @ 1/PERSON-MINUTE
REMOVE HI-TRAC FROM SPENT FUEL POOL	8.5	3C	1	*117.8*	*16.7*	*16.7*	17 FEET @ 2 FT/MIN (CRANE SPEED)
DECONTAMINATE HI-TRAC BOTTOM	10	3D	1	*142.0*	*23.7*	*23.7*	LONG HANDLED TOOLS, PRELIMINARY DECON
TAKE SMEARS OF HI-TRAC EXTERIOR SURFACES	5	5B	1	*185.3*	*15.4*	*15.4*	50 SMEARS @ 10 SMEARS/MINUTE
DISCONNECT ANNULUS OVERPRESSURE SYSTEM	0.5	5C	1	*82.7*	*0.7*	*0.7*	QUICK DISCONNECT COUPLING
SET HI-TRAC IN CASK PREPARATION AREA	10	4A	1	*46.4*	*7.7*	*7.7*	100 FT @ 10 FT/MIN (CRANE SPEED)
REMOVE NEUTRON SHIELD JACKET FILL PLUG	2	4A	1	*46.4*	*1.5*	*1.5*	SINGLE PLUG, NO SPECIAL TOOLS
INSTALL NEUTRON SHIELD JACKET FILL PLUG	2	5B	1	*185.3*	*6.2*	*6.2*	SINGLE PLUG, NO SPECIAL TOOLS
DISCONNECT LID RETENTION SYSTEM	6	5A	2	*37.3*	*3.7*	*7.5*	24 BOLTS @ 1 BOLT/PERSON MINUTES
MEASURE DOSE RATES AT MPC LID	3	5A	1	*37.3*	*1.9*	*1.9*	TELESCOPING DETECTOR USED

† See notes at bottom of Table 10.3.4.

Table 10.3.1a
HI-STORM 100 SYSTEM LOADING OPERATIONS USING THE 125-TON HI-TRAC TRANSFER CASK ESTIMATED OPERATIONAL EXPOSURES† (~~57,500~~*75,000* MWD/MTU, ~~125~~-YEAR COOLED PWR FUEL)

ACTION	DURATION (MINUTES)	OPERATOR LOCATION (FIGURE 10.3.1)	NUMBER OF OPERATORS	DOSE RATE AT OPERATOR LOCATION (MREM/HR)	DOSE TO INDIVIDUAL (MREM)	TOTAL DOSE (PERSON-MREM)	ASSUMPTIONS
DECONTAMINATE AND SURVEY HI-TRAC	103	5B	1	*185.3*	*318.1*	*318.1*	490 SQ-FT@5 SQ-FT/PRERSON-MINUTE+50 SMEARS@10 SMEARS/MINUTE
INSTALL TEMPORARY SHIELD	16	6A	2	*18.7*	*5.0*	*10.0*	8 SEGMENTS @ 1 SEGMENT/PERSON MIN
FILL TEMPORARY SHIELD RING	25	6A	1	*18.7*	*7.8*	*7.8*	230 GAL @10GPM, LONG HANDLED SPRAY WAND
ATTACH DRAIN LINE TO HI-TRAC DRAIN PORT	0.5	5C	1	*82.7*	*0.7*	*0.7*	QUICK DISCONNECT COUPLING
INSTALL RVOAs	2	6A	1	*18.7*	*0.6*	*0.6*	SINGLE THREADED CONNECTION X 2 RVOAs
ATTACH WATER PUMP TO DRAIN PORT	2	6A	1	*18.7*	*0.6*	*0.6*	POSITION PUMP SELF PRIMING
DISCONNECT WATER PUMP	5	6A	1	*18.7*	*1.6*	*1.6*	DRAIN HOSES MOVE PUMP
DECONTAMINATE MPC LID TOP SURFACE AND SHELL AREA ABOVE INFLATABLE ANNULUS SEAL	6	6A	1	*18.7*	*1.9*	*1.9*	30 SQ-FT @5 SQ-FT/MINUTE+10 SMEARS@10 SMEARS/MINUTE
REMOVE INFLATABLE ANNULUS SEAL	3	6A	1	*18.7*	*0.9*	*0.9*	SEAL PULLS OUT DIRECTLY
SURVEY MPC LID TOP SURFACES AND ACCESSIBLE AREAS OF TOP THREE INCHES OF MPC SHELL	1	6A	1	*18.7*	*0.3*	*0.3*	10 SMEARS@10 SMEARS/MINUTE
INSTALL ANNULUS SHIELD	2	6A	1	*18.7*	*0.6*	*0.6*	SHIELD PLACED BY HAND
CENTER LID IN MPC SHELL	20	6A	3	*18.7*	*6.2*	*18.7*	CONSULTATION WITH CALVERT CLIFFS
INSTALL MPC LID SHIMS	12	6A	2	*18.7*	*3.7*	*7.5*	MEASURED DURING WELD MOCKUP TESTING
POSITION AWS BASEPLATE SHIELD ON MPC LID	20	7A	2	*18.7*	*6.2*	*12.5*	ALIGN AND REMOVE 4 SHACKLES
INSTALL AUTOMATED WELDING SYSTEM ROBOT	8	7A	2	*18.7*	*2.5*	*5.0*	ALIGN AND REMOVE 4 SHACKLES/4 QUICK CONNECTS@1/MIN
PERFORM NDE ON LID WELD	230	7A	1	*18.7*	*71.7*	*71.7*	MEASURED DURING WELD MOCKUP TESTING
ATTACH DRAIN LINE TO VENT PORT	1	7A	1	*18.7*	*0.3*	*0.3*	1" THREADED FITTING NO TOOLS
VISUALLY EXAMINE MPC LID-TO-SHELL WELD FOR LEAKAGE OF WATER	10	7A	1	*18.7*	*3.1*	*3.1*	10 MIN TEST DURATION
DISCONNECT WATER FILL LINE AND DRAIN LINE	2	7A	1	*18.7*	*0.6*	*0.6*	1" THREADED FITTING NO TOOLS X 2

Table 10.3.1a
HI-STORM 100 SYSTEM LOADING OPERATIONS USING THE 125-TON HI-TRAC TRANSFER CASK ESTIMATED OPERATIONAL EXPOSURES† (~~57,500~~*75,000* MWD/MTU, ~~125~~-YEAR COOLED PWR FUEL)

ACTION	DURATION (MINUTES)	OPERATOR LOCATION (FIGURE 10.3.1)	NUMBER OF OPERATORS	DOSE RATE AT OPERATOR LOCATION (MREM/HR)	DOSE TO INDIVIDUAL (MREM)	TOTAL DOSE (PERSON-MREM)	ASSUMPTIONS
REPEAT LIQUID PENETRANT EXAMINATION ON MPC LID FINAL PASS	45	7A	1	*18.7*	*14.0*	*14.0*	5 MIN TO APPLY, 7 MIN TO WIPE, 5 APPLY DEV, INSP (24 IN/MIN)
ATTACH GAS SUPPLY TO VENT PORT	1	7A	1	*18.7*	*0.3*	*0.3*	1" THREADED FITTING NO TOOLS
ATTACH DRAIN LINE TO DRAIN PORT	1	7A	1	*18.7*	*0.3*	*0.3*	1" THREADED FITTING NO TOOLS
Deleted~~CONNECT MSLD SNIFFER TO AUTOMATED WELDING SYSTEM~~	4	~~8A~~	~~1~~	~~*37.9*~~	~~*2.5*~~	~~*2.5*~~	~~SIMPLE ATTACHMENT NO TOOLS~~
Deleted~~DISCONNECT MSLD SNIFFER FROM AUTOMATED WELDING SYSTEM~~	4	~~8A~~	~~1~~	~~*37.9*~~	~~*2.5*~~	~~*2.5*~~	~~SIMPLE ATTACHMENT NO TOOLS~~
ATTACH DRAIN LINE TO VENT PORT	1	8A	1	*37.9*	*0.6*	*0.6*	1" THREADED FITTING NO TOOLS
ATTACH WATER FILL LINE TO DRAIN PORT	1	8A	1	*37.9*	*0.6*	*0.6*	1" THREADED FITTING NO TOOLS
DISCONNECT WATER FILL DRAIN LINES FROM MPC	2	8A	1	*37.9*	*1.3*	*1.3*	1" THREADED FITTING NO TOOLS X 2
ATTACH HELIUM SUPPLY TO VENT PORT	1	8A	1	*37.9*	*0.6*	*0.6*	1" THREADED FITTING NO TOOLS
ATTACH DRAIN LINE TO DRAIN PORT	1	8A	1	*37.9*	*0.6*	*0.6*	1" THREADED FITTING NO TOOLS
DISCONNECT GAS SUPPLY LINE FROM MPC	1	8A	1	*37.9*	*0.6*	*0.6*	1" THREADED FITTING NO TOOLS
DISCONNECT DRAIN LINE FROM MPC	1	8A	1	*37.9*	*0.6*	*0.6*	1" THREADED FITTING NO TOOLS
ATTACHM MOISTURE REMOVAL SYSTEM TO VENT AND DRAIN PORT RVOAs	2	8A	1	*37.9*	*1.3*	*1.3*	1" THREADED FITTING NO TOOLS
DISCONNECT MOISTURE REMOVAL SYSTEM FROM MPC	2	8A	1	*37.9*	*1.3*	*1.3*	1" THREADED FITTING NO TOOLS X 2
CLOSE DRAIN PORT RVOA CAP AND REMOVE DRAIN PORT RVOA	1.5	8A	1	*37.9*	*0.9*	*0.9*	SINGLE THREADED CONNECTION (1 RVOA)
ATTACH HELIUM BACKFILL SYSTEM TO VENT PORT	1	8A	1	*37.9*	*0.6*	*0.6*	1" THREADED FITTING NO TOOLS
DISCONNECT HBS FROM MPC	1	8A	1	*37.9*	*0.6*	*0.6*	1" THREADED FITTING NO TOOLS
CLOSE VENT PORT RVOA AND DISCONNECT VENT PORT RVOA	1.5	8A	1	*37.9*	*0.9*	*0.9*	SINGLE THREADED CONNECTION (1 RVOA)
WIPE INSIDE AREA OF VENT AND DRAIN PORT RECESSES	2	8A	1	*37.9*	*1.3*	*1.3*	2 PORTS, 1 MIN/PORT

Table 10.3.1a
HI-STORM 100 SYSTEM LOADING OPERATIONS USING THE 125-TON HI-TRAC TRANSFER CASK ESTIMATED OPERATIONAL EXPOSURES† (~~57,500~~*75,000* MWD/MTU, ~~125~~-YEAR COOLED PWR FUEL)

ACTION	DURATION (MINUTES)	OPERATOR LOCATION (FIGURE 10.3.1)	NUMBER OF OPERATORS	DOSE RATE AT OPERATOR LOCATION (MREM/HR)	DOSE TO INDIVIDUAL (MREM)	TOTAL DOSE (PERSON-MREM)	ASSUMPTIONS
PLACE COVER PLATE OVER VENT PORT RECESS	1	8A	1	*37.9*	*0.6*	*0.6*	INSTALLED BY HAND NO TOOLS (2/MIN)
PERFORM NDE ON VENT AND DRAIN COVER PLATE WELD	100	8A	1	*37.9*	*63.2*	*63.2*	MEASURED DURING WELD MOCKUP TESTING
~~FLUSH CAVITY WITH HELIUM AND~~ INSTALL SET SCREWS	2	8A	1	*37.9*	*1.3*	*1.3*	4 SET SCREWS @2/MINUTE
PLUG WELD OVER SET SCREWS	8	8A	1	*37.9*	*5.1*	*5.1*	FOUR SINGLE SPOT WELDS @ 1 PER 2 MINTES
Deleted~~INSTALL MSLD OVER VENT PORT COVER PLATE~~	~~2~~	~~8A~~	~~1~~	~~*37.9*~~	~~*1.3*~~	~~*1.3*~~	~~INSTALLED BY HAND NO TOOLS~~
Deleted~~INSTALL MSLD OVER DRAIN PORT COVER PLATE~~	~~2~~	~~8A~~	~~1~~	~~*37.9*~~	~~*1.3*~~	~~*1.3*~~	~~INSTALLED BY HAND NO TOOLS~~
INSTALL AND ALIGN CLOSURE RING	5	8A	1	*37.9*	*3.2*	*3.2*	INSTALLED BY HAND NO TOOLS
PERFORM NDE ON CLOSURE RING WELDS	185	8A	1	*37.9*	*116.9*	*116.9*	MEASURED DURING WELD MOCKUP TESTING
RIG AWS TO CRANE	12	8A	1	*37.9*	*7.6*	*7.6*	10 MIN TO DISCONNECT LINES, 4 SHACKLES@2/MIN
Section 8.1.6							
REMOVE ANNULUS SHIELD	1	8A	1	*37.9*	*0.6*	*0.6*	SHIELD PLACED BY HAND
ATTACH DRAIN LINE TO HI-TRAC	1	9D	1	*354.2*	*5.9*	*5.9*	1" THREADED FITTING NO TOOLS
POSITION HI-TRAC TOP LID	10	9B	2	*37.9*	*6.3*	*12.6*	VERTICAL FLANGED CONNECTION
TORQUE TOP LID BOLTS	12	9B	1	*37.9*	*7.6*	*7.6*	24 BOLTS AT 2/MIN (INSTALL AND TORQUE,1 PASS)
INSTALL MPC LIFT CLEATS AND MPC SLINGS	25	9A	2	*158.5*	*66.0*	*132.1*	INSTALL CLEATS AND HYDRO TORQUE 4 BOLTS
REMOVE TEMPORARY SHIELD RING DRAIN PLUGS	1	9B	1	*37.9*	*0.6*	*0.6*	8 PLUGS @ 8/MIN
REMOVE TEMPORARY SHIELD RING SEGMENTS	4	9A	1	*158.5*	*10.6*	*10.6*	REMOVED BY HAND NO TOOLS (8 SEGS@2/MIN)
ATTACH MPC SLINGS TO LIFT YOKE	4	9A	2	*158.5*	*10.6*	*21.1*	INSTALLED BY HAND NO TOOLS
POSITION HI-TRAC ABOVE TRANSFER STEP	15	9C	1	*117.8*	*29.5*	*29.5*	100 FT @ 10 FT/MIN (CRANE SPEED)+ 5MIN TO ALIGN
REMOVE BOTTOM LID BOLTS	6	10A	1	*354.2*	*35.4*	*35.4*	36 BOLTS@6 BOLTS/MIN IMPACT TOOLS USED
INSTALL TRANSFER LID BOLTS	18	11B	1	*354.2*	*106.3*	*106.3*	36 BOLTS @ 2/MIN IMPACT TOOLS USED 1 PASS
DISCONNECT MPC SLINGS	4	9A	2	*158.5*	*10.6*	*21.1*	INSTALLED BY HAND NO TOOLS
Section 8.1.7							

Table 10.3.1a
HI-STORM 100 SYSTEM LOADING OPERATIONS USING THE 125-TON HI-TRAC TRANSFER CASK ESTIMATED OPERATIONAL EXPOSURES† (~~57,500~~*75,000* MWD/MTU, ~~125~~-YEAR COOLED PWR FUEL)

ACTION	DURATION (MINUTES)	OPERATOR LOCATION (FIGURE 10.3.1)	NUMBER OF OPERATORS	DOSE RATE AT OPERATOR LOCATION (MREM/HR)	DOSE TO INDIVIDUAL (MREM)	TOTAL DOSE (PERSON-MREM)	ASSUMPTIONS
POSITION HI-TRAC ON TRANSPORT DEVICE	20	11A	2	*117.8*	*39.3*	*78.5*	ALIGN TRUNNIONS, DISCONNECT LIFT YOKE
TRANSPORT HI-TRAC TO OUTSIDE TRANSFER LOCATION	90	12A	3	*26.4*	*39.6*	*118.8*	DRIVER AND 2 SPOTTERS
ATTACH OUTSIDE LIFTING DEVICE LIFT LINKS	2	12A	2	*26.4*	*0.9*	*1.8*	2 LINKS@1/MIN
MATE OVERPACKS	10	13B	2	*118.5*	*19.8*	*39.5*	ALIGNMENT GUIDES USED
ATTACH MPC SLINGS TO MPC LIFT CLEATS	10	13A	2	*158.5*	*26.4*	*52.8*	2 SLINGS@5MIN/SLING NO TOOLS
REMOVE TRANSFER LID DOOR LOCKING PINS AND OPEN DOORS	4	13B	2	*118.5*	*7.9*	*15.8*	2 PINS@2MIN/PIN
INSTALL TRIM PLATES	4	13B	2	*118.5*	*7.9*	*15.8*	INSTALLED BY HAND
DISCONNECT SLINGS FROM MPC LIFTING DEVICE	10	13A	2	*158.5*	*26.4*	*52.8*	2 SLINGS@5MIN/SLING
REMOVE MPC LIFT CLEATS AND MPC SLINGS	10	14A	1	*362.5*	*60.4*	*60.4*	4 BOLTS,NO TORQUING
INSTALL HOLE PLUGS IN EMPTY MPC BOLT HOLES	2	14A	1	*362.5*	*12.1*	*12.1*	4 PLUGS AT 2/MIN NO TORQUING
REMOVE HI-STORM VENT DUCT SHIELD INSERTS	2	15A	1	*45.5*	*1.5*	*1.5*	4 SHACKLES@2/MIN
REMOVE ALIGNMENT DEVICE	4	15A	1	*45.5*	*3.0*	*3.0*	REMOVED BY HAND NO TOOLS (4 PCS@1/MIN)
INSTALL HI-STORM LID AND INSTALL LID STUDS/NUTS	25	16A	2	*7.3*	*3.1*	*6.1*	INSTALL LID AND HYDRO TORQUE 4 BOLTS
INSTALL HI-STORM EXIT VENT GAMMA SHIELD CROSS PLATES	4	16B	1	*73.9*	*4.9*	*4.9*	4 PCS @ 1/MIN INSTALL BY HAND NO TOOLS
INSTALL TEMPERATURE ELEMENTS	20	16B	1	*73.9*	*24.6*	*24.6*	4@5MIN/TEMPERATURE ELEMENT
INSTALL EXIT VENT SCREENS	20	16B	1	*73.9*	*24.6*	*24.6*	4 SCREENS@5MIN/SCREEN
REMOVE HI-STORM LID LIFTING DEVICE	2	16A	1	*7.3*	*0.2*	*0.2*	4 SHACKLES@2/MIN
INSTALL HOLE PLUGS IN EMPTY HOLES	2	16A	1	*7.3*	*0.2*	*0.2*	4 PLUGS AT 2/MIN NO TORQUING
PERFORM SHIELDING EFFECTIVENESS TESTING	16	16D	2	*43.8*	*11.7*	*23.3*	16 POINTS@1 MIN
SECURE HI-STORM TO TRANSPORT DEVICE	10	16A	2	*7.3*	*1.2*	*2.4*	ASSUMES AIR PAD
TRANSFER HI-STORM TO ITS DESIGNATED STORAGE LOCATION	40	16C	1	*25.5*	*17.0*	*17.0*	200 FEET @ 4FT/MIN

Table 10.3.1a
HI-STORM 100 SYSTEM LOADING OPERATIONS USING THE 125-TON HI-TRAC TRANSFER CASK ESTIMATED OPERATIONAL EXPOSURES† (~~57,500~~*75,000* MWD/MTU, ~~125~~-YEAR COOLED PWR FUEL)

ACTION	DURATION (MINUTES)	OPERATOR LOCATION (FIGURE 10.3.1)	NUMBER OF OPERATORS	DOSE RATE AT OPERATOR LOCATION (MREM/HR)	DOSE TO INDIVIDUAL (MREM)	TOTAL DOSE (PERSON-MREM)	ASSUMPTIONS
INSERT HI-STORM LIFTING JACKS	4	16D	1	*43.8*	*2.9*	*2.9*	4 JACKS@1/MIN
REMOVE AIR PAD	5	16D	2	*43.8*	*3.6*	*7.3*	1 PAD MOVED BY HAND
REMOVE HI-STORM LIFTING JACKS	4	16D	1	*43.8*	*2.9*	*2.9*	4 JACKS@1/MIN
INSTALL INLET VENT SCREENS/CROSS PLATES	20	16D	1	*43.8*	*14.6*	*14.6*	4 SCREENS@5MIN/SCREEN
PERFORM AIR TEMPERATURE RISE TEST	8	16B	1	*73.9*	*9.8*	*9.8*	8 MEASUREMENTS@1/MIN
TOTAL						~~796.0~~ *1,797.2*~~804.8~~ PERSON-MREM	

Table 10.3.1b
HI-STORM 100 SYSTEM LOADING OPERATIONS USING THE 100-TON HI-TRAC TRANSFER CASK ESTIMATED OPERATIONAL EXPOSURES† (~~42,500~~*46,000* MWD/MTU, ~~5~~*3*-YEAR COOLED PWR FUEL)

ACTION	DURATION (MINUTES)	OPERATOR LOCATION (FIGURE 10.3.1)	NUMBER OF OPERATORS	DOSE RATE AT OPERATOR LOCATION (MREM/HR)	DOSE TO INDIVIDUAL (MREM)	TOTAL DOSE (PERSON-MREM)	ASSUMPTIONS
Section 8.1.4							
LOAD PRE-SELECTED FUEL ASSEMBLIES INTO MPC	1020	1	2	3	51.0	102.0	15 MINUTES PER ASSEMBLY/68 ASSY
PERFORM POST-LOADING VISUAL VERIFICATION OF ASSEMBLY IDENTIFICATION	68	1	2	3	3.4	6.8	1 MINUTES PER ASSY/68 ASSY
Section 8.1.5							
INSTALL MPC LID AND ATTACH LIFT YOKE	45	2	2	*3*	*2.3*	*4.5*	CONSULTATION WITH CALVERT CLIFFS
RAISE HI-TRAC TO SURFACE OF SPENT FUEL POOL	20	2	2	*3*	*1.0*	*2.0*	40 FEET @ 2 FT/MINUTE (CRANE SPEED)
SURVEY MPC LID FOR HOT PARTICLES	3	3A	1	*31.1*	*1.6*	*1.6*	TELESCOPING DETECTOR USED
VERIFY MPC LID IS SEATED	0.5	3A	1	*31.1*	*0.3*	*0.3*	VISUAL VERIFICATION FROM 3 METERS
INSTALL LID RETENTION SYSTEM BOLTS	6	3B	2	*76.3*	*7.6*	*15.3*	24 BOLTS @ 1/PERSON-MINUTE
REMOVE HI-TRAC FROM SPENT FUEL POOL	8.5	3C	1	*663.4*	*94.0*	*94.0*	17 FEET @ 2 FT/MIN (CRANE SPEED)
DECONTAMINATE HI-TRAC BOTTOM	10	3D	1	*432.5*	*72.1*	*72.1*	LONG HANDLED TOOLS, PRELIMINARY DECON
TAKE SMEARS OF HI-TRAC EXTERIOR SURFACES	5	5B	1	*919.1*	*76.6*	*76.6*	50 SMEARS @ 10 SMEARS/MINUTE
DISCONNECT ANNULUS OVERPRESSURE SYSTEM	0.5	5C	1	*241.8*	*2.0*	*2.0*	QUICK DISCONNECT COUPLING
SET HI-TRAC IN CASK PREPARATION AREA	10	4A	1	*76.3*	*12.7*	*12.7*	100 FT @ 10 FT/MIN (CRANE SPEED)
REMOVE NEUTRON SHIELD JACKET FILL PLUG	2	4A	1	*76.3*	*2.5*	*2.5*	SINGLE PLUG, NO SPECIAL TOOLS
INSTALL NEUTRON SHIELD JACKET FILL PLUG	2	5B	1	*919.1*	*30.6*	*30.6*	SINGLE PLUG, NO SPECIAL TOOLS

† See notes at bottom of Table 10.3.4.

Table 10.3.1b
HI-STORM 100 SYSTEM LOADING OPERATIONS USING THE 100-TON HI-TRAC TRANSFER CASK ESTIMATED OPERATIONAL EXPOSURES† (~~42,500~~*46,000* MWD/MTU, ~~5~~*3*-YEAR COOLED PWR FUEL)

ACTION	DURATION (MINUTES)	OPERATOR LOCATION (FIGURE 10.3.1)	NUMBER OF OPERATORS	DOSE RATE AT OPERATOR LOCATION (MREM/HR)	DOSE TO INDIVIDUAL (MREM)	TOTAL DOSE (PERSON-MREM)	ASSUMPTIONS
DISCONNECT LID RETENTION SYSTEM	6	5A	2	*62.5*	*6.3*	*12.5*	24 BOLTS @ 1 BOLT/PERSON MINUTES
MEASURE DOSE RATES AT MPC LID	3	5A	1	*62.5*	*3.1*	*3.1*	TELESCOPING DETECTOR USED
DECONTAMINATE AND SURVEY HI-TRAC	103	5B	1	*919.1*	*1577.8*	*1577.8*	490 SQ-FT@5 SQ-FT/PRERSON-MINUTE+50 SMEARS@10 SMEARS/MINUTE
INSTALL TEMPORARY SHIELD	16	6A	2	*31.3*	*8.3*	*16.7*	8 SEGMENTS @ 1 SEGMENT/PERSON MIN
FILL TEMPORARY SHIELD RING	25	6A	1	*31.3*	*13.0*	*13.0*	230 GAL @10GPM, LONG HANDLED SPRAY WAND
ATTACH DRAIN LINE TO HI-TRAC DRAIN PORT	0.5	5C	1	*241.8*	*2.0*	*2.0*	QUICK DISCONNECT COUPLING
INSTALL RVOAs	2	6A	1	*31.3*	*1.0*	*1.0*	SINGLE THREADED CONNECTION X 2 RVOAs
ATTACH WATER PUMP TO DRAIN PORT	2	6A	1	*31.3*	*1.0*	*1.0*	POSITION PUMP SELF PRIMING
DISCONNECT WATER PUMP	5	6A	1	*31.3*	*2.6*	*2.6*	DRAIN HOSES MOVE PUMP
DECONTAMINATE MPC LID TOP SURFACE AND SHELL AREA ABOVE INFLATABLE ANNULUS SEAL	6	6A	1	*31.3*	*3.1*	*3.1*	30 SQ-FT @5 SQ-FT/MINUTE+10 SMEARS@10 SMEARS/MINUTE
REMOVE INFLATABLE ANNULUS SEAL	3	6A	1	*31.3*	*1.6*	*1.6*	SEAL PULLS OUT DIRECTLY
SURVEY MPC LID TOP SURFACES AND ACCESSIBLE AREAS OF TOP THREE INCHES OF MPC SHELL	1	6A	1	*31.3*	*0.5*	*0.5*	10 SMEARS@10 SMEARS/MINUTE
INSTALL ANNULUS SHIELD	2	6A	1	*31.3*	*1.0*	*1.0*	SHIELD PLACED BY HAND
CENTER LID IN MPC SHELL	20	6A	3	*31.3*	*10.4*	*31.3*	CONSULTATION WITH CALVERT CLIFFS
INSTALL MPC LID SHIMS	12	6A	2	*31.3*	*6.3*	*12.5*	MEASURED DURING WELD MOCKUP TESTING
POSITION AWS BASEPLATE SHIELD ON MPC LID	20	7A	2	*31.3*	*10.4*	*20.9*	ALIGN AND REMOVE 4 SHACKLES

Table 10.3.1b
HI-STORM 100 SYSTEM LOADING OPERATIONS USING THE 100-TON HI-TRAC TRANSFER CASK ESTIMATED OPERATIONAL EXPOSURES† (~~42,500~~*46,000* MWD/MTU, ~~5~~*3*-YEAR COOLED PWR FUEL)

ACTION	DURATION (MINUTES)	OPERATOR LOCATION (FIGURE 10.3.1)	NUMBER OF OPERATORS	DOSE RATE AT OPERATOR LOCATION (MREM/HR)	DOSE TO INDIVIDUAL (MREM)	TOTAL DOSE (PERSON-MREM)	ASSUMPTIONS
INSTALL AUTOMATED WELDING SYSTEM ROBOT	8	7A	2	*31.3*	*4.2*	*8.3*	ALIGN AND REMOVE 4 SHACKLES/4 QUICK CONNECTS@1/MIN
PERFORM NDE ON LID WELD	230	7A	1	*31.3*	*120.0*	*120.0*	MEASURED DURING WELD MOCKUP TESTING
ATTACH DRAIN LINE TO VENT PORT	1	7A	1	*31.3*	*0.5*	*0.5*	1" THREADED FITTING NO TOOLS
VISUALLY EXAMINE MPC LID-TO-SHELL WELD FOR LEAKAGE OF WATER	10	7A	1	*31.3*	*5.2*	*5.2*	10 MIN TEST DURATION
DISCONNECT WATER FILL LINE AND DRAIN LINE	2	7A	1	*31.3*	*1.0*	*1.0*	1" THREADED FITTING NO TOOLS X 2
REPEAT LIQUID PENETRANT EXAMINATION ON MPC LID FINAL PASS	45	7A	1	*31.3*	*23.5*	*23.5*	5 MIN TO APPLY, 7 MIN TO WIPE, 5 APPLY DEV, INSP (24 IN/MIN)
ATTACH GAS SUPPLY TO VENT PORT	1	7A	1	*31.3*	*0.5*	*0.5*	1" THREADED FITTING NO TOOLS
ATTACH DRAIN LINE TO DRAIN PORT	1	7A	1	*31.3*	*0.5*	*0.5*	1" THREADED FITTING NO TOOLS
~~CONNECT MSLD SNIFFER TO AUTOMATED WELDING SYSTEM~~	4	~~8A~~	~~1~~	~~*60.0*~~	~~*4.0*~~	~~*4.0*~~	~~SIMPLE ATTACHMENT NO TOOLS~~
~~DISCONNECT MSLD SNIFFER FROM AUTOMATED WELDING SYSTEM~~	4	~~8A~~	~~1~~	~~*60.0*~~	~~*4.0*~~	~~*4.0*~~	~~SIMPLE ATTACHMENT NO TOOLS~~
ATTACH DRAIN LINE TO VENT PORT	1	8A	1	*60.0*	*1.0*	*1.0*	1" THREADED FITTING NO TOOLS
ATTACH WATER FILL LINE TO DRAIN PORT	1	8A	1	*60.0*	*1.0*	*1.0*	1" THREADED FITTING NO TOOLS
DISCONNECT WATER FILL DRAIN LINES FROM MPC	2	8A	1	*60.0*	*2.0*	*2.0*	1" THREADED FITTING NO TOOLS X 2
ATTACH HELIUM SUPPLY TO VENT PORT	1	8A	1	*60.0*	*1.0*	*1.0*	1" THREADED FITTING NO TOOLS
ATTACH DRAIN LINE TO DRAIN PORT	1	8A	1	*60.0*	*1.0*	*1.0*	1" THREADED FITTING NO TOOLS
DISCONNECT GAS SUPPLY LINE FROM MPC	1	8A	1	*60.0*	*1.0*	*1.0*	1" THREADED FITTING NO TOOLS

Table 10.3.1b
HI-STORM 100 SYSTEM LOADING OPERATIONS USING THE 100-TON HI-TRAC TRANSFER CASK
ESTIMATED OPERATIONAL EXPOSURES† (~~42,500~~*46,000* MWD/MTU, ~~5~~*3*-YEAR COOLED PWR FUEL)

ACTION	DURATION (MINUTES)	OPERATOR LOCATION (FIGURE 10.3.1)	NUMBER OF OPERATORS	DOSE RATE AT OPERATOR LOCATION (MREM/HR)	DOSE TO INDIVIDUAL (MREM)	TOTAL DOSE (PERSON-MREM)	ASSUMPTIONS
DISCONNECT DRAIN LINE FROM MPC	1	8A	1	*60.0*	*1.0*	*1.0*	1" THREADED FITTING NO TOOLS
ATTACH MOISTURE REMOVAL SYSTEM () TO VENT AND DRAIN PORT RVOAs	2	8A	1	*60.0*	*2.0*	*2.0*	1" THREADED FITTING NO TOOLS
DISCONNECT MOISTURE REMOVAL SYSTEM FROM MPC	2	8A	1	*60.0*	*2.0*	*2.0*	1" THREADED FITTING NO TOOLS X 2
CLOSE DRAIN PORT RVOA CAP AND REMOVE DRAIN PORT RVOA	1.5	8A	1	*60.0*	*1.5*	*1.5*	SINGLE THREADED CONNECTION (1 RVOA)
ATTACH HELIUM BACKFILL SYSTEM TO VENT PORT	1	8A	1	*60.0*	*1.0*	*1.0*	1" THREADED FITTING NO TOOLS
DISCONNECT HBS FROM MPC	1	8A	1	*60.0*	*1.0*	*1.0*	1" THREADED FITTING NO TOOLS
CLOSE VENT PORT RVOA AND DISCONNECT VENT PORT RVOA	1.5	8A	1	*60.0*	*1.5*	*1.5*	SINGLE THREADED CONNECTION (1 RVOA)
WIPE INSIDE AREA OF VENT AND DRAIN PORT RECESSES	2	8A	1	*60.0*	*2.0*	*2.0*	2 PORTS, 1 MIN/PORT
PLACE COVER PLATE OVER VENT PORT RECESS	1	8A	1	*60.0*	*1.0*	*1.0*	INSTALLED BY HAND NO TOOLS (2/MIN)
PERFORM NDE VENT AND DRAIN COVER PLATE WELD	100	8A	1	*60.0*	*100.0*	*100.0*	MEASURED DURING WELD MOCKUP TESTING
~~FLUSH CAVITY WITH HELIUM AND~~ INSTALL SET SCREWS	2	8A	1	*60.0*	*2.0*	*2.0*	4 SET SCREWS @2/MINUTE
PLUG WELD OVER ET SCREWS	8	8A	1	*60.0*	*8.0*	*8.0*	FOUR SINGLE SPOT WELDS @ 1 PER 2 MINTES
Deleted~~INSTALL MSLD OVER VENT PORT COVER PLATE~~	~~2~~	~~8A~~	~~1~~	~~*60.0*~~	~~*2.0*~~	~~*2.0*~~	~~INSTALLED BY HAND NO TOOLS~~
Deleted~~INSTALL MSLD OVER DRAIN PORT COVER PLATE~~	~~2~~	~~8A~~	~~1~~	~~*60.0*~~	~~*2.0*~~	~~*2.0*~~	~~INSTALLED BY HAND NO TOOLS~~
INSTALL AND ALIGN CLOSURE RING	5	8A	1	*60.0*	*5.0*	*5.0*	INSTALLED BY HAND NO TOOLS
PERFORM NDE ON CLOSURE RING WELDS	185	8A	1	*60.0*	*185.0*	*185.0*	MEASURED DURING WELD MOCKUP TESTING
RIG AWS TO CRANE	12	8A	1	*60.0*	*12.0*	*12.0*	10 MIN TO DISCONNECT LINES, 4 SHACKLES@2/MIN
Section 8.1.6							
REMOVE ANNULUS SHIELD	1	8A	1	*60.0*	*1.0*	*1.0*	SHIELD PLACED BY HAND

Table 10.3.1b
HI-STORM 100 SYSTEM LOADING OPERATIONS USING THE 100-TON HI-TRAC TRANSFER CASK ESTIMATED OPERATIONAL EXPOSURES† (~~42,500~~46,000 MWD/MTU, ~~5~~3-YEAR COOLED PWR FUEL)

ACTION	DURATION (MINUTES)	OPERATOR LOCATION (FIGURE 10.3.1)	NUMBER OF OPERATORS	DOSE RATE AT OPERATOR LOCATION (MREM/HR)	DOSE TO INDIVIDUAL (MREM)	TOTAL DOSE (PERSON-MREM)	ASSUMPTIONS
ATTACH DRAIN LINE TO HI-TRAC	1	9D	1	1806.3	30.1	30.1	1" THREADED FITTING NO TOOLS
POSITION HI-TRAC TOP LID	10	9B	2	60.0	10.0	20.0	VERTICAL FLANGED CONNECTION
TORQUE TOP LID BOLTS	12	9B	1	60.0	12.0	12.0	24 BOLTS AT 2/MIN (INSTALL AND TORQUE,1 PASS)
INSTALL MPC LIFT CLEATS AND MPC SLINGS	25	9A	2	247.7	103.2	206.4	INSTALL CLEATS AND HYDRO TORQUE 4 BOLTS
REMOVE TEMPORARY SHIELD RING DRAIN PLUGS	1	9B	1	60.0	1.0	1.0	8 PLUGS @ 8/MIN
REMOVE TEMPORARY SHIELD RING SEGMENTS	4	9A	1	247.7	16.5	16.5	REMOVED BY HAND NO TOOLS (8 SEGS@2/MIN)
ATTACH MPC SLINGS TO LIFT YOKE	4	9A	2	247.7	16.5	33.0	INSTALLED BY HAND NO TOOLS
POSITION HI-TRAC ABOVE TRANSFER STEP	15	9C	1	740.6	185.2	185.2	100 FT @ 10 FT/MIN (CRANE SPEED)+ 5MIN TO ALIGN
REMOVE BOTTOM LID BOLTS	6	10A	1	1806.3	180.6	180.6	36 BOLTS@6 BOLTS/MIN IMPACT TOOLS USED
INSTALL TRANSFER LID BOLTS	18	11B	1	1806.3	541.9	541.9	36 BOLTS @ 2/MIN IMPACT TOOLS USED 1 PASS
DISCONNECT MPC SLINGS	4	9A	2	247.7	16.5	33.0	INSTALLED BY HAND NO TOOLS
Section 8.1.7							
POSITION HI-TRAC ON TRANSPORT DEVICE	20	11A	2	740.6	246.9	493.7	ALIGN TRUNNIONS, DISCONNECT LIFT YOKE
TRANSPORT HI-TRAC TO OUTSIDE TRANSFER LOCATION	90	12A	3	26.4	39.6	118.8	DRIVER AND 2 SPOTTERS
ATTACH OUTSIDE LIFTING DEVICE LIFT LINKS	2	12A	2	26.4	0.9	1.8	2 LINKS@1/MIN
MATE OVERPACKS	10	13B	2	561.8	93.6	187.3	ALIGNMENT GUIDES USED
ATTACH MPC SLINGS TO MPC LIFT CLEATS	10	13A	2	247.7	41.3	82.6	2 SLINGS@5MIN/SLING NO TOOLS
REMOVE TRANSFER LID DOOR LOCKING PINS AND OPEN DOORS	4	13B	2	561.8	37.5	74.9	2 PINS@2MIN/PIN
INSTALL TRIM PLATES	4	13B	2	561.8	37.5	74.9	INSTALLED BY HAND
DISCONNECT SLINGS FROM MPC LIFTING DEVICE	10	13A	2	247.7	41.3	82.6	2 SLINGS@5MIN/SLING
REMOVE MPC LIFT CLEATS AND MPC SLINGS	10	14A	1	362.5	60.4	60.4	4 BOLTS,NO TORQUING

Table 10.3.1b
HI-STORM 100 SYSTEM LOADING OPERATIONS USING THE 100-TON HI-TRAC TRANSFER CASK ESTIMATED OPERATIONAL EXPOSURES† (~~42,500~~*46,000* MWD/MTU, ~~5~~*3*-YEAR COOLED PWR FUEL)

ACTION	DURATION (MINUTES)	OPERATOR LOCATION (FIGURE 10.3.1)	NUMBER OF OPERATORS	DOSE RATE AT OPERATOR LOCATION (MREM/HR)	DOSE TO INDIVIDUAL (MREM)	TOTAL DOSE (PERSON-MREM)	ASSUMPTIONS
INSTALL HOLE PLUGS IN EMPTY MPC BOLT HOLES	2	14A	1	*362.5*	*12.1*	*12.1*	4 PLUGS AT 2/MIN NO TORQUING
REMOVE HI-STORM VENT DUCT SHIELD INSERTS	2	15A	1	*45.5*	*1.5*	*1.5*	4 SHACKLES@2/MIN
REMOVE ALIGNMENT DEVICE	4	15A	1	*45.5*	*3.0*	*3.0*	REMOVED BY HAND NO TOOLS (4 PCS@1/MIN)
INSTALL HI-STORM LID AND INSTALL LID STUDS/NUTS	25	16A	2	*7.3*	*3.1*	*6.1*	INSTALL LID AND HYDRO TORQUE 4 BOLTS
INSTALL HI-STORM EXIT VENT GAMMA SHIELD CROSS PLATES	4	16B	1	*73.9*	*4.9*	*4.9*	4 PCS @ 1/MIN INSTALL BY HAND NO TOOLS
INSTALL TEMPERATURE ELEMENTS	20	16B	1	*73.9*	*24.6*	*24.6*	4@5MIN/TEMPERATURE ELEMENT
INSTALL EXIT VENT SCREENS	20	16B	1	*73.9*	*24.6*	*24.6*	4 SCREENS@5MIN/SCREEN
REMOVE HI-STORM LID LIFTING DEVICE	2	16A	1	*7.3*	*0.2*	*0.2*	4 SHACKLES@2/MIN
INSTALL HOLE PLUGS IN EMPTY HOLES	2	16A	1	*7.3*	*0.2*	*0.2*	4 PLUGS AT 2/MIN NO TORQUING
PERFORM SHIELDING EFFECTIVENESS TESTING	16	16D	2	*43.8*	*11.7*	*23.3*	16 POINTS@1 MIN
SECURE HI-STORM TO TRANSPORT DEVICE	10	16A	2	*7.3*	*1.2*	*2.4*	ASSUMES AIR PAD
TRANSFER HI-STORM TO ITS DESIGNATED STORAGE LOCATION	40	16C	1	*25.5*	*17.0*	*17.0*	200 FEET @ 4FT/MIN
INSERT HI-STORM LIFTING JACKS	4	16D	1	*43.8*	*2.9*	*2.9*	4 JACKS@1/MIN
REMOVE AIR PAD	5	16D	2	*43.8*	*3.6*	*7.3*	1 PAD MOVED BY HAND
REMOVE HI-STORM LIFTING JACKS	4	16D	1	*43.8*	*2.9*	*2.9*	4 JACKS@1/MIN
INSTALL INLET VENT SCREENS/CROSS PLATES	20	16D	1	*43.8*	*14.6*	*14.6*	4 SCREENS@5MIN/SCREEN
PERFORM AIR TEMPERATURE RISE TEST	8	16B	1	*73.9*	*9.8*	*9.8*	8 MEASUREMENTS@1/MIN
TOTAL						~~2755~~*5,210.8*~~22.8~~ PERSON-MREM	

Table 10.3.1c
HI-STORM 100 SYSTEM LOADING OPERATIONS USING THE 125-TON HI-TRAC 125D TRANSFER CASK
ESTIMATED OPERATIONAL EXPOSURES† (~~57,500~~*75,000* MWD/MTU, ~~125~~-YEAR COOLED PWR FUEL)

ACTION	DURATION (MINUTES)	OPERATOR LOCATION (FIGURE 10.3.1)	NUMBER OF OPERATORS	DOSE RATE AT OPERATOR LOCATION (MREM/HR)	DOSE TO INDIVIDUAL (MREM)	TOTAL DOSE (PERSON-MREM)	ASSUMPTIONS
Section 8.1.4							
LOAD PRE-SELECTED FUEL ASSEMBLIES INTO MPC	1020	1	2	1.0	17.0	34.0	15 MINUTES PER ASSEMBLY/68 ASSY
PERFORM POST-LOADING VISUAL VERIFICATION OF ASSEMBLY IDENTIFICATION	68	1	2	1.0	1.1	2.3	1 MINUTES PER ASSY/68 ASSY
Section 8.1.5							
INSTALL MPC LID AND ATTACH LIFT YOKE	45	2	2	*2.0*	*1.5*	*3.0*	CONSULTATION WITH CALVERT CLIFFS
RAISE HI-TRAC TO SURFACE OF SPENT FUEL POOL	20	2	2	*2.0*	*0.7*	*1.3*	40 FEET @ 2 FT/MINUTE (CRANE SPEED)
SURVEY MPC LID FOR HOT PARTICLES	3	3A	1	*31.1*	*1.6*	*1.6*	TELESCOPING DETECTOR USED
VERIFY MPC LID IS SEATED	0.5	3A	1	*31.1*	*0.3*	*0.3*	VISUAL VERIFICATION FROM 3 METERS
INSTALL LID RETENTION SYSTEM BOLTS	6	3B	2	*46.4*	*4.6*	*9.3*	24 BOLTS @ 1/PERSON-MINUTE
REMOVE HI-TRAC FROM SPENT FUEL POOL	8.5	3C	1	*117.8*	*16.7*	*16.7*	17 FEET @ 2 FT/MIN (CRANE SPEED)
DECONTAMINATE HI-TRAC BOTTOM	10	3D	1	*142.0*	*23.7*	*23.7*	LONG HANDLED TOOLS, PRELIMINARY DECON
TAKE SMEARS OF HI-TRAC EXTERIOR SURFACES	5	5B	1	*185.3*	*15.4*	*15.4*	50 SMEARS @ 10 SMEARS/MINUTE
DISCONNECT ANNULUS OVERPRESSURE SYSTEM	0.5	5C	1	*82.7*	*0.7*	*0.7*	QUICK DISCONNECT COUPLING
SET HI-TRAC IN CASK PREPARATION AREA	10	4A	1	*46.4*	*7.7*	*7.7*	100 FT @ 10 FT/MIN (CRANE SPEED)
REMOVE NEUTRON SHIELD JACKET FILL PLUG	2	4A	1	*46.4*	*1.5*	*1.5*	SINGLE PLUG, NO SPECIAL TOOLS
INSTALL NEUTRON SHIELD JACKET FILL PLUG	2	5B	1	*185.3*	*6.2*	*6.2*	SINGLE PLUG, NO SPECIAL TOOLS

† See notes at bottom of Table 10.3.4.

Table 10.3.1c
HI-STORM 100 SYSTEM LOADING OPERATIONS USING THE 125-TON HI-TRAC 125D TRANSFER CASK ESTIMATED OPERATIONAL EXPOSURES† (~~57,500~~*75,000* MWD/MTU, ~~125~~-YEAR COOLED PWR FUEL)

ACTION	DURATION (MINUTES)	OPERATOR LOCATION (FIGURE 10.3.1)	NUMBER OF OPERATORS	DOSE RATE AT OPERATOR LOCATION (MREM/HR)	DOSE TO INDIVIDUAL (MREM)	TOTAL DOSE (PERSON-MREM)	ASSUMPTIONS
DISCONNECT LID RETENTION SYSTEM	6	5A	2	*37.3*	*3.7*	*7.5*	24 BOLTS @ 1 BOLT/PERSON MINUTES
MEASURE DOSE RATES AT MPC LID	3	5A	1	*37.3*	*1.9*	*1.9*	TELESCOPING DETECTOR USED
DECONTAMINATE AND SURVEY HI-TRAC	103	5B	1	*185.3*	*318.1*	*318.1*	490 SQ-FT@5 SQ-FT/PRERSON-MINUTE+50 SMEARS@10 SMEARS/MINUTE
INSTALL TEMPORARY SHIELD	16	6A	2	*18.7*	*5.0*	*10.0*	8 SEGMENTS @ 1 SEGMENT/PERSON MIN
FILL TEMPORARY SHIELD RING	25	6A	1	*18.7*	*7.8*	*7.8*	230 GAL @10GPM, LONG HANDLED SPRAY WAND
ATTACH DRAIN LINE TO HI-TRAC DRAIN PORT	0.5	5C	1	*82.7*	*0.7*	*0.7*	QUICK DISCONNECT COUPLING
INSTALL RVOAs	2	6A	1	*18.7*	*0.6*	*0.6*	SINGLE THREADED CONNECTION X 2 RVOAs
ATTACH WATER PUMP TO DRAIN PORT	2	6A	1	*18.7*	*0.6*	*0.6*	POSITION PUMP SELF PRIMING
DISCONNECT WATER PUMP	5	6A	1	*18.7*	*1.6*	*1.6*	DRAIN HOSES MOVE PUMP
DECONTAMINATE MPC LID TOP SURFACE AND SHELL AREA ABOVE INFLATABLE ANNULUS SEAL	6	6A	1	*18.7*	*1.9*	*1.9*	30 SQ-FT @5 SQ-FT/MINUTE+10 SMEARS@10 SMEARS/MINUTE
REMOVE INFLATABLE ANNULUS SEAL	3	6A	1	*18.7*	*0.9*	*0.9*	SEAL PULLS OUT DIRECTLY
SURVEY MPC LID TOP SURFACES AND ACCESSIBLE AREAS OF TOP THREE INCHES OF MPC SHELL	1	6A	1	*18.7*	*0.3*	*0.3*	10 SMEARS@10 SMEARS/MINUTE
INSTALL ANNULUS SHIELD	2	6A	1	*18.7*	*0.6*	*0.6*	SHIELD PLACED BY HAND
CENTER LID IN MPC SHELL	20	6A	3	*18.7*	*6.2*	*18.7*	CONSULTATION WITH CALVERT CLIFFS
INSTALL MPC LID SHIMS	12	6A	2	*18.7*	*3.7*	*7.5*	MEASURED DURING WELD MOCKUP TESTING
POSITION AWS BASEPLATE SHIELD ON MPC LID	20	7A	2	*18.7*	*6.2*	*12.5*	ALIGN AND REMOVE 4 SHACKLES
INSTALL AUTOMATED WELDING SYSTEM ROBOT	8	7A	2	*18.7*	*2.5*	*5.0*	ALIGN AND REMOVE 4 SHACKLES/4 QUICK CONNECTS@1/MIN

Table 10.3.1c
HI-STORM 100 SYSTEM LOADING OPERATIONS USING THE 125-TON HI-TRAC 125D TRANSFER CASK ESTIMATED OPERATIONAL EXPOSURES† (~~57,500~~*75,000* MWD/MTU, ~~125~~-YEAR COOLED PWR FUEL)

ACTION	DURATION (MINUTES)	OPERATOR LOCATION (FIGURE 10.3.1)	NUMBER OF OPERATORS	DOSE RATE AT OPERATOR LOCATION (MREM/HR)	DOSE TO INDIVIDUAL (MREM)	TOTAL DOSE (PERSON-MREM)	ASSUMPTIONS
PERFORM NDE OF LID WELD	230	7A	1	*18.7*	*71.7*	*71.7*	MEASURED DURING WELD MOCKUP TESTING
ATTACH DRAIN LINE TO VENT PORT	1	7A	1	*18.7*	*0.3*	*0.3*	1" THREADED FITTING NO TOOLS
VISUALLY EXAMINE MPC LID-TO-SHELL WELD FOR LEAKAGE OF WATER	10	7A	1	*18.7*	*3.1*	*3.1*	10 MIN TEST DURATION
DISCONNECT WATER FILL LINE AND DRAIN LINE	2	7A	1	*18.7*	*0.6*	*0.6*	1" THREADED FITTING NO TOOLS X 2
REPEAT LIQUID PENETRANT EXAMINATION ON MPC LID FINAL PASS	45	7A	1	*18.7*	*14.0*	*14.0*	5 MIN TO APPLY, 7 MIN TO WIPE, 5 APPLY DEV, INSP (24 IN/MIN)
ATTACH GAS SUPPLY TO VENT PORT	1	7A	1	*18.7*	*0.3*	*0.3*	1" THREADED FITTING NO TOOLS
ATTACH DRAIN LINE TO DRAIN PORT	1	7A	1	*18.7*	*0.3*	*0.3*	1" THREADED FITTING NO TOOLS
Deleted ~~CONNECT MSLD SNIFFER TO AUTOMATED WELDING SYSTEM~~	4	~~8A~~	~~1~~	~~*37.9*~~	~~*2.5*~~	~~*2.5*~~	~~SIMPLE ATTACHMENT NO TOOLS~~
Deleted ~~DISCONNECT MSLD SNIFFER FROM AUTOMATED WELDING SYSTEM~~	4	~~8A~~	~~1~~	~~*37.9*~~	~~*2.5*~~	~~*2.5*~~	~~SIMPLE ATTACHMENT NO TOOLS~~
ATTACH DRAIN LINE TO VENT PORT	1	8A	1	*37.9*	*0.6*	*0.6*	1" THREADED FITTING NO TOOLS
ATTACH WATER FILL LINE TO DRAIN PORT	1	8A	1	*37.9*	*0.6*	*0.6*	1" THREADED FITTING NO TOOLS
DISCONNECT WATER FILL DRAIN LINES FROM MPC	2	8A	1	*37.9*	*1.3*	*1.3*	1" THREADED FITTING NO TOOLS X 2
ATTACH HELIUM SUPPLY TO VENT PORT	1	8A	1	*37.9*	*0.6*	*0.6*	1" THREADED FITTING NO TOOLS
ATTACH DRAIN LINE TO DRAIN PORT	1	8A	1	*37.9*	*0.6*	*0.6*	1" THREADED FITTING NO TOOLS
DISCONNECT GAS SUPPLY LINE FROM MPC	1	8A	1	*37.9*	*0.6*	*0.6*	1" THREADED FITTING NO TOOLS
DISCONNECT DRAIN LINE FROM MPC	1	8A	1	*37.9*	*0.6*	*0.6*	1" THREADED FITTING NO TOOLS

Table 10.3.1c

HI-STORM 100 SYSTEM LOADING OPERATIONS USING THE 125-TON HI-TRAC 125D TRANSFER CASK ESTIMATED OPERATIONAL EXPOSURES† (~~57,500~~*75,000* MWD/MTU, ~~125~~-YEAR COOLED PWR FUEL)

ACTION	DURATION (MINUTES)	OPERATOR LOCATION (FIGURE 10.3.1)	NUMBER OF OPERATORS	DOSE RATE AT OPERATOR LOCATION (MREM/HR)	DOSE TO INDIVIDUAL (MREM)	TOTAL DOSE (PERSON-MREM)	ASSUMPTIONS
ATTACHM MOISTURE REMOVAL SYSTEM TO VENT AND DRAIN PORT RVOAs	2	8A	1	*37.9*	*1.3*	*1.3*	1" THREADED FITTING NO TOOLS
DISCONNECT MOISTURE REMOVAL SYSTEM FROM MPC	2	8A	1	*37.9*	*1.3*	*1.3*	1" THREADED FITTING NO TOOLS X 2
CLOSE DRAIN PORT RVOA CAP AND REMOVE DRAIN PORT RVOA	1.5	8A	1	*37.9*	*0.9*	*0.9*	SINGLE THREADED CONNECTION (1 RVOA)
ATTACH HELIUM BACKFILL SYSTEM TO VENT PORT	1	8A	1	*37.9*	*0.6*	*0.6*	1" THREADED FITTING NO TOOLS
DISCONNECT HBS FROM MPC	1	8A	1	*37.9*	*0.6*	*0.6*	1" THREADED FITTING NO TOOLS
CLOSE VENT PORT RVOA AND DISCONNECT VENT PORT RVOA	1.5	8A	1	*37.9*	*0.9*	*0.9*	SINGLE THREADED CONNECTION (1 RVOA)
WIPE INSIDE AREA OF VENT AND DRAIN PORT RECESSES	2	8A	1	*37.9*	*1.3*	*1.3*	2 PORTS, 1 MIN/PORT
PLACE COVER PLATE OVER VENT PORT RECESS	1	8A	1	*37.9*	*0.6*	*0.6*	INSTALLED BY HAND NO TOOLS (2/MIN)
PERFORM NDE ON VENT AND DRAIN COVER PLATE WELD	100	8A	1	*37.9*	*63.2*	*63.2*	MEASURED DURING WELD MOCKUP TESTING
~~FLUSH CAVITY WITH HELIUM AND~~ INSTALL SET SCREWS	2	8A	1	*37.9*	*1.3*	*1.3*	4 SET SCREWS @2/MINUTE
PLUG WELD OVER SET SCREWS	8	8A	1	*37.9*	*5.1*	*5.1*	FOUR SINGLE SPOT WELDS @ 1 PER 2 MINTES
Deleted ~~INSTALL MSLD OVER VENT PORT COVER PLATE~~	~~2~~	~~8A~~	~~1~~	~~*37.9*~~	~~*1.3*~~	~~*1.3*~~	~~INSTALLED BY HAND NO TOOLS~~
Deleted ~~INSTALL MSLD OVER DRAIN PORT COVER PLATE~~	~~2~~	~~8A~~	~~1~~	~~*37.9*~~	~~*1.3*~~	~~*1.3*~~	~~INSTALLED BY HAND NO TOOLS~~
INSTALL AND ALIGN CLOSURE RING	5	8A	1	*37.9*	*3.2*	*3.2*	INSTALLED BY HAND NO TOOLS
PERFORM A NDE ON CLOSURE RING WELDS	185	8A	1	*37.9*	*116.9*	*116.9*	MEASURED DURING WELD MOCKUP TESTING
RIG AWS TO CRANE	12	8A	1	*37.9*	*7.6*	*7.6*	10 MIN TO DISCONNECT LINES, 4 SHACKLES@2/MIN

Table 10.3.1c
HI-STORM 100 SYSTEM LOADING OPERATIONS USING THE 125-TON HI-TRAC 125D TRANSFER CASK ESTIMATED OPERATIONAL EXPOSURES† (~~57,500~~*75,000* MWD/MTU, ~~125~~-YEAR COOLED PWR FUEL)

ACTION	DURATION (MINUTES)	OPERATOR LOCATION (FIGURE 10.3.1)	NUMBER OF OPERATORS	DOSE RATE AT OPERATOR LOCATION (MREM/HR)	DOSE TO INDIVIDUAL (MREM)	TOTAL DOSE (PERSON-MREM)	ASSUMPTIONS
Section 8.1.6							
REMOVE ANNULUS SHIELD	1	8A	1	*37.9*	*0.6*	*0.6*	SHIELD PLACED BY HAND
ATTACH DRAIN LINE TO HI-TRAC	1	9D	1	*354.2*	*5.9*	*5.9*	1" THREADED FITTING NO TOOLS
POSITION HI-TRAC TOP LID	10	9B	2	*37.9*	*6.3*	*12.6*	VERTICAL FLANGED CONNECTION
TORQUE TOP LID BOLTS	12	9B	1	*37.9*	*7.6*	*7.6*	24 BOLTS AT 2/MIN (INSTALL AND TORQUE,1 PASS)
INSTALL MPC LIFT CLEATS AND MPC SLINGS	25	9A	2	*158.5*	*66.0*	*132.1*	INSTALL CLEATS AND HYDRO TORQUE 4 BOLTS
REMOVE TEMPORARY SHIELD RING DRAIN PLUGS	1	9B	1	*37.9*	*0.6*	*0.6*	8 PLUGS @ 8/MIN
REMOVE TEMPORARY SHIELD RING SEGMENTS	4	9A	1	*158.5*	*10.6*	*10.6*	REMOVED BY HAND NO TOOLS (8 SEGS@2/MIN)
ATTACH MPC SLINGS TO LIFT YOKE	4	9A	2	*158.5*	*10.6*	*21.1*	INSTALLED BY HAND, NO TOOLS
Section 8.1.7							
POSITION HI-TRAC ON TRANSPORT DEVICE	20	11A	2	*117.8*	*39.3*	*78.5*	ALIGN TRUNNIONS, DISCONNECT LIFT YOKE
TRANSPORT HI-TRAC TO OUTSIDE TRANSFER LOCATION	90	12A	3	*26.4*	*39.6*	*118.8*	DRIVER AND 2 SPOTTERS
ATTACH OUTSIDE LIFTING DEVICE LIFT LINKS	2	12A	2	*26.4*	*0.9*	*1.8*	2 LINKS@1/MIN
MATE OVERPACKS	10	13B	2	*118.5*	*19.8*	*39.5*	ALIGNMENT GUIDES USED
ATTACH MPC LIFT SLINGS TO MPC LIFT CLEATS	10	13A	2	*158.5*	*26.4*	*52.8*	2 SLINGS@5MIN/SLING NO TOOLS
REMOVE MATING DEVICE LOCKING PINS AND OPEN DRAWER	40	13B	2	*118.5*	*79.0*	*158.0*	2 PINS@2MIN/PIN
INSTALL TRIM PLATES	4	13B	2	*118.5*	*7.9*	*15.8*	INSTALLED BY HAND
DISCONNECT SLINGS FROM MPC LIFTING DEVICE	10	13A	2	*158.5*	*26.4*	*52.8*	2 SLINGS@5MIN/SLING
REMOVE MPC LIFT CLEATS AND MPC LIFT SLINGS	10	14A	1	*362.5*	*60.4*	*60.4*	4 BOLTS,NO TORQUING

Table 10.3.1c
HI-STORM 100 SYSTEM LOADING OPERATIONS USING THE 125-TON HI-TRAC 125D TRANSFER CASK ESTIMATED OPERATIONAL EXPOSURES† (~~57,500~~*75,000* MWD/MTU, ~~125~~-YEAR COOLED PWR FUEL)

ACTION	DURATION (MINUTES)	OPERATOR LOCATION (FIGURE 10.3.1)	NUMBER OF OPERATORS	DOSE RATE AT OPERATOR LOCATION (MREM/HR)	DOSE TO INDIVIDUAL (MREM)	TOTAL DOSE (PERSON-MREM)	ASSUMPTIONS
INSTALL HOLE PLUGS IN EMPTY MPC BOLT HOLES	2	14A	1	*362.5*	*12.1*	*12.1*	4 PLUGS AT 2/MIN NO TORQUING
REMOVE HI-STORM VENT DUCT SHIELD INSERTS	2	15A	1	*45.5*	*1.5*	*1.5*	4 SHACKLES@2/MIN
REMOVE MATING DEVICE	10	15A	1	*45.5*	*7.6*	*7.6*	3 BOLTS @ 2 MINUTES PER BOLT
INSTALL HI-STORM LID AND INSTALL LID STUDS/NUTS	25	16A	2	*7.3*	*3.1*	*6.1*	INSTALL LID AND HYDRO TORQUE 4 BOLTS
INSTALL HI-STORM EXIT VENT GAMMA SHIELD CROSS PLATES	4	16B	1	*73.9*	*4.9*	*4.9*	4 PCS @ 1/MIN INSTALL BY HAND NO TOOLS
INSTALL TEMPERATURE ELEMENTS	20	16B	1	*73.9*	*24.6*	*24.6*	4@5MIN/TEMPERATURE ELEMENT
INSTALL EXIT VENT SCREENS	20	16B	1	*73.9*	*24.6*	*24.6*	4 SCREENS@5MIN/SCREEN
REMOVE HI-STORM LID LIFTING DEVICE	2	16A	1	*7.3*	*0.2*	*0.2*	4 SHACKLES@2/MIN
INSTALL HOLE PLUGS IN EMPTY HOLES	2	16A	1	*7.3*	*0.2*	*0.2*	4 PLUGS AT 2/MIN NO TORQUING
PERFORM SHIELDING EFFECTIVENESS TESTING	16	16D	2	*43.8*	*11.7*	*23.3*	16 POINTS@1 MIN
SECURE HI-STORM TO TRANSPORT DEVICE	10	16A	2	*7.3*	*1.2*	*2.4*	ASSUMES AIR PAD
TRANSFER HI-STORM TO ITS DESIGNATED STORAGE LOCATION	40	16C	1	*25.5*	*17.0*	*17.0*	200 FEET @ 4FT/MIN
INSERT HI-STORM LIFTING JACKS	4	16D	1	*43.8*	*2.9*	*2.9*	4 JACKS@1/MIN
REMOVE AIR PAD	5	16D	2	*43.8*	*3.6*	*7.3*	1 PAD MOVED BY HAND
REMOVE HI-STORM LIFTING JACKS	4	16D	1	*43.8*	*2.9*	*2.9*	4 JACKS@1/MIN
INSTALL INLET VENT SCREENS/CROSS PLATES	20	16D	1	*43.8*	*14.6*	*14.6*	4 SCREENS@5MIN/SCREEN
PERFORM AIR TEMPERATURE RISE TEST	8	16B	1	*73.9*	*9.8*	*9.8*	8 MEASUREMENTS@1/MIN
TOTAL							~~779.3~~*1751.7*~~9.3~~ PERSON-MREM

Table 10.3.2a
HI-STORM 100 SYSTEM UNLOADING OPERATIONS USING THE 125-TON HI-TRAC TRANSFER CASK ESTIMATED OPERATIONAL EXPOSURES† (~~57,500~~*75,000* MWD/MTU, ~~*125*~~-YEAR COOLED PWR FUEL)

ACTION	DURATION (MINUTES)	OPERATOR LOCATION (FIGURE 10.3.1)	NUMBER OF OPERATORS	DOSE RATE AT OPERATOR LOCATION (MREM/HR)	DOSE TO INDIVIDUAL (MREM)	TOTAL DOSE (PERSON-MREM)	ASSUMPTIONS
Section 8.3.2 (Step Sequence Varies By Site and Mode of Transport)							
REMOVE INLET VENT SCREENS	20	16D	1	*43.8*	*14.6*	*14.6*	4 SCREENS@5MIN/SCREEN
INSERT HI-STORM LIFTING JACKS	4	16D	1	*43.8*	*2.9*	*2.9*	4 JACKS@1/MIN
INSERT AIR PAD	5	16D	2	*43.8*	*3.6*	*7.3*	1 PAD MOVED BY HAND
REMOVE HI-STORM LIFTING JACKS	4	16D	1	*43.8*	*2.9*	*2.9*	4 JACKS@1/MIN
TRANSFER HI-STORM TO MPC TRANSFER LOCATION	40	16C	1	*25.5*	*17.0*	*17.0*	200 FEET @ 4FT/MIN
REMOVE HI-STORM LID STUDS/NUTS	10	16A	1	*7.3*	*1.2*	*1.2*	4 BOLTS NO TORQUE
REMOVE HI-STORM LID LIFTING HOLE PLUGS AND INSTALL LID LIFTING SLING	2	16A	1	*7.3*	*0.2*	*0.2*	4 PLUGS AT 2/MIN NO TORQUING
REMOVE GAMMA SHIELD CROSS PLATES	4	16B	1	*73.9*	*4.9*	*4.9*	4 PLATES@1/MIN
REMOVE TEMPERATURE ELEMENTS	8	16B	1	*73.9*	*9.8*	*9.8*	4 TEMP. ELEMENTS @ 2MIN/TEMP. ELEMENT NO TORQUE
REMOVE HI-STORM LID	2	16A	1	*7.3*	*0.2*	*0.2*	4 SHACKLES@2/MIN
INSTALL HI-STORM VENT DUCT SHIELD INSERTS	2	15A	1	*45.5*	*1.5*	*1.5*	4 SHACKLES@2/MIN
INSTALL ALIGNMENT DEVICE	4	15A	1	*45.5*	*3.0*	*3.0*	REMOVED BY HAND NO TOOLS (4 PCS@1/MIN)
REMOVE MPC LIFT CLEAT HOLE PLUGS	2	14A	1	*362.5*	*12.1*	*12.1*	4 PLUGS AT 2/MIN NO TORQUING
INSTALL MPC LIFT CLEATS AND MPC SLINGS	2	14A	1	*362.5*	*12.1*	*12.1*	4 PLUGS AT 2/MIN NO TORQUING
ALIGN HI-TRAC OVER HI-STORM AND MATE OVERPACKS	10	13B	2	*118.5*	*19.8*	*39.5*	ALIGNMENT GUIDES USED
PULL MPC SLINGS THROUGH TOP LID HOLE	10	13A	2	*158.5*	*26.4*	*52.8*	2 SLINGS@5MIN/SLING
INSTALL TRIM PLATES	4	13B	2	*118.5*	*7.9*	*15.8*	INSTALLED BY HAND NO FASTENERS

† See notes at bottom of Table 10.3.4.

Table 10.3.2a
HI-STORM 100 SYSTEM UNLOADING OPERATIONS USING THE 125-TON HI-TRAC TRANSFER CASK ESTIMATED OPERATIONAL EXPOSURES† (~~57,500~~*75,000* MWD/MTU, ~~125~~-YEAR COOLED PWR FUEL)

ACTION	DURATION (MINUTES)	OPERATOR LOCATION (FIGURE 10.3.1)	NUMBER OF OPERATORS	DOSE RATE AT OPERATOR LOCATION (MREM/HR)	DOSE TO INDIVIDUAL (MREM)	TOTAL DOSE (PERSON-MREM)	ASSUMPTIONS
ATTACH MPC SLING TO LIFTING DEVICE	10	13A	1	*158.5*	*26.4*	*26.4*	2 SLINGS@5MIN/SLING NO BOLTING
CLOSE HI-TRAC DOORS AND INSTALL DOOR LOCKING PINS	4	13B	2	*118.5*	*7.9*	*15.8*	2 PINS@2MIN/PIN
DISCONNECT SLINGS FROM MPC LIFT CLEATS	10	13A	2	*158.5*	*26.4*	*52.8*	2 SLINGS@5MIN/SLING
DOWNEND HI-TRAC ON TRANSPORT FRAME	20	12A	2	*26.4*	*8.8*	*17.6*	ALIGN TRUNNIONS, DISCONNECT LIFT YOKE
TRANSPORT HI-TRAC TO FUEL BUILDING	90	12A	1	*26.4*	*39.6*	*39.6*	DRIVER RECEIVES MOST DOSE
UPEND HI-TRAC	20	12A	2	*26.4*	*8.8*	*17.6*	ALIGN TRUNNIONS, DISCONNECT LIFT YOKE
Section 8.3.3							
MOVE HI-TRAC TO TRANSFER SLIDE	20	11A	2	*117.8*	*39.3*	*78.5*	ALIGN TRUNNIONS, DISCONNECT LIFT YOKE
ATTACH MPC SLINGS	4	9A	2	*158.5*	*10.6*	*21.1*	INSTALLED BY HAND NO TOOLS
REMOVE TRANSFER LID BOLTS	6	11B	1	*354.2*	*35.4*	*35.4*	36 BOLTS@6 BOLTS/MIN IMPACT TOOLS USED
INSTALL POOL LID BOLTS	18	10A	1	*354.2*	*106.3*	*106.3*	36 BOLTS @ 2/MIN IMPACT TOOLS USED 1 PASS
DISCONNECT MPC SLINGS AND LIFT CLEATS	10	9A	1	*158.5*	*26.4*	*26.4*	4 BOLTS,NO TORQUING
PLACE HI-TRAC IN PREPARATION AREA	15	9C	1	*117.8*	*29.5*	*29.5*	100 FT @ 10 FT/MIN (CRANE SPEED)+ 5MIN TO ALIGN
REMOVE TOP LID BOLTS	6	9B	1	*37.9*	*3.8*	*3.8*	24 BOLTS AT 4/MIN (NO TORQUE IMPACT TOOLS)
REMOVE HI-TRAC TOP LID	2	6A	1	*18.7*	*0.6*	*0.6*	4 SHACKLES@2/MIN
ATTACH WATER FILL LINE TO HI-TRAC DRAIN PORT	0.5	9D	1	*354.2*	*3.0*	*3.0*	QUICK DISCONNECT NO TOOLS
INSTALL BOLT PLUGS OR WATERPROOF TAPE FROM HI-TRAC TOP BOLT HOLES	9	8A	1	*37.9*	*5.7*	*5.7*	18 HOLES@2/MIN
CORE DRILL CLOSURE RING AND VENT AND DRAIN PORT COVER PLATES	40	7A	2	*18.7*	*12.5*	*24.9*	20 MINUTES TO INSTALL/ALIGN +10 MIN/COVER

Table 10.3.2a
HI-STORM 100 SYSTEM UNLOADING OPERATIONS USING THE 125-TON HI-TRAC TRANSFER CASK ESTIMATED OPERATIONAL EXPOSURES† (~~57,500~~*75,000* MWD/MTU, ~~1~~*25*-YEAR COOLED PWR FUEL)

ACTION	DURATION (MINUTES)	OPERATOR LOCATION (FIGURE 10.3.1)	NUMBER OF OPERATORS	DOSE RATE AT OPERATOR LOCATION (MREM/HR)	DOSE TO INDIVIDUAL (MREM)	TOTAL DOSE (PERSON-MREM)	ASSUMPTIONS
REMOVE CLOSURE RING SECTION AND VENT AND DRAIN PORT COVER PLATES	1	8A	1	*37.9*	*0.6*	*0.6*	2 COVERS@2/MIN NO TOOLS
ATTACH RVOAS	2	8A	1	*37.9*	*1.3*	*1.3*	SINGLE THREADED CONNECTION (1 RVOA)
ATTACH A SAMPLE BOTTLE TO VENT PORT RVOA	0.5	8A	1	*37.9*	*0.3*	*0.3*	1" THREADED FITTING NO TOOLS
GATHER A GAS SAMPLE FROM MPC	0.5	8A	1	*37.9*	*0.3*	*0.3*	SMALL BALL VALVE
CLOSE VENT PORT CAP AND DISCONNECT SAMPLE BOTTLE	1	8A	1	*37.9*	*0.6*	*0.6*	1" THREADED FITTING NO TOOLS
ATTACH COOL-DOWN SYSTEM TO RVOAs	2	8A	1	*37.9*	*1.3*	*1.3*	1" THREADED FITTING NO TOOLS X 2
DISCONNECT GAS LINES TO VENT AND DRAIN PORT RVOAs	1	8A	1	*37.9*	*0.6*	*0.6*	1" THREADED FITTING NO TOOLS
VACUUM TOP SURFACES OF MPC AND HI-TRAC	10	6A	1	*18.7*	*3.1*	*3.1*	SHOP VACUUM WITH WAND + HAND WIPE
REMOVE ANNULUS SHIELD	1	8A	1	*37.9*	*0.6*	*0.6*	SHIELD PLACED BY HAND
MANUALLY INSTALL INFLATABLE SEAL	10	6A	2	*18.7*	*3.1*	*6.2*	CONSULTATION WITH CALVERT CLIFFS
OPEN NEUTRON SHIELD JACKET DRAIN VALVE	2	5C	1	*82.7*	*2.8*	*2.8*	SINGLE THREADED CONNECTION
CLOSE NEUTRON SHIELD JACKET DRAIN VALVE	2	5C	1	*82.7*	*2.8*	*2.8*	SINGLE THREADED CONNECTION
REMOVE MPC LID LIFTING HOLE PLUGS	2	5A	1	*37.3*	*1.2*	*1.2*	4 PLUGS AT 2/MIN NO TORQUING
ATTACH LID RETENTION SYSTEM	12	5A	1	*37.3*	*7.5*	*7.5*	24 BOLTS @ 2 MINUTES/BOLT
ATTACH ANNULUS OVERPRESSURE SYSTEM	0.5	5C	1	*82.7*	*0.7*	*0.7*	QUICK DISCONNECT NO TOOLS
POSITION HI-TRAC OVER CASK LOADING AREA	10	5C	1	*82.7*	*13.8*	*13.8*	100 FT @ 10 FT/MIN (CRANE SPEED)
LOWER HI-TRAC INTO SPENT FUEL POOL	8.5	3C	1	*117.8*	*16.7*	*16.7*	17 FEET @ 2 FT/MIN (CRANE SPEED)
REMOVE LID RETENTION BOLTS	12	3B	1	*46.4*	*9.3*	*9.3*	24 BOLTS @ 2/MINUTE
PLACE HI-TRAC ON FLOOR	20	2	2	*2.0*	*0.7*	*1.3*	40 FEET @ 2 FT/MINUTE (CRANE SPEED)

Table 10.3.2a
HI-STORM 100 SYSTEM UNLOADING OPERATIONS USING THE 125-TON HI-TRAC TRANSFER CASK ESTIMATED OPERATIONAL EXPOSURES† (~~57,500~~*75,000* MWD/MTU, ~~125~~-YEAR COOLED PWR FUEL)

ACTION	DURATION (MINUTES)	OPERATOR LOCATION (FIGURE 10.3.1)	NUMBER OF OPERATORS	DOSE RATE AT OPERATOR LOCATION (MREM/HR)	DOSE TO INDIVIDUAL (MREM)	TOTAL DOSE (PERSON-MREM)	ASSUMPTIONS
REMOVE MPC LID	20	2	2	*2.0*	*0.7*	*1.3*	CONSULTATION WITH CALVERT CLIFFS
Section 8.3.4							
REMOVE SPENT FUEL ASSEMBLIES FROM MPC	1020	1	2	*1.0*	*17.0*	*34.0*	15 MINUTES PER ASSEMBLY/68 ASSY
TOTAL						~~380.0~~*809.5* PERSON-MREM	

Table 10.3.2b
HI-STORM 100 SYSTEM UNLOADING OPERATIONS USING THE 100-TON HI-TRAC TRANSFER CASK ESTIMATED OPERATIONAL EXPOSURES† (~~42,500~~*46,000* MWD/MTU, ~~5~~3-YEAR COOLED PWR FUEL)

ACTION	DURATION (MINUTES)	OPERATOR LOCATION (FIGURE 10.3.1)	NUMBER OF OPERATORS	DOSE RATE AT OPERATOR LOCATION (MREM/HR)	DOSE TO INDIVIDUAL (MREM)	TOTAL DOSE (PERSON-MREM)	ASSUMPTIONS
Section 8.3.2 (Step Sequence Varies By Site and Mode of Transport							
REMOVE INLET VENT SCREENS	20	16D	1	*43.8*	*14.6*	*14.6*	4 SCREENS@5MIN/SCREEN
INSERT HI-STORM LIFTING JACKS	4	16D	1	*43.8*	*2.9*	*2.9*	4 JACKS@1/MIN
INSERT AIR PAD	5	16D	2	*43.8*	*3.6*	*7.3*	1 PAD MOVED BY HAND
REMOVE HI-STORM LIFTING JACKS	4	16D	1	*43.8*	*2.9*	*2.9*	4 JACKS@1/MIN
TRANSFER HI-STORM TO MPC TRANSFER LOCATION	40	16C	1	*25.5*	*17.0*	*17.0*	200 FEET @ 4FT/MIN
REMOVE HI-STORM LID STUDS/NUTS	10	16A	1	*7.3*	*1.2*	*1.2*	4 BOLTS NO TORQUE
REMOVE HI-STORM LID LIFTING HOLE PLUGS AND INSTALL LID LIFTING SLING	2	16A	1	*7.3*	*0.2*	*0.2*	4 PLUGS AT 2/MIN NO TORQUING
REMOVE GAMMA SHIELD CROSS PLATES	4	16B	1	*73.9*	*4.9*	*4.9*	4 PLATES@1/MIN
REMOVE TEMPERATURE ELEMENTS	8	16B	1	*73.9*	*9.8*	*9.8*	4 TEMP. ELEMENTS @ 2MIN/TEMP. ELEMENT NO TORQUE
REMOVE HI-STORM LID	2	16A	1	*7.3*	*0.2*	*0.2*	4 SHACKLES@2/MIN
INSTALL HI-STORM VENT DUCT SHIELD INSERTS	2	15A	1	*45.5*	*1.5*	*1.5*	4 SHACKLES@2/MIN
INSTALL ALIGNMENT DEVICE	4	15A	1	*45.5*	*3.0*	*3.0*	REMOVED BY HAND NO TOOLS (4 PCS@1/MIN)
REMOVE MPC LIFT CLEAT HOLE PLUGS	2	14A	1	*362.5*	*12.1*	*12.1*	4 PLUGS AT 2/MIN NO TORQUING
INSTALL MPC LIFT CLEATS AND MPC SLINGS	2	14A	1	*362.5*	*12.1*	*12.1*	4 PLUGS AT 2/MIN NO TORQUING
ALIGN HI-TRAC OVER HI-STORM AND MATE OVERPACKS	10	13B	2	*561.8*	*93.6*	*187.3*	ALIGNMENT GUIDES USED
PULL MPC SLINGS THROUGH TOP LID HOLE	10	13A	2	*247.7*	*41.3*	*82.6*	2 SLINGS@5MIN/SLING
INSTALL TRIM PLATES	4	13B	2	*561.8*	*37.5*	*74.9*	INSTALLED BY HAND NO FASTENERS

† See notes at bottom of Table 10.3.4.

Table 10.3.2b
HI-STORM 100 SYSTEM UNLOADING OPERATIONS USING THE 100-TON HI-TRAC TRANSFER CASK ESTIMATED OPERATIONAL EXPOSURES† (~~42,500~~*46,000* MWD/MTU, ~~5~~*3*-YEAR COOLED PWR FUEL)

ACTION	DURATION (MINUTES)	OPERATOR LOCATION (FIGURE 10.3.1)	NUMBER OF OPERATORS	DOSE RATE AT OPERATOR LOCATION (MREM/HR)	DOSE TO INDIVIDUAL (MREM)	TOTAL DOSE (PERSON-MREM)	ASSUMPTIONS
ATTACH MPC SLING TO LIFTING DEVICE	10	13A	1	*247.7*	*41.3*	*41.3*	2 SLINGS@5MIN/SLING NO BOLTING
CLOSE HI-TRAC DOORS AND INSTALL DOOR LOCKING PINS	4	13B	2	*561.8*	*37.5*	*74.9*	2 PINS@2MIN/PIN
DISCONNECT SLINGS FROM MPC LIFT CLEATS	10	13A	2	*247.7*	*41.3*	*82.6*	2 SLINGS@5MIN/SLING
DOWNEND HI-TRAC ON TRANSPORT FRAME	20	12A	2	*26.4*	*8.8*	*17.6*	ALIGN TRUNNIONS, DISCONNECT LIFT YOKE
TRANSPORT HI-TRAC TO FUEL BUILDING	90	12A	1	*26.4*	*39.6*	*39.6*	DRIVER RECEIVES MOST DOSE
UPEND HI-TRAC	20	12A	2	*26.4*	*8.8*	*17.6*	ALIGN TRUNNIONS, DISCONNECT LIFT YOKE
Section 8.3.3							
MOVE HI-TRAC TO TRANSFER SLIDE	20	11A	2	*740.6*	*246.9*	*493.7*	ALIGN TRUNNIONS, DISCONNECT LIFT YOKE
ATTACH MPC SLINGS	4	9A	2	*247.7*	*16.5*	*33.0*	INSTALLED BY HAND NO TOOLS
REMOVE TRANSFER LID BOLTS	6	11B	1	*1806.3*	*180.6*	*180.6*	36 BOLTS@6 BOLTS/MIN IMPACT TOOLS USED
INSTALL POOL LID BOLTS	18	10A	1	*1806.3*	*541.9*	*541.9*	36 BOLTS @ 2/MIN IMPACT TOOLS USED 1 PASS
DISCONNECT MPC SLINGS AND LIFT CLEATS	10	9A	1	*247.7*	*41.3*	*41.3*	4 BOLTS,NO TORQUING
PLACE HI-TRAC IN PREPARATION AREA	15	9C	1	*740.6*	*185.2*	*185.2*	100 FT @ 10 FT/MIN (CRANE SPEED)+ 5MIN TO ALIGN
REMOVE TOP LID BOLTS	6	9B	1	*60.0*	*6.0*	*6.0*	24 BOLTS AT 4/MIN (NO TORQUE IMPACT TOOLS)
REMOVE HI-TRAC TOP LID	2	6A	1	*31.3*	*1.0*	*1.0*	4 SHACKLES@2/MIN
ATTACH WATER FILL LINE TO HI-TRAC DRAIN PORT	0.5	9D	1	*1806.3*	*15.1*	*15.1*	QUICK DISCONNECT NO TOOLS
INSTALL BOLT PLUGS OR WATERPROOF TAPE FROM HI-TRAC TOP BOLT HOLES	9	8A	1	*60.0*	*9.0*	*9.0*	18 HOLES@2/MIN
CORE DRILL CLOSURE RING AND VENT AND DRAIN PORT COVER PLATES	40	7A	2	*31.3*	*20.9*	*41.7*	20 MINUTES TO INSTALL/ALIGN +10 MIN/COVER

Table 10.3.2b
HI-STORM 100 SYSTEM UNLOADING OPERATIONS USING THE 100-TON HI-TRAC TRANSFER CASK
ESTIMATED OPERATIONAL EXPOSURES† (~~42,500~~*46,000* MWD/MTU, *5*3-YEAR COOLED PWR FUEL)

ACTION	DURATION (MINUTES)	OPERATOR LOCATION (FIGURE 10.3.1)	NUMBER OF OPERATORS	DOSE RATE AT OPERATOR LOCATION (MREM/HR)	DOSE TO INDIVIDUAL (MREM)	TOTAL DOSE (PERSON-MREM)	ASSUMPTIONS
REMOVE CLOSURE RING SECTION AND VENT AND DRAIN PORT COVER PLATES	1	8A	1	*60.0*	*1.0*	*1.0*	2 COVERS@2/MIN NO TOOLS
ATTACH RVOAS	2	8A	1	*60.0*	*2.0*	*2.0*	SINGLE THREADED CONNECTION (1 RVOA)
ATTACH A SAMPLE BOTTLE TO VENT PORT RVOA	0.5	8A	1	*60.0*	*0.5*	*0.5*	1" THREADED FITTING NO TOOLS
GATHER A GAS SAMPLE FROM MPC	0.5	8A	1	*60.0*	*0.5*	*0.5*	SMALL BALL VALVE
CLOSE VENT PORT CAP AND DISCONNECT SAMPLE BOTTLE	1	8A	1	*60.0*	*1.0*	*1.0*	1" THREADED FITTING NO TOOLS
ATTACH COOL-DOWN SYSTEM TO RVOAs	2	8A	1	*60.0*	*2.0*	*2.0*	1" THREADED FITTING NO TOOLS X 2
DISCONNECT GAS LINES TO VENT AND DRAIN PORT RVOAs	1	8A	1	*60.0*	*1.0*	*1.0*	1" THREADED FITTING NO TOOLS
VACUUM TOP SURFACES OF MPC AND HI-TRAC	10	6A	1	*31.3*	*5.2*	*5.2*	SHOP VACUUM WITH WAND + HAND WIPE
REMOVE ANNULUS SHIELD	1	8A	1	*60.0*	*1.0*	*1.0*	SHIELD PLACED BY HAND
MANUALLY INSTALL INFLATABLE SEAL	10	6A	2	*31.3*	*5.2*	*10.4*	CONSULTATION WITH CALVERT CLIFFS
OPEN NEUTRON SHIELD JACKET DRAIN VALVE	2	5C	1	*241.8*	*8.1*	*8.1*	SINGLE THREADED CONNECTION
CLOSE NEUTRON SHIELD JACKET DRAIN VALVE	2	5C	1	*241.8*	*8.1*	*8.1*	SINGLE THREADED CONNECTION
REMOVE MPC LID LIFTING HOLE PLUGS	2	5A	1	*62.5*	*2.1*	*2.1*	4 PLUGS AT 2/MIN NO TORQUING
ATTACH LID RETENTION SYSTEM	12	5A	1	*62.5*	*12.5*	*12.5*	24 BOLTS @ 2 MINUTES/BOLT
ATTACH ANNULUS OVERPRESSURE SYSTEM	0.5	5C	1	*241.8*	*2.0*	*2.0*	QUICK DISCONNECT NO TOOLS
POSITION HI-TRAC OVER CASK LOADING AREA	10	5C	1	*241.8*	*40.3*	*40.3*	100 FT @ 10 FT/MIN (CRANE SPEED)
LOWER HI-TRAC INTO SPENT FUEL POOL	8.5	3C	1	*663.4*	*94.0*	*94.0*	17 FEET @ 2 FT/MIN (CRANE SPEED)
REMOVE LID RETENTION BOLTS	12	3B	1	*76.3*	*15.3*	*15.3*	24 BOLTS @ 2/MINUTE
PLACE HI-TRAC ON FLOOR	20	2	2	*3*	*1.0*	*2.0*	40 FEET @ 2 FT/MINUTE (CRANE SPEED)

Table 10.3.2b
HI-STORM 100 SYSTEM UNLOADING OPERATIONS USING THE 100-TON HI-TRAC TRANSFER CASK ESTIMATED OPERATIONAL EXPOSURES† (~~42,500~~*46,000* MWD/MTU, 5*3*-YEAR COOLED PWR FUEL)

ACTION	DURATION (MINUTES)	OPERATOR LOCATION (FIGURE 10.3.1)	NUMBER OF OPERATORS	DOSE RATE AT OPERATOR LOCATION (MREM/HR)	DOSE TO INDIVIDUAL (MREM)	TOTAL DOSE (PERSON-MREM)	ASSUMPTIONS
REMOVE MPC LID	20	2	2	*3*	*1.0*	*2.0*	CONSULTATION WITH CALVERT CLIFFS
Section 8.3.4							
REMOVE SPENT FUEL ASSEMBLIES FROM MPC	1020	1	2	*3*	*51.0*	*102.0*	15 MINUTES PER ASSEMBLY/68 ASSY
TOTAL						~~*1387.72*~~*569.7* PERSON-MREM	

Table 10.3.2c
HI-STORM 100 SYSTEM UNLOADING OPERATIONS USING THE 125-TON HI-TRAC 125D TRANSFER CASK ESTIMATED OPERATIONAL EXPOSURES† (~~57,500~~*75,000* MWD/MTU, ~~125~~-YEAR COOLED PWR FUEL)

ACTION	DURATION (MINUTES)	OPERATOR LOCATION (FIGURE 10.3.1)	NUMBER OF OPERATORS	DOSE RATE AT OPERATOR LOCATION (MREM/HR)	DOSE TO INDIVIDUAL (MREM)	TOTAL DOSE (PERSON-MREM)	ASSUMPTIONS
Section 8.3.2 (Step Sequence Varies By Site and Mode of Transport)							
REMOVE INLET VENT SCREENS	20	16D	1	*43.8*	*14.6*	*14.6*	4 SCREENS@5MIN/SCREEN
INSERT HI-STORM LIFTING JACKS	4	16D	1	*43.8*	*2.9*	*2.9*	4 JACKS@1/MIN
INSERT AIR PAD	5	16D	2	*43.8*	*3.6*	*7.3*	1 PAD MOVED BY HAND
REMOVE HI-STORM LIFTING JACKS	4	16D	1	*43.8*	*2.9*	*2.9*	4 JACKS@1/MIN
TRANSFER HI-STORM TO MPC TRANSFER LOCATION	40	16C	1	*25.5*	*17.0*	*17.0*	200 FEET @ 4FT/MIN
REMOVE HI-STORM LID STUDS/NUTS	10	16A	1	*7.3*	*1.2*	*1.2*	4 BOLTS NO TORQUE
REMOVE HI-STORM LID LIFTING HOLE PLUGS AND INSTALL LID LIFTING SLING	2	16A	1	*7.3*	*0.2*	*0.2*	4 PLUGS AT 2/MIN NO TORQUING
REMOVE GAMMA SHIELD CROSS PLATES	4	16B	1	*73.9*	*4.9*	*4.9*	4 PLATES@1/MIN
REMOVE TEMPERATURE ELEMENTS	8	16B	1	*73.9*	*9.8*	*9.8*	4 TEMP. ELEMENTS @ 2MIN/TEMP. ELEMENT NO TORQUE
REMOVE HI-STORM LID	2	16A	1	*7.3*	*0.2*	*0.2*	4 SHACKLES@2/MIN
INSTALL HI-STORM VENT DUCT SHIELD INSERTS	2	15A	1	*45.5*	*1.5*	*1.5*	4 SHACKLES@2/MIN
INSTALL MATING DEVICE WITH POOL LID	10	15A	1	*45.5*	*7.6*	*7.6*	3 BOLTS AT 2 MINUTES PER BOLT
REMOVE MPC LIFT CLEAT HOLE PLUGS	2	14A	1	*362.5*	*12.1*	*12.1*	4 PLUGS AT 2/MIN NO TORQUING
INSTALL MPC LIFT CLEATS AND MPC SLINGS	2	14A	1	*362.5*	*12.1*	*12.1*	4 PLUGS AT 2/MIN NO TORQUING

† See notes at bottom of Table 10.3.4.

Table 10.3.2c
HI-STORM 100 SYSTEM UNLOADING OPERATIONS USING THE 125-TON HI-TRAC 125D TRANSFER CASK ESTIMATED OPERATIONAL EXPOSURES† (~~57,500~~*75,000* MWD/MTU, ~~125~~-YEAR COOLED PWR FUEL)

ACTION	DURATION (MINUTES)	OPERATOR LOCATION (FIGURE 10.3.1)	NUMBER OF OPERATORS	DOSE RATE AT OPERATOR LOCATION (MREM/HR)	DOSE TO INDIVIDUAL (MREM)	TOTAL DOSE (PERSON-MREM)	ASSUMPTIONS
ALIGN HI-TRAC OVER HI-STORM AND MATE OVERPACKS	10	13B	2	*118.5*	*19.8*	*39.5*	ALIGNMENT GUIDES USED
PULL MPC SLINGS THROUGH TOP LID HOLE	10	13A	2	*158.5*	*26.4*	*52.8*	2 SLINGS@5MIN/SLING
INSTALL TRIM PLATES	4	13B	2	*118.5*	*7.9*	*15.8*	INSTALLED BY HAND NO FASTENERS
ATTACH MPC SLING TO LIFTING DEVICE	10	13A	1	*158.5*	*26.4*	*26.4*	2 SLINGS@5MIN/SLING NO BOLTING
CLOSE MATING DEVICE DRAWER AND BOLT-UP POOL LID	36	13B	2	*118.5*	*71.1*	*142.2*	2 PINS@2MIN/PIN, 16 BOLTS @ 2MIN/BOLT
DISCONNECT SLINGS FROM MPC LIFT CLEATS	10	13A	2	*158.5*	*26.4*	*52.8*	2 SLINGS@5MIN/SLING
DOWNEND HI-TRAC ON TRANSPORT FRAME	20	12A	2	*26.4*	*8.8*	*17.6*	ALIGN TRUNNIONS, DISCONNECT LIFT YOKE
TRANSPORT HI-TRAC TO FUEL BUILDING	90	12A	1	*26.4*	*39.6*	*39.6*	DRIVER RECEIVES MOST DOSE
UPEND HI-TRAC	20	12A	2	*26.4*	*8.8*	*17.6*	ALIGN TRUNNIONS, DISCONNECT LIFT YOKE
Section 8.3.3							
PLACE HI-TRAC IN PREPARATION AREA	15	9C	1	*117.8*	*29.5*	*29.5*	100 FT @ 10 FT/MIN (CRANE SPEED)+ 5MIN TO ALIGN
REMOVE TOP LID BOLTS	6	9B	1	*37.9*	*3.8*	*3.8*	24 BOLTS AT 4/MIN (NO TORQUE IMPACT TOOLS)
REMOVE HI-TRAC TOP LID	2	6A	1	*18.7*	*0.6*	*0.6*	4 SHACKLES@2/MIN
ATTACH WATER FILL LINE TO HI-TRAC DRAIN PORT	0.5	9D	1	*354.2*	*3.0*	*3.0*	QUICK DISCONNECT NO TOOLS
INSTALL BOLT PLUGS OR WATERPROOF TAPE FROM HI-TRAC TOP BOLT HOLES	9	8A	1	*37.9*	*5.7*	*5.7*	18 HOLES@2/MIN
CORE DRILL CLOSURE RING AND VENT AND DRAIN PORT COVER PLATES	40	7A	2	*18.7*	*12.5*	*24.9*	20 MINUTES TO INSTALL/ALIGN +10 MIN/COVER

Table 10.3.2c
HI-STORM 100 SYSTEM UNLOADING OPERATIONS USING THE 125-TON HI-TRAC 125D TRANSFER CASK ESTIMATED OPERATIONAL EXPOSURES† (~~57,500~~*75,000* MWD/MTU, ~~125~~-YEAR COOLED PWR FUEL)

ACTION	DURATION (MINUTES)	OPERATOR LOCATION (FIGURE 10.3.1)	NUMBER OF OPERATORS	DOSE RATE AT OPERATOR LOCATION (MREM/HR)	DOSE TO INDIVIDUAL (MREM)	TOTAL DOSE (PERSON-MREM)	ASSUMPTIONS
REMOVE CLOSURE RING SECTION AND VENT AND DRAIN PORT COVER PLATES	1	8A	1	*37.9*	*0.6*	*0.6*	2 COVERS@2/MIN NO TOOLS
ATTACH RVOAS	2	8A	1	*37.9*	*1.3*	*1.3*	SINGLE THREADED CONNECTION (1 RVOA)
ATTACH A SAMPLE BOTTLE TO VENT PORT RVOA	0.5	8A	1	*37.9*	*0.3*	*0.3*	1" THREADED FITTING NO TOOLS
GATHER A GAS SAMPLE FROM MPC	0.5	8A	1	*37.9*	*0.3*	*0.3*	SMALL BALL VALVE
CLOSE VENT PORT CAP AND DISCONNECT SAMPLE BOTTLE	1	8A	1	*37.9*	*0.6*	*0.6*	1" THREADED FITTING NO TOOLS
ATTACH COOL-DOWN SYSTEM TO RVOAs	2	8A	1	*37.9*	*1.3*	*1.3*	1" THREADED FITTING NO TOOLS X 2
DISCONNECT GAS LINES TO VENT AND DRAIN PORT RVOAs	1	8A	1	*37.9*	*0.6*	*0.6*	1" THREADED FITTING NO TOOLS
VACUUM TOP SURFACES OF MPC AND HI-TRAC	10	6A	1	*18.7*	*3.1*	*3.1*	SHOP VACUUM WITH WAND + HAND WIPE
REMOVE ANNULUS SHIELD	1	8A	1	*37.9*	*0.6*	*0.6*	SHIELD PLACED BY HAND
MANUALLY INSTALL INFLATABLE SEAL	10	6A	2	*18.7*	*3.1*	*6.2*	CONSULTATION WITH CALVERT CLIFFS
OPEN NEUTRON SHIELD JACKET DRAIN VALVE	2	5C	1	*82.7*	*2.8*	*2.8*	SINGLE THREADED CONNECTION
CLOSE NEUTRON SHIELD JACKET DRAIN VALVE	2	5C	1	*82.7*	*2.8*	*2.8*	SINGLE THREADED CONNECTION
REMOVE MPC LID LIFTING HOLE PLUGS	2	5A	1	*37.3*	*1.2*	*1.2*	4 PLUGS AT 2/MIN NO TORQUING
ATTACH LID RETENTION SYSTEM	12	5A	1	*37.3*	*7.5*	*7.5*	24 BOLTS @ 2 MINUTES/BOLT
ATTACH ANNULUS OVERPRESSURE SYSTEM	0.5	5C	1	*82.7*	*0.7*	*0.7*	QUICK DISCONNECT NO TOOLS
POSITION HI-TRAC OVER CASK LOADING AREA	10	5C	1	*82.7*	*13.8*	*13.8*	100 FT @ 10 FT/MIN (CRANE SPEED)

Table 10.3.2c
HI-STORM 100 SYSTEM UNLOADING OPERATIONS USING THE 125-TON HI-TRAC 125D TRANSFER CASK ESTIMATED OPERATIONAL EXPOSURES† (~~57,500~~*75,000* MWD/MTU, ~~125~~-YEAR COOLED PWR FUEL)

ACTION	DURATION (MINUTES)	OPERATOR LOCATION (FIGURE 10.3.1)	NUMBER OF OPERATORS	DOSE RATE AT OPERATOR LOCATION (MREM/HR)	DOSE TO INDIVIDUAL (MREM)	TOTAL DOSE (PERSON-MREM)	ASSUMPTIONS
LOWER HI-TRAC INTO SPENT FUEL POOL	8.5	3C	1	*117.8*	*16.7*	*16.7*	17 FEET @ 2 FT/MIN (CRANE SPEED)
REMOVE LID RETENTION BOLTS	12	3B	1	*46.4*	*9.3*	*9.3*	24 BOLTS @ 2/MINUTE
PLACE HI-TRAC ON FLOOR	20	2	2	*2.0*	*0.7*	*1.3*	40 FEET @ 2 FT/MINUTE (CRANE SPEED)
REMOVE MPC LID	20	2	2	*2.0*	*0.7*	*1.3*	CONSULTATION WITH CALVERT CLIFFS
Section 8.3.4							
REMOVE SPENT FUEL ASSEMBLIES FROM MPC	1020	1	2	*1.0*	*17.0*	*34.0*	15 MINUTES PER ASSEMBLY/68 ASSY
TOTAL							~~329.1~~*672.6* PERSON-MREM

Table 10.3.3a
MPC TRANSFER INTO THE HI-STORM 100 SYSTEM DIRECTLY FROM TRANSPORT USING THE 125-TON HI-TRAC TRANSFER CASK
ESTIMATED OPERATIONAL EXPOSURES† (~~57,500~~*75,000* MWD/MTU, ~~125~~-YEAR COOLED PWR FUEL)

ACTION	DURATION (MINUTES)	OPERATOR LOCATION (FIGURE 10.3.1)	NUMBER OF OPERATORS	DOSE RATE AT OPERATOR LOCATION (MREM/HR)	DOSE TO INDIVIDUAL (MREM)	TOTAL DOSE (PERSON-MREM)	ASSUMPTIONS
Section 8.5.2							
MEASURE HI-STAR DOSE RATES	16	17A	2	*14.1*	*3.8*	*7.5*	16 POINTS@1 POINT/MIN
REMOVE PERSONNEL BARRIER	10	17C	2	*21.5*	*3.6*	*7.2*	ATTACH SLING REMOVE 8 LOCKS
PERFORM REMOVABLE CONTAMINATION SURVEYS	1	17C	1	*21.5*	*0.4*	*0.4*	10 SMEARS@10 SMEARS/MINUTE
REMOVE IMPACT LIMITERS	16	17A	2	*14.1*	*3.8*	*7.5*	ATTACH FRAME REMOVE 22 BOLTS IMPACT TOOLS
REMOVE TIE-DOWN	6	17A	2	*14.1*	*1.4*	*2.8*	ATTACH 2-LEGGED SLING REMOVE 4 BOLTS
PERFORM A VISUAL INSPECTION OF OVERPACK	10	17B	1	*9.0*	*1.5*	*1.5*	CHECKSHEET USED
REMOVE REMOVABLE SHEAR RING SEGMENTS	4	17A	1	*14.1*	*0.9*	*0.9*	4 BOLTS EACH @2/MIN X 2 SEGMENTS
UPEND HI-STAR OVERPACK	20	17B	2	*9.0*	*3.0*	*6.0*	DISCONNECT LIFT YOKE
INSTALL TEMPORARY SHIELD RING SEGMENTS	16	18A	1	*7.1*	*1.9*	*1.9*	8 SEGMENTS @ 2 MIN/SEGMENT
FILL TEMPORARY SHIELD RING SEGMENTS	25	18A	1	*7.1*	*3.0*	*3.0*	230 GAL @10GPM, LONG HANDLED SPRAYER
REMOVE OVERPACK VENT PORT COVER PLATE	2	18A	1	*7.1*	*0.2*	*0.2*	4 BOLTS @2/MIN
ATTACH BACKFILL TOOL	2	18A	1	*7.1*	*0.2*	*0.2*	4 BOLTS @2/MIN
OPEN/CLOSE VENT PORT PLUG	0.5	18A	1	*7.1*	*0.1*	*0.1*	SINGLE TURN BY HAND NO TOOLS
REMOVE CLOSURE PLATE BOLTS	39	18A	2	*7.1*	*4.6*	*9.2*	52 BOLTS@4/MIN X 3 PASSES
REMOVE OVERPACK CLOSURE PLATE	2	18A	1	*7.1*	*0.2*	*0.2*	4 SHACKLES@2/MIN
INSTALL HI-STAR SEAL SURFACE PROTECTOR	2	19B	1	*7.1*	*0.2*	*0.2*	PLACED BY HAND NO TOOLS
INSTALL TRANSFER COLLAR ON HI-STAR	10	19B	2	*7.1*	*1.2*	*2.4*	ALIGN AND POSITION REMOVE 4 SHACKLES

† See notes at bottom of Table 10.3.4.

Table 10.3.3a
MPC TRANSFER INTO THE HI-STORM 100 SYSTEM DIRECTLY FROM TRANSPORT USING THE 125-TON HI-TRAC TRANSFER CASK
ESTIMATED OPERATIONAL EXPOSURES† (~~57,500~~*75,000* MWD/MTU, ~~125~~-YEAR COOLED PWR FUEL)

ACTION	DURATION (MINUTES)	OPERATOR LOCATION (FIGURE 10.3.1)	NUMBER OF OPERATORS	DOSE RATE AT OPERATOR LOCATION (MREM/HR)	DOSE TO INDIVIDUAL (MREM)	TOTAL DOSE (PERSON-MREM)	ASSUMPTIONS
REMOVE MPC LIFT CLEAT HOLE PLUGS	2	19A	1	*362.5*	*12.1*	*12.1*	4 PLUGS AT 2/MIN NO TORQUING
INSTALL MPC LIFT CLEATS AND LIFT SLING	25	19A	2	*362.5*	*151.0*	*302.1*	INSTALL CLEATS AND HYDRO TORQUE 4 BOLTS
MATE OVERPACKS	10	20B	2	*118.5*	*19.8*	*39.5*	ALIGNMENT GUIDES USED
REMOVE DOOR LOCKING PINS AND OPEN DOORS	4	20B	2	*118.5*	*7.9*	*15.8*	2 PINS@2/MIN
INSTALL TRIM PLATES	4	20B	2	*118.5*	*7.9*	*15.8*	INSTALLED BY HAND NO FASTENERS
Section 8.5.3							
REMOVE TRIM PLATES	4	20B	2	*118.5*	*7.9*	*15.8*	INSTALLED BY HAND NO FASTENERS
DISCONNECT SLINGS FROM MPC LIFTING DEVICE	10	20A	2	*158.5*	*26.4*	*52.8*	2 SLINGS@5/MIN
INSTALL TRIM PLATES	4	13B	2	*118.5*	*7.9*	*15.8*	INSTALLED BY HAND NO FASTENERS
REMOVE MPC LIFT CLEATS AND MPC LIFT SLINGS	10	14A	1	*362.5*	*60.4*	*60.4*	4 BOLTS,NO TORQUING
INSTALL HOLE PLUGS IN EMPTY MPC BOLT HOLES	2	14A	1	*362.5*	*12.1*	*12.1*	4 PLUGS AT 2/MIN NO TORQUING
REMOVE HI-STORM VENT DUCT SHIELD INSERTS	2	15A	1	*45.5*	*1.5*	*1.5*	4 SHACKLES@2/MIN
REMOVE ALIGNMENT DEVICE	4	15A	1	*45.5*	*3.0*	*3.0*	REMOVED BY HAND NO TOOLS (4 PCS@1/MIN)
INSTALL HI-STORM LID AND INSTALL LID STUDS/NUTS	25	16A	2	*7.3*	*3.1*	*6.1*	INSTALL LID AND HYDRO TORQUE 4 BOLTS
INSTALL HI-STORM EXIT VENT GAMMA SHIELD CROSS PLATES	4	16B	1	*73.9*	*4.9*	*4.9*	4 PCS @ 1/MIN INSTALL BY HAND NO TOOLS
INSTALL TEMPERATURE ELEMENTS	20	16B	1	*73.9*	*24.6*	*24.6*	4@5MIN/TEMPERATURE ELEMENT
INSTALL EXIT VENT SCREENS	20	16B	1	*73.9*	*24.6*	*24.6*	4 SCREENS@5MIN/SCREEN
REMOVE HI-STORM LID LIFTING DEVICE	2	16A	1	*7.3*	*0.2*	*0.2*	4 SHACKLES@2/MIN

Table 10.3.3a
MPC TRANSFER INTO THE HI-STORM 100 SYSTEM DIRECTLY FROM TRANSPORT USING THE 125-TON HI-TRAC TRANSFER CASK
ESTIMATED OPERATIONAL EXPOSURES† (~~57,500~~*75,000* MWD/MTU, ~~12~~*5*-YEAR COOLED PWR FUEL)

ACTION	DURATION (MINUTES)	OPERATOR LOCATION (FIGURE 10.3.1)	NUMBER OF OPERATORS	DOSE RATE AT OPERATOR LOCATION (MREM/HR)	DOSE TO INDIVIDUAL (MREM)	TOTAL DOSE (PERSON-MREM)	ASSUMPTIONS
INSTALL HOLE PLUGS IN EMPTY HOLES	2	16A	1	*7.3*	*0.2*	*0.2*	4 PLUGS AT 2/MIN NO TORQUING
PERFORM SHIELDING EFFECTIVENESS TESTING	16	16D	1	*43.8*	*11.7*	*11.7*	16POINTS@1 MIN
SECURE HI-STORM TO TRANSPORT DEVICE	10	16A	1	*7.3*	*1.2*	*1.2*	ASSUMES AIR PAD
TRANSFER HI-STORM TO ITS DESIGNATED STORAGE LOCATION	40	16C	1	*25.5*	*17.0*	*17.0*	200 FEET @ 4FT/MIN
INSERT HI-STORM LIFTING JACKS	4	16D	1	*43.8*	*2.9*	*2.9*	4 JACKS@1/MIN
REMOVE AIR PAD	5	16D	1	*43.8*	*3.6*	*3.6*	1 PAD MOVED BY HAND
REMOVE HI-STORM LIFTING JACKS	4	16D	1	*43.8*	*2.9*	*2.9*	4 JACKS@1/MIN
INSTALL INLET VENT SCREENS	20	16D	1	*43.8*	*14.6*	*14.6*	4 SCREENS@5MIN/SCREEN
PERFORM AIR TEMPERATURE RISE TEST	8	16B	1	*73.9*	*9.8*	*9.8*	8 MEASMT@1/MIN
TOTAL						~~465.~~*4722.6* PERSON-MREM	

Table 10.3.3b
MPC TRANSFER INTO THE HI-STORM 100 SYSTEM DIRECTLY FROM TRANSPORT USING THE 100-TON HI-TRAC TRANSFER CASK
ESTIMATED OPERATIONAL EXPOSURES† (~~42,500~~46,000 MWD/MTU, 53-YEAR COOLED PWR FUEL)

ACTION	DURATION (MINUTES)	OPERATOR LOCATION (FIGURE 10.3.1)	NUMBER OF OPERATORS	DOSE RATE AT OPERATOR LOCATION (MREM/HR)	DOSE TO INDIVIDUAL (MREM)	TOTAL DOSE (PERSON-MREM)	ASSUMPTIONS
Section 8.5.2							
MEASURE HI-STAR DOSE RATES	16	17A	2	14.1	3.8	7.5	16 POINTS@1 POINT/MIN
REMOVE PERSONNEL BARRIER	10	17C	2	21.5	3.6	7.2	ATTACH SLING REMOVE 8 LOCKS
PERFORM REMOVABLE CONTAMINATION SURVEYS	1	17C	1	21.5	0.4	0.4	10 SMEARS@10 SMEARS/MINUTE
REMOVE IMPACT LIMITERS	16	17A	2	14.1	3.8	7.5	ATTACH FRAME REMOVE 22 BOLTS IMPACT TOOLS
REMOVE TIE-DOWN	6	17A	2	14.1	1.4	2.8	ATTACH 2-LEGGED SLING REMOVE 4 BOLTS
PERFORM A VISUAL INSPECTION OF OVERPACK	10	17B	1	9.0	1.5	1.5	CHECKSHEET USED
REMOVE REMOVABLE SHEAR RING SEGMENTS	4	17A	1	14.1	0.9	0.9	4 BOLTS EACH @2/MIN X 2 SEGMENTS
UPEND HI-STAR OVERPACK	20	17B	2	9.0	3.0	6.0	DISCONNECT LIFT YOKE
INSTALL TEMPORARY SHIELD RING SEGMENTS	16	18A	1	7.1	1.9	1.9	8 SEGMENTS @ 2 MIN/SEGMENT
FILL TEMPORARY SHIELD RING SEGMENTS	25	18A	1	7.1	3.0	3.0	230 GAL @10GPM, LONG HANDLED SPRAYER
REMOVE OVERPACK VENT PORT COVER PLATE	2	18A	1	7.1	0.2	0.2	4 BOLTS @2/MIN
ATTACH BACKFILL TOOL	2	18A	1	7.1	0.2	0.2	4 BOLTS @2/MIN
OPEN/CLOSE VENT PORT PLUG	0.5	18A	1	7.1	0.1	0.1	SINGLE TURN BY HAND NO TOOLS
REMOVE CLOSURE PLATE BOLTS	39	18A	2	7.1	4.6	9.2	52 BOLTS@4/MIN X 3 PASSES
REMOVE OVERPACK CLOSURE PLATE	2	18A	1	7.1	0.2	0.2	4 SHACKLES@2/MIN
INSTALL HI-STAR SEAL SURFACE PROTECTOR	2	19B	1	7.1	0.2	0.2	PLACED BY HAND NO TOOLS
INSTALL TRANSFER COLLAR ON HI-STAR	10	19B	2	7.1	1.2	2.4	ALIGN AND POSITION REMOVE 4 SHACKLES

† See notes at bottom of Table 10.3.4.

Table 10.3.3b
MPC TRANSFER INTO THE HI-STORM 100 SYSTEM DIRECTLY FROM TRANSPORT USING THE 100-TON HI-TRAC TRANSFER CASK
ESTIMATED OPERATIONAL EXPOSURES† (~~42,500~~*46,000* MWD/MTU, ~~5~~*3*-YEAR COOLED PWR FUEL)

ACTION	DURATION (MINUTES)	OPERATOR LOCATION (FIGURE 10.3.1)	NUMBER OF OPERATORS	DOSE RATE AT OPERATOR LOCATION (MREM/HR)	DOSE TO INDIVIDUAL (MREM)	TOTAL DOSE (PERSON-MREM)	ASSUMPTIONS
REMOVE MPC LIFT CLEAT HOLE PLUGS	2	19A	1	*362.5*	*12.1*	*12.1*	4 PLUGS AT 2/MIN NO TORQUING
INSTALL MPC LIFT CLEATS AND LIFT SLING	25	19A	2	*362.5*	*151.0*	*302.1*	INSTALL CLEATS AND HYDRO TORQUE 4 BOLTS
MATE OVERPACKS	10	20B	2	*561.8*	*93.6*	*187.3*	ALIGNMENT GUIDES USED
REMOVE DOOR LOCKING PINS AND OPEN DOORS	4	20B	2	*561.8*	*37.5*	*74.9*	2 PINS@2/MIN
INSTALL TRIM PLATES	4	20B	2	*561.8*	*37.5*	*74.9*	INSTALLED BY HAND NO FASTENERS
Section 8.5.3							
REMOVE TRIM PLATES	4	20B	2	*561.8*	*37.5*	*74.9*	INSTALLED BY HAND NO FASTENERS
DISCONNECT SLINGS FROM MPC LIFTING DEVICE	10	20A	2	*247.7*	*41.3*	*82.6*	2 SLINGS@5/MIN
REMOVE TRIM PLATES	4	13B	2	*561.8*	*37.5*	*74.9*	INSTALLED BY HAND NO FASTENERS
REMOVE MPC LIFT CLEATS AND MPC LIFT SLINGS	10	14A	1	*362.5*	*60.4*	*60.4*	4 BOLTS,NO TORQUING
INSTALL HOLE PLUGS IN EMPTY MPC BOLT HOLES	2	14A	1	*362.5*	*12.1*	*12.1*	4 PLUGS AT 2/MIN NO TORQUING
REMOVE HI-STORM VENT DUCT SHIELD INSERTS	2	15A	1	*45.5*	*1.5*	*1.5*	4 SHACKLES@2/MIN
REMOVE ALIGNMENT DEVICE	4	15A	1	*45.5*	*3.0*	*3.0*	REMOVED BY HAND NO TOOLS (4 PCS@1/MIN)
INSTALL HI-STORM LID AND INSTALL LID STUDS/NUTS	25	16A	2	*7.3*	*3.1*	*6.1*	INSTALL LID AND HYDRO TORQUE 4 BOLTS
INSTALL HI-STORM EXIT VENT GAMMA SHIELD CROSS PLATES	4	16B	1	*73.9*	*4.9*	*4.9*	4 PCS @ 1/MIN INSTALL BY HAND NO TOOLS
INSTALL TEMPERATURE ELEMENTS	20	16B	1	*73.9*	*24.6*	*24.6*	4@5MIN/TEMPERATURE ELEMENT
INSTALL EXIT VENT SCREENS	20	16B	1	*73.9*	*24.6*	*24.6*	4 SCREENS@5MIN/SCREEN
REMOVE HI-STORM LID LIFTING DEVICE	2	16A	1	*7.3*	*0.2*	*0.2*	4 SHACKLES@2/MIN

Table 10.3.3b
MPC TRANSFER INTO THE HI-STORM 100 SYSTEM DIRECTLY FROM TRANSPORT USING THE 100-TON HI-TRAC TRANSFER CASK
ESTIMATED OPERATIONAL EXPOSURES† (~~42,500~~*46,000* MWD/MTU, ~~5~~*3*-YEAR COOLED PWR FUEL)

ACTION	DURATION (MINUTES)	OPERATOR LOCATION (FIGURE 10.3.1)	NUMBER OF OPERATORS	DOSE RATE AT OPERATOR LOCATION (MREM/HR)	DOSE TO INDIVIDUAL (MREM)	TOTAL DOSE (PERSON-MREM)	ASSUMPTIONS
INSTALL HOLE PLUGS IN EMPTY HOLES	2	16A	1	*7.3*	*0.2*	*0.2*	4 PLUGS AT 2/MIN NO TORQUING
PERFORM SHIELDING EFFECTIVENESS TESTING	16	16D	1	*43.8*	*11.7*	*11.7*	16POINTS@1 MIN
SECURE HI-STORM TO TRANSPORT DEVICE	10	16A	1	*7.3*	*1.2*	*1.2*	ASSUMES AIR PAD
TRANSFER HI-STORM TO ITS DESIGNATED STORAGE LOCATION	40	16C	1	*25.5*	*17.0*	*17.0*	200 FEET @ 4FT/MIN
INSERT HI-STORM LIFTING JACKS	4	16D	1	*43.8*	*2.9*	*2.9*	4 JACKS@1/MIN
REMOVE AIR PAD	5	16D	1	*43.8*	*3.6*	*3.6*	1 PAD MOVED BY HAND
REMOVE HI-STORM LIFTING JACKS	4	16D	1	*43.8*	*2.9*	*2.9*	4 JACKS@1/MIN
INSTALL INLET VENT SCREENS	20	16D	1	*43.8*	*14.6*	*14.6*	4 SCREENS@5MIN/SCREEN
PERFORM AIR TEMPERATURE RISE TEST	8	16B	1	*73.9*	*9.8*	*9.8*	8 MEASMT@1/MIN
TOTAL						~~718.8~~*1136.5* PERSON-MREM	

Table 10.3.3c
MPC TRANSFER INTO THE HI-STORM 100 SYSTEM DIRECTLY FROM TRANSPORT USING THE 125-TON HI-TRAC 125D TRANSFER CASK
ESTIMATED OPERATIONAL EXPOSURES† (~~57,500~~*75,000* MWD/MTU, ~~125~~-YEAR COOLED PWR FUEL)

ACTION	DURATION (MINUTES)	OPERATOR LOCATION (FIGURE 10.3.1)	NUMBER OF OPERATORS	DOSE RATE AT OPERATOR LOCATION (MREM/HR)	DOSE TO INDIVIDUAL (MREM)	TOTAL DOSE (PERSON-MREM)	ASSUMPTIONS
Section 8.5.2							
MEASURE HI-STAR DOSE RATES	16	17A	2	*14.1*	*3.8*	*7.5*	16 POINTS@1 POINT/MIN
REMOVE PERSONNEL BARRIER	10	17C	2	*21.5*	*3.6*	*7.2*	ATTACH SLING REMOVE 8 LOCKS
PERFORM REMOVABLE CONTAMINATION SURVEYS	1	17C	1	*21.5*	*0.4*	*0.4*	10 SMEARS@10 SMEARS/MINUTE
REMOVE IMPACT LIMITERS	16	17A	2	*14.1*	*3.8*	*7.5*	ATTACH FRAME REMOVE 22 BOLTS IMPACT TOOLS
REMOVE TIE-DOWN	6	17A	2	*14.1*	*1.4*	*2.8*	ATTACH 2-LEGGED SLING REMOVE 4 BOLTS
PERFORM A VISUAL INSPECTION OF OVERPACK	10	17B	1	*9.0*	*1.5*	*1.5*	CHECKSHEET USED
REMOVE REMOVABLE SHEAR RING SEGMENTS	4	17A	1	*14.1*	*0.9*	*0.9*	4 BOLTS EACH @2/MIN X 2 SEGMENTS
UPEND HI-STAR OVERPACK	20	17B	2	*9.0*	*3.0*	*6.0*	DISCONNECT LIFT YOKE
INSTALL TEMPORARY SHIELD RING SEGMENTS	16	18A	1	*7.1*	*1.9*	*1.9*	8 SEGMENTS @ 2 MIN/SEGMENT
FILL TEMPORARY SHIELD RING SEGMENTS	25	18A	1	*7.1*	*3.0*	*3.0*	230 GAL @10GPM, LONG HANDLED SPRAYER
REMOVE OVERPACK VENT PORT COVER PLATE	2	18A	1	*7.1*	*0.2*	*0.2*	4 BOLTS @2/MIN
ATTACH BACKFILL TOOL	2	18A	1	*7.1*	*0.2*	*0.2*	4 BOLTS @2/MIN
OPEN/CLOSE VENT PORT PLUG	0.5	18A	1	*7.1*	*0.1*	*0.1*	SINGLE TURN BY HAND NO TOOLS
REMOVE CLOSURE PLATE BOLTS	39	18A	2	*7.1*	*4.6*	*9.2*	52 BOLTS@4/MIN X 3 PASSES

† See notes at bottom of Table 10.3.4.

Table 10.3.3c
MPC TRANSFER INTO THE HI-STORM 100 SYSTEM DIRECTLY FROM TRANSPORT USING THE 125-TON HI-TRAC 125D TRANSFER CASK
ESTIMATED OPERATIONAL EXPOSURES† (~~57,500~~*75,000* MWD/MTU, ~~*125*~~-YEAR COOLED PWR FUEL)

ACTION	DURATION (MINUTES)	OPERATOR LOCATION (FIGURE 10.3.1)	NUMBER OF OPERATORS	DOSE RATE AT OPERATOR LOCATION (MREM/HR)	DOSE TO INDIVIDUAL (MREM)	TOTAL DOSE (PERSON-MREM)	ASSUMPTIONS
REMOVE OVERPACK CLOSURE PLATE	2	18A	1	*7.1*	*0.2*	*0.2*	4 SHACKLES@2/MIN
INSTALL HI-STAR SEAL SURFACE PROTECTOR	2	19B	1	*7.1*	*0.2*	*0.2*	PLACED BY HAND NO TOOLS
INSTALL MATING DEVICE ON HI-STAR	20	19B	2	*7.1*	*2.4*	*4.7*	ALIGN AND BOLT INTO PLACE
REMOVE MPC LIFT CLEAT HOLE PLUGS	2	19A	1	*362.5*	*12.1*	*12.1*	4 PLUGS AT 2/MIN NO TORQUING
INSTALL MPC LIFT CLEATS AND LIFT SLING	25	19A	2	*362.5*	*151.0*	*302.1*	INSTALL CLEATS AND HYDRO TORQUE 4 BOLTS
MATE OVERPACKS	10	20B	2	*118.5*	*19.8*	*39.5*	ALIGNMENT GUIDES USED
REMOVE LOCKING PINS AND OPEN DRAWER	4	20B	2	*118.5*	*7.9*	*15.8*	2 PINS@2/MIN
INSTALL TRIM PLATES	4	20B	2	*118.5*	*7.9*	*15.8*	INSTALLED BY HAND NO FASTENERS
Section 8.5.3							
REMOVE TRIM PLATES	4	20B	2	*118.5*	*7.9*	*15.8*	INSTALLED BY HAND NO FASTENERS
RAISE THE POOL LID AND BOLT INTO PLACE ON HI-TRAC	32	20B	2	*118.5*	*63.2*	*126.4*	2 MINS/BOLT, 16 BOLTS
DISCONNECT SLINGS FROM MPC LIFTING DEVICE	10	20A	2	*158.5*	*26.4*	*52.8*	2 SLINGS@5/MIN
INSTALL TRIM PLATES	4	13B	2	*118.5*	*7.9*	*15.8*	INSTALLED BY HAND NO FASTENERS
REMOVE MPC LIFT CLEATS AND MPC LIFT SLINGS	10	14A	1	*362.5*	*60.4*	*60.4*	4 BOLTS,NO TORQUING
INSTALL HOLE PLUGS IN EMPTY MPC BOLT HOLES	2	14A	1	*362.5*	*12.1*	*12.1*	4 PLUGS AT 2/MIN NO TORQUING
REMOVE HI-STORM VENT DUCT SHIELD INSERTS	2	15A	1	*45.5*	*1.5*	*1.5*	4 SHACKLES@2/MIN
REMOVE THE MATINGE DEVICE	6	15A	1	*45.5*	*4.5*	*4.5*	3 BOLTS AT 2 MINUTES PER BOLTS

Table 10.3.3c
MPC TRANSFER INTO THE HI-STORM 100 SYSTEM DIRECTLY FROM TRANSPORT USING THE 125-TON HI-TRAC 125D TRANSFER CASK
ESTIMATED OPERATIONAL EXPOSURES† (~~57,500~~*75,000* MWD/MTU, ~~125~~-YEAR COOLED PWR FUEL)

ACTION	DURATION (MINUTES)	OPERATOR LOCATION (FIGURE 10.3.1)	NUMBER OF OPERATORS	DOSE RATE AT OPERATOR LOCATION (MREM/HR)	DOSE TO INDIVIDUAL (MREM)	TOTAL DOSE (PERSON-MREM)	ASSUMPTIONS
INSTALL HI-STORM LID AND INSTALL LID STUDS/NUTS	25	16A	2	*7.3*	*3.1*	*6.1*	INSTALL LID AND HYDRO TORQUE 4 BOLTS
INSTALL HI-STORM EXIT VENT GAMMA SHIELD CROSS PLATES	4	16B	1	*73.9*	*4.9*	*4.9*	4 PCS @ 1/MIN INSTALL BY HAND NO TOOLS
INSTALL TEMPERATURE ELEMENTS	20	16B	1	*73.9*	*24.6*	*24.6*	4@5MIN/TEMPERATURE ELEMENT
INSTALL EXIT VENT SCREENS	20	16B	1	*73.9*	*24.6*	*24.6*	4 SCREENS@5MIN/SCREEN
REMOVE HI-STORM LID LIFTING DEVICE	2	16A	1	*7.3*	*0.2*	*0.2*	4 SHACKLES@2/MIN
INSTALL HOLE PLUGS IN EMPTY HOLES	2	16A	1	*7.3*	*0.2*	*0.2*	4 PLUGS AT 2/MIN NO TORQUING
PERFORM SHIELDING EFFECTIVENESS TESTING	16	16D	1	*43.8*	*11.7*	*11.7*	16POINTS@1 MIN
SECURE HI-STORM TO TRANSPORT DEVICE	10	16A	1	*7.3*	*1.2*	*1.2*	ASSUMES AIR PAD
TRANSFER HI-STORM TO ITS DESIGNATED STORAGE LOCATION	40	16C	1	*25.5*	*17.0*	*17.0*	200 FEET @ 4FT/MIN
INSERT HI-STORM LIFTING JACKS	4	16D	1	*43.8*	*2.9*	*2.9*	4 JACKS @1/MIN
REMOVE AIR PAD	5	16D	1	*43.8*	*3.6*	*3.6*	1 PAD MOVED BY HAND
REMOVE HI-STORM LIFTING JACKS	4	16D	1	*43.8*	*2.9*	*2.9*	4 JACKS @1/MIN
INSTALL INLET VENT SCREENS	20	16D	1	*43.8*	*14.6*	*14.6*	4 SCREENS@5MIN/SCREEN
PERFORM AIR TEMPERATURE RISE TEST	8	16B	1	*73.9*	*9.8*	*9.8*	8 MEASMT@1/MIN
TOTAL							~~514.0~~*852.9* PERSON–MREM

Table 10.3.4
ESTIMATED EXPOSURES FOR HI-STORM 100 SURVEILLANCE AND MAINTENANCE

ACTIVITY	ESTIMATED PERSONNEL	ESTIMATED HOURS PER YEAR	ESTIMATED DOSE RATE (MREM/HR)	OCCUPATIONAL DOSE TO INDIVIDUAL (PERSON-MREM)
SECURITY SURVEILLANCE	1	30	3	90
ANNUAL MAINTENANCE	2	15	10	300

Notes for Tables 10.3.1a, 10.3.1b, 10.3.1c, 10.3.2a, 10.3.2b, 10.3.2c, 10.3.3a, 10.3.3b, 10.3.3c and 10..3.4:

1. Refer to Chapter 8 for detailed description of activities.
2 Number of operators may be set to 1 to simplify calculations where the duration is indirectly proportional to the number of operators. The total dose is equivalent in both respects.
3 HI-STAR 100 Operations assume that the cooling time is at least 10 years.

10.4 ESTIMATED COLLECTIVE DOSE ASSESSMENT

10.4.1 Controlled Area Boundary Dose for Normal Operations

10CFR72.104 [10.0.1] limits the annual dose equivalent to any real individual at the controlled area boundary to a maximum of 25 mrem to the whole body, 75 mrem to the thyroid, and 25 mrem for any other critical organ. This includes contributions from all uranium fuel cycle operations in the region.

It is not feasible to predict bounding controlled area boundary dose rates on a generic basis since radiation from plant and other sources; the location and the layout of an ISFSI; and the number and configuration of casks are necessarily site-specific. In order to compare the performance of the HI-STORM 100 System with the regulatory requirements, sample ISFSI arrays were analyzed in Chapter 5. These represent a full array of design basis fuel assemblies. Users are required to perform a site specific dose analysis for their particular situation in accordance with 10CFR72.212 [10.0.1]. The analysis must account for the ISFSI (size, configuration, fuel assembly specifics) and any other radiation from uranium fuel cycle operations within the region.

Table 5.1.9 presents dose rates at various distances from sample ISFSI arrays for the design basis burnup and cooling time which results in the highest off-site dose for the combination of maximum burnup and minimum cooling times analyzed in Chapter 5. 10CFR72.106 [10.0.1] specifies that the minimum distance from the ISFSI to the controlled area boundary is 100 meters. Therefore this was the minimum distance analyzed in Chapter 5. As a summary of Chapter 5, Table 10.4.1 presents the annual dose results for a single overpack at 100 and ~~200~~*-250* meters and a 2x5 array of HI-STORM 100 systems at ~~350~~*-450* meters. These annual doses are based on a full array of design basis fuel with a burnup of ~~52~~*47*,500 MWD/MTU and ~~5~~*3*-year cooling. This burnup and cooling time combination conservatively bounds the allowable burnup and cooling times listed in ~~the Technical Specifications~~*Appendix B to the CoCSection 2.1.9*. In addition, 100% occupancy (8760 hours) is conservatively assumed. In the calculation of the annual dose, the casks were positioned on an infinite slab of soil to account for earth-shine effects. These results indicate that the calculated annual dose is less than the regulatory limit of 25 mrem/year at a distance of ~~200~~*-250* meters for a single cask and at ~~350~~*-450* meters for a 2x5 array of HI-STORM 100 Systems containing design basis fuel. These results are presented only as an illustration to demonstrate that the HI-STORM 100 System is in compliance with 10CFR72.104[10.0.1]. Neither the distances nor the array configurations become part of the Technical Specifications. Rather, users are required to perform a site specific analyses to demonstrate compliance with 10CFR72.104[10.0.1] contributors and 10CFR20[10.1.1].

An additional contributor to the controlled area boundary dose is the loaded HI-TRAC transfer cask, if the HI-TRAC is to be used at the ISFSI outside of the fuel building. Table 10.4.2 provides dose rates at 100, 200, and 300 meters for a 100-ton HI-TRAC transfer cask loaded with design basis fuel. The 100-ton HI-TRAC dose rates bound the 125-ton HI-TRAC by large margins. Based on the short duration that the loaded HI-

TRAC is used outside at the ISFSI, the HI-STORM 100 System is in compliance with 10CFR72.104[10.0.1] when worst-case design basis fuel is loaded in all fuel cell locations. However, users are required to perform a site specific analysis to demonstrate compliance with 10CFR72.104[10.0.1] and 10CFR20[10.1.1] taking into account the actual site boundary distance and fuel characteristics.

Section 7.1 provides a discussion as to how the Holtec MPC design, welding, testing, and inspection requirements meet the guidance of ISG-18 such that leakage from the confinement boundary may be considered non-credible. Therefore, there is no additional dose contribution due to leakage from the welded MPC. ~~A minor contributor to the minimum controlled area boundary is the normal storage condition leakage from the welded MPC. Although leakage is not expected, Section 7.2 provides an analysis for the annual dose equivalent based on a continuous leak from the MPC. The annual dose equivalent to an individual at the minimum controlled area boundary based on the assumed leakage rate and continuous occupancy is presented in Table 7.3.8.~~ The site licensee is required to perform a site-specific dose evaluation of all dose contributors as part of the ISFSI design. This evaluation will account for the location of the controlled area boundary, the total number of casks on the ISFSI and the effects of the radiation from uranium fuel cycle operations within the region.

10.4.2 Controlled Area Boundary Dose for Off-Normal Conditions

As demonstrated in Section 11.1, the postulated off-normal conditions (off-normal pressure, off-normal environmental temperatures, leakage of one MPC weld, partial blockage of air inlets, and off-normal handling of HI-TRAC) do not result in the degradation of the HI-STORM 100 System shielding effectiveness. Therefore, the dose at the controlled area boundary from direct radiation for off-normal conditions is equal to that of normal conditions.

~~However, the annual dose at the controlled area boundary as a result of an assumed effluent release under off-normal conditions is different than that under normal conditions. Under off-normal conditions, 10% of the fuel rods are assumed to have been breached, in lieu of 1% of the fuel rods for normal conditions. The resulting annual dose equivalent to an individual at the minimum controlled area boundary, based on the assumed leakage rate and continuous occupancy, is presented in Table 7.3.8. The analysis to determine the off-normal dose at the controlled area boundary is described in Section 7.2.~~

10.4.3 Controlled Area Boundary Dose for Accident Conditions

10CFR72.106 [10.0.1] specifies that the maximum doses allowed to any individual at the controlled area boundary from any design basis accident (See Subsection 10.1.2). In

addition, it is specified that the minimum distance from the ISFSI to the controlled area boundary be at least 100 meters.

~~Subsection 7.3 *and Table 7.3.8* demonstrates that the resultant doses for a non-mechanistic postulated breach of the MPC confinement boundary at the regulatory minimum site boundary distance of 100 meters is presented in Table 7.3.8 within the regulatory limits specified in 10CFR72.106 [10.0.1].~~

Chapter 11 presents the results of the evaluations performed to demonstrate that the HI-STORM 100 System can withstand the effects of all accident conditions and natural phenomena without the corresponding radiation doses exceeding the requirements of 10CFR72.106 [10.0.1]. The accident events addressed in Chapter 11 include: handling accidents, tip-over, fire, tornado, flood, earthquake, 100 percent fuel rod rupture, confinement boundary leakage, explosion, lightning, burial under debris, extreme environmental temperature, partial blockage of MPC basket air inlets, and 100% blockage of air inlets.

The worst-case shielding consequence of the accidents evaluated in Section 11.2 for the loaded HI-STORM overpack assumes that as a result of a fire, the outer-most one inch of the concrete experiences temperatures above the concrete's design temperature. Therefore, the shielding effectiveness of this outer-most one inch of concrete is degraded. However, with over 25 inches of concrete providing shielding, the loss of one inch will have a negligible effect on the dose at the controlled area boundary.

The worst case shielding consequence of the accidents evaluated in Section 11.2 for the loaded HI-TRAC transfer cask assumes that as a result of a fire, tornado missile, or handling accident, all of the water in the water jacket is lost. The shielding analysis of the 100-ton HI-TRAC transfer cask with complete loss of the water from the water jacket is discussed in Section 5.1.2. These results bound those for the 125-Ton HI-TRAC transfer cask by a large margin. The results in that section show that the resultant dose rate at the 100-meter controlled area boundary would be approximately ~~1.474~~*.06* mrem/hour for the loaded HI-TRAC transfer cask during the accident condition. At the calculated dose rate, it would take approximately ~~141~~*51* days for the dose at the controlled area boundary to reach 5 rem. This length of time is sufficient to implement and complete the corrective actions outlined in Chapter 11. Therefore, the dose requirement of 10CFR72.106 [10.0.1] is satisfied. Once again, this dose is calculated assuming design basis fuel in all fuel cell locations. Users will need to perform site-specific analysis considering the actual site boundary distance and fuel characteristics.

Table 10.4.1

ANNUAL DOSE FOR ARRAYS OF HI-STORM 100 OVERPACKS
WITH DESIGN BASIS ZIRCALOY CLAD FUEL
~~52~~47,500 MWD/MTU AND ~~5~~3-YEAR COOLING

Array Configuration	1 Cask	1 Cask	2x5 Array
Annual Dose (mrem/year)[†]	~~130.0290.3~~ *307.9*	~~20.1922.83~~ *24.10*	~~18.6415.52~~ *16.29*
Distance to Controlled Area Boundary (meters)[††], [†††]	100	~~200~~*250*	~~350~~*450*

[†] 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 for ~~5~~3-year cooling as specified in ~~the Technical Specifications~~*Appendix B to the* ~~CoC~~*Section 2.1.9* is lower than the burnup analyzed for the design basis fuel used in this table.

Table 10.4.2
DOSE RATE FOR THE 100-TON HI-TRAC TRANSFER CASK
WITH DESIGN BASIS ZIRCALOY CLAD FUEL

Fuel Burnup & Cooling Time	100 Meters	200 Meters	300 Meters
~~42,500~~*46,000* MWD/MTU & ~~5~~*3* Years	~~0.43~~*0.98* mrem/hr	~~0.07~~*0.15* mrem/hr	~~0.02~~*0.04* mrem/hr
~~52,500~~*75,000* MWD/MTU & ~~10~~ *5* Years	~~0.27~~*0.80* mrem/hr	~~0.04~~*0.12* mrem/hr	~~0.01~~*0.03* mrem/hr

CHAPTER 11†: ACCIDENT ANALYSIS

This chapter presents the evaluation of the HI-STORM 100 System for the effects of off-normal and postulated accident conditions. The design basis off-normal and postulated accident events, including those resulting from mechanistic and non-mechanistic causes as well as those caused by natural phenomena, are identified in Sections 2.2.2 and 2.2.3. For each postulated event, the event cause, means of detection, consequences, and corrective action are discussed and evaluated. As applicable, the evaluation of consequences includes structural, thermal, shielding, criticality, confinement, and radiation protection evaluations for the effects of each design event.

The structural, thermal, shielding, criticality, and confinement features and performance of the HI-STORM 100 System are discussed in Chapters 3, 4, 5, 6, and 7. The evaluations provided in this chapter are based on the design features and evaluations described therein.

Chapter 11 is in full compliance with NUREG-1536; no exceptions are taken.

11.1 OFF-NORMAL CONDITIONS

~~During normal storage operations of the HI-STORM 100 System it is possible that an off-normal situation could occur.~~ Off-normal operations, as defined in accordance with ANSI/ANS-57.9, are those conditions which, although not occurring regularly, are expected to occur no more than once a year. In this section, design events pertaining to off-normal operation for expected operational occurrences are considered. The off-normal conditions are listed in Subsection 2.2.2.

The following off-normal operation events have been considered in the design of the HI-STORM 100:

Off-Normal Pressures
Off-Normal Environmental Temperatures
Leakage of One MPC Seal Weld
Partial Blockage of Air Inlets
Off-Normal Handling of HI-TRAC Transfer Cask

For each event, the postulated cause of the event, detection of the event, analysis of the event effects and consequences, corrective actions, and radiological impact from the event are presented.

---

† This chapter has been prepared in the format and section organization set forth in Regulatory Guide 3.61. However, the material content of this chapter also fulfills the requirements of NUREG-1536. Pagination and numbering of sections, figures, and tables are consistent with the convention set down in Chapter 1, Section 1.0, herein. Finally, all terms-of-art used in this chapter are consistent with the terminology of the glossary (Table 1.0.1) and component nomenclature of the Bill-of-Materials (Section 1.5).

The results of the evaluations performed herein demonstrate that the HI-STORM 100 System can withstand the effects of off-normal events without affecting function, and are in compliance with the applicable acceptance criteria. The following sections present the evaluation of the HI-STORM 100 System for the design basis off-normal conditions that demonstrate that the requirements of 10CFR72.122 are satisfied, and that the corresponding radiation doses satisfy the requirements of 10CFR72.106(b) and 10CFR20.

The load combinations evaluated for off-normal conditions are defined in Table 2.2.14. The load combinations include both normal and off-normal loads. The off-normal load combination evaluations are discussed in Section 11.1.5.

11.1.1 Off-Normal Pressures

The sole pressure boundary in the HI-STORM 100 System is the MPC ~~internal pressure boundary~~*enclosure vessel*. The off-normal pressure condition is specified in Section 2.2.2.1. The off-normal pressure for the MPC internal cavity is a function of the initial helium fill pressure and the temperature obtained with maximum decay heat load design basis fuel. The maximum off-normal environmental temperature is 100°F with full solar insolation. The MPC internal pressure *is evaluated with* ~~is further increased by the conservative assumption that~~ 10% of the fuel rods rupture*d* and 100% of the *rods* fill gas~~,~~ and 30% of the fission gases ~~are~~ released to the cavity.

11.1.1.1 Postulated Cause of Off-Normal Pressure

After fuel assembly loading, the MPC is drained, dried, and backfilled with an inert gas (helium) to assure long-term fuel cladding integrity during dry storage. Therefore, the probability of failure of intact fuel rods in dry storage is low. Nonetheless, the event is postulated and evaluated.

11.1.1.2 Detection of Off-Normal Pressure

The HI-STORM 100 System is designed to withstand the MPC off-normal internal pressure without any effects on its ability to meet its safety requirements. There is no requirement for detection of off-normal pressure and, therefore, no monitoring is required.

11.1.1.3 Analysis of Effects and Consequences of Off-Normal Pressure

Chapter 4 calculates the MPC internal pressure with an ambient temperature of 80°F, 10% fuel rods ruptured, full insolation, and maximum decay heat, and reports the maximum value of ~~*75.0*~~*106.6* psig in Table 4.4.14 at an average temperature of ~~*513.6*~~ *522.8*°K. Using this pressure, the off-normal temperature of 100°F (*bounding temperature rise* ~~ΔT~~ of 20°F or 11.1°K), and the ideal gas law, the off-normal resultant pressure ~~is~~ *(*calculated *below)* ~~to be~~ *is* below the *off*-normal condition MPC internal design pressure *(Table 2.2.1 in Chapter 2).*

$$\frac{P_1}{P_2} = \frac{T_1}{T_2}$$

$$P_2 = \frac{P_1 T_2}{T_1}$$

$$P_2 = \frac{(106.6\,\text{psig} + 14.7)\,(522.8°\text{K} + 11.1°\text{K})}{522.8°\text{K}}$$

$$P_2 = 123.9\,\text{psia or } 109.2\,\text{psig}$$

It should be noted that this bounding temperature rise does not take any credit for the increase in thermosiphon action that would accompany the pressure increase that results from both the temperature rise and the addition of the gaseous fission products to the MPC cavity. As any such increase in thermosiphon action would decrease the temperature rise, the calculated pressure is higher than would actually occur.~~The off-normal MPC internal design pressure of 100 psig (Table 2.2.1) has been established to bound the off-normal condition. Therefore, no additional analysis is required.~~

Structural

The structural evaluation of the MPC enclosure vessel for off-normal internal pressure conditions is ~~equivalent to the evaluation at normal internal pressures, since the normal design pressure was set at a value which would encompass the off-normal pressure. Therefore, the resulting stresses from the off-normal condition are equivalent to that of the normal condition and are well within the short-term allowable values, as~~ discussed in Section 3.4. *The stresses resulting from the off-normal pressure are confirmed to be bounded by the applicable pressure boundary stress limits.*

Thermal

The MPC internal pressure for off-normal conditions is calculated as presented above. As can be seen from the value above, the ~~100 psig~~ design basis internal pressure for off-normal conditions used in the structural evaluation *(Table 2.2.1)* bounds the calculated value above.

Shielding

There is no effect on the shielding performance of the system as a result of this off-normal event.

Criticality

There is no effect on the criticality control features of the system as a result of this off-normal event.

Confinement

There is no effect on the confinement function of the MPC as a result of this off-normal event. As discussed in the structural evaluation above, all stresses remain within allowable values, assuring confinement boundary integrity.

Radiation Protection

Since there is no degradation in shielding or confinement capabilities as discussed above, there is no effect on occupational or public exposures as a result of this off-normal event.

Based on this evaluation, it is concluded that the off-normal pressure does not affect the safe operation of the HI-STORM 100 System.

11.1.1.4 Corrective Action for Off-Normal Pressure

The HI-STORM 100 System is designed to withstand the off-normal pressure without any effects on its ability to maintain safe storage conditions. There is no corrective action requirement for off-normal pressure.

11.1.1.5 Radiological Impact of Off-Normal Pressure

The event of off-normal pressure has no radiological impact because the confinement barrier and shielding integrity are not affected.

11.1.2 Off-Normal Environmental Temperatures

The HI-STORM 100 System is designed for use at any site in the United States. Off-normal environmental temperatures of -40 to 100°F (HI-STORM overpack) and 0 to 100°F (HI-TRAC transfer cask) have been conservatively selected to bound off-normal temperatures at these sites. The off-normal temperature range affects the entire HI-STORM 100 System and must be evaluated against the allowable component design temperatures. ~~This~~*The* off-normal ~~event is of a short duration, therefore the resultant~~ temperatures are evaluated against the ~~accident~~ *off-normal* condition temperature limits ~~as~~ listed in Table 2.2.3.

11.1.2.1 Postulated Cause of Off-Normal Environmental Temperatures

The off-normal environmental temperature is postulated as a constant ambient temperature caused by extreme weather conditions. To determine the effects of the off-normal temperatures, it is conservatively assumed that these temperatures persist for a sufficient duration to allow the HI-STORM 100 System to achieve thermal equilibrium. Because of the large mass of the HI-STORM 100 System with its corresponding large thermal inertia and the limited duration for the off-normal temperatures, this assumption is conservative.

11.1.2.2 Detection of Off-Normal Environmental Temperatures

The HI-STORM 100 System is designed to withstand the off-normal environmental temperatures without any effects on its ability to maintain safe storage conditions. There is no requirement for detection of off-normal environmental temperatures for the HI-STORM overpack and MPC. Chapter 2 provides operational limitations to the use of the HI-TRAC transfer cask at temperatures of ≤32°F and prohibits use of the HI-TRAC transfer cask below 0°F.

11.1.2.3 Analysis of Effects and Consequences of Off-Normal Environmental Temperatures

The off-normal event considering an environmental temperature of 100°F for a duration sufficient to reach thermal equilibrium is evaluated with respect to design temperatures listed in Table 2.2.3. The evaluation is performed with design basis fuel with the maximum decay heat and the most restrictive thermal resistance. The 100°F environmental temperature is applied with full solar insolation.

The HI-STORM 100 System maximum temperatures for components close to the design basis temperatures are listed in Subsection 4.4. These temperatures are conservatively calculated at an environmental temperature of 80°F. The maximum off-normal environmental temperature is 100°F, which is an increase of 20°F. ~~Including the effect of a hypothetical 10% rod rupture condition on the MPC cavity gas conductivity, c~~*Including this* ~~Conservatively~~ *as a* bounding temperatures *increment* ~~*for all MPC designs (Table 1.2.1)* of the MPC-68 and MPC-24~~ *over the 80°F ambient temperature solutions of Chapter 4,* ~~are calculated to be as listed in~~ *the HI-STORM temperatures are computed and provided in* Table 11.1.1. As illustrated by the table, all the maximum off-normal temperatures are below the ~~short-term condition~~ *off-normal* design ~~basis~~ temperatures *for the HI-STORM System (Table 2.2.3).* The maximum temperatures are the peak values and are based on the conservative assumptions applied in this analysis. The component temperatures for the HI-TRAC listed in Table 4.5.2 are all based on the maximum off-normal environmental temperature. The off-normal environmental temperature is of a short duration (several consecutive days would be highly unlikely) and the resultant temperatures are evaluated against short-term temperature limits. Therefore, all the HI-STORM 100 System maximum off-normal temperatures meet the design requirements.

Additionally, the off-normal environmental temperature generates a pressure that is *bounded by that* evaluated in Subsection 11.1.1. The off-normal MPC cavity pressure is less than the design basis pressure listed in Table 2.2.1.

The off-normal event considering an environmental temperature of -40°F and no solar insolation for a duration sufficient to reach thermal equilibrium is evaluated with respect to material design temperatures of the HI-STORM overpack. The HI-STORM overpack and MPC are conservatively assumed to reach -40°F throughout the structure. The minimum off-normal environmental temperature specified for the HI-TRAC transfer cask is 0°F and the HI-TRAC is conservatively assumed to reach 0°F throughout the structure. For ambient temperatures from 0° to 32°F, ~~a 25%~~

~~ethylene glycol solution is~~*antifreeze must be* added to the demineralized water in the water jacket to prevent freezing. Chapter 3, Subsection 3.1.2.3, details the structural analysis and testing performed to assure prevention of brittle fracture failure of the HI-STORM 100 System.

Structural

The effect on the MPC for the upper off-normal thermal conditions (i.e., 100°F) is an increase in the internal pressure. As shown in Subsection 11.1.1.3, the resultant pressure is ~~well~~ below the *off-normal* design pressure~~-~~*(Table 2.2.1 in Chapter 2)* ~~of 100 psig used in the structural analysis~~. The effect of the lower off-normal thermal conditions (i.e., -40°F) ~~results in~~ *requires* an evaluation of the potential for brittle fracture. ~~That~~*Such an evaluation* is ~~discussed~~ *presented* in Section 3.1.2.3.

Thermal

The resulting off-normal system and fuel assembly cladding temperatures for the hot conditions are provided in Table 11.1.1 for the HI-STORM overpack and MPC. As can be seen from this table, all temperatures for off-normal conditions are within the short-term allowable values ~~described~~*listed* in Table 2.2.3.

Shielding

There is no effect on the shielding performance of the system as a result of this off-normal event.

Criticality

There is no effect on the criticality control features of the system as a result of this off-normal event.

Confinement

There is no effect on the confinement function of the MPC as a result of this off-normal event.

Radiation Protection

Since there is no degradation in shielding or confinement capabilities as discussed above, there is no effect on occupational or public exposures as a result of this off-normal event.

Based on this evaluation, it is concluded that the specified off-normal environmental temperatures do not affect the safe operation of the HI-STORM 100 System.

11.1.2.4 Corrective Action for Off-Normal Environmental Temperatures

The HI-STORM 100 System is designed to withstand the off-normal environmental temperatures without any effects on its ability to maintain safe storage conditions. There are no corrective actions required for off-normal environmental temperatures.

11.1.2.5 Radiological Impact of Off-Normal Environmental Temperatures

Off-normal environmental temperatures have no radiological impact, as the confinement barrier and shielding integrity are not affected.

11.1.3 Leakage of One Seal

The HI-STORM 100 System has a reliable welded boundary to contain radioactive fission products within the confinement boundary. The radioactivity confinement boundary is defined by the MPC shell, baseplate, MPC lid, and vent and drain port cover plates. The closure ring provides a redundant welded closure to the release of radioactive material from the MPC cavity through the field-welded MPC lid closures. Confinement boundary welds are inspected by radiography or ultrasonic examination except for field welds that are examined by the liquid penetrant method on the root (for multi-pass welds) and final pass, at a minimum. Field welds are performed on the MPC lid, the MPC vent and drain port covers, and the MPC closure ring. ~~The welds on the MPC lid, and vent and drain port covers are leakage tested.~~ Additionally, the MPC lid weld is subjected to a hydrostatic test to verify its integrity.

Section 7.1 provides a discussion as to how the MPC design, welding, testing and inspection requirements meet the guidance of ISG-18 such that leakage from the confinement boundary may be considered non-credible.
~~The MPC lid-to-MPC shell weld is postulated to fail to confirm the safety of the HI-STORM 100 confinement boundary. The failure of the MPC lid weld is equivalent to the MPC drain or vent port cover weld failing. The MPC lid-to-shell weld has been selected because it is the main closure weld performed in the field for the MPC. It is extremely unlikely that the weld examination, helium leakage testing and hydrostatic testing would fail to detect a poorly welded closure plate. The MPC lid weld failure affects the MPC confinement boundary; however, no leakage will occur.~~

11.1.3.1 Postulated Cause of Leakage of One Seal in the Confinement Boundary

There is no credible cause for the leakage of one seal in the confinement boundary. The conditions analyzed in Chapter 3 shows that the confinement boundary components are maintained within their Code-allowable stress limits under all normal and off-normal storage conditions. The MPC fabrication and closure welds meet the requirements of ISG-18, such that leakage from the confinement boundary is not considered credible. Therefore, there is no event that could cause leakage of one seal in the confinement boundary.

~~Failure of the MPC confinement boundary is highly unlikely. The MPC confinement boundary is shown to withstand all normal, off-normal, and accident conditions. There are no credible conditions that could damage the integrity of the MPC confinement boundary. The MPC lid-to-MPC shell weld is liquid penetrant inspected on the root and final pass, volumetrically inspected or liquid penetrant inspected on multiple passes, hydrostatically tested, and helium leak tested. The initial integrity of the closure welds will be maintained throughout the design life because the MPC is stored within the HI-STORM overpack which provides physical protection and a weather shield. Failure of the MPC lid-to-MPC shell weld would require all of the following:~~

~~1. Improper weld by a qualified welding machine or welder using approved welding procedures.~~

~~2. Failure to detect the unacceptable indication during the liquid penetrant or volumetric inspections performed by a qualified inspector in accordance with approved procedures.~~

~~3. Failure of the qualified leakage test equipment to detect the leak in accordance with approved procedures.~~

~~4. Failure to detect the unacceptable leak during the hydrostatic test performed by qualified personnel in accordance with approved procedures.~~

~~The evaluation of the failure of the MPC lid-to-MPC shell weld has been postulated to demonstrate the safety of the HI-STORM 100 confinement system and cannot be derived from a credible loading condition.~~

11.1.3.2 Detection of Leakage of One Seal in the Confinement Boundary

The HI-STORM 100 System is designed ~~to withstand the~~*such that* leakage of one field weld in the confinement boundary ~~without any effects on its ability to meet its safety requirements~~*is not considered a credible scenario*. ~~As the HI-STORM 100 System can withstand the failure of one field weld with no leakage~~*Therefore*, there is no requirement to detect leakage from one seal.

~~11.1.3.3 Analysis of Effects and Consequences of Leakage of One Seal in the Confinement Boundary~~

~~If the MPC lid-to-MPC shell weld were to fail, the MPC closure ring will retain the design pressure. The analysis of the MPC closure ring's ability to retain the design pressure is provided in Appendix 3.E of the HI-STAR TSAR Docket Number 72-1008. The consequences of the MPC lid-to-MPC shell weld failure are that the MPC closure ring maintains the integrity of the confinement boundary.~~

~~Structural~~

~~The stress evaluation of the closure ring is discussed in Appendix 3.E *of the HI-STAR TSAR Docket Number 72-1008*. All stresses are within the allowable values.~~

~~Thermal~~

~~There is no effect on the thermal performance of the system as a result of this off-normal event.~~

~~Shielding~~

~~There is no effect on the shielding performance of the system as a result of this off-normal event.~~

~~Criticality~~

~~There is no effect on the criticality control features of the system as a result of this off-normal event.~~

~~Confinement~~

~~There is no effect on the confinement function of the MPC as a result of this off-normal event.~~

~~Radiation Protection~~

~~Since there is no degradation in shielding or confinement capabilities as discussed above, there is no effect on occupational or public exposures as a result of this off-normal event.~~

~~Based on this evaluation, it is concluded that the specified off-normal leakage of one seal event does not affect the safe operation of the HI-STORM 100 System.~~

11.1.3.~~3~~4 Corrective Action for Leakage of One Seal in the Confinement Boundary

There is no corrective action required for the failure of one weld in the closure system of the confinement boundary. Leakage of one weld in the confinement boundary closure system ~~does not affect the HI-STORM 100 System's ability to operate safely~~*is not a credible event.*

11.1.3.~~4~~5 Radiological Impact of Leakage of One Seal in the Confinement Boundary

The off-normal event of the failure of one weld in the confinement boundary closure system has no radiological impact because ~~the~~ *leakage from the* confinement barrier is not ~~breached and shielding is not affected~~*credible.*

11.1.4 Partial Blockage of Air Inlets

The HI-STORM 100 System is designed with fine mesh screens on the inlet and outlet air ducts. These screens ensure the air ducts are protected from the incursion of foreign objects. There are four air inlet ducts 90° apart and it is highly unlikely that blowing debris during normal or off-normal

operation could block all air inlet ducts. As required by the design criteria presented in Chapter 2, it is conservatively assumed that two of the four air inlet ducts are blocked. The blocked air inlet ducts are assumed to be completely blocked with an ambient temperature of 80°F (Table 2.2.2), full solar insolation, and maximum SNF decay heat values. This condition is analyzed to demonstrate the inherent thermal stability of the HI-STORM 100 System.
~~An additional evaluation is performed with three of the four air inlet ducts. While not required by the HI-STORM System design criteria, this additional evaluation is performed as a parametric study of the effects of incremental duct blockage. The purpose of the parametric study is to demonstrate the robustness of the HI-STORM System design beyond the design basis.~~

11.1.4.1 Postulated Cause of Partial Blockage of Air Inlets

It is conservatively assumed that the blocked air inlet ducts are completely blocked, although ~~mesh~~ screens prevent foreign objects from entering the ducts. The mesh screens are either inspected periodically or the outlet duct air temperature is monitored~~ as specified by Technical Specifications in Appendix A to the CoC~~. It is, however, possible that blowing debris may block two air inlet ducts of the overpack. ~~As already stated, the blockage of three inlet ducts is evaluated only to demonstrate the limited effects of additional incremental duct blockage.~~

11.1.4.2 Detection of Partial Blockage of Air Inlets

The detection of the partial blockage of air inlet ducts will occur during the routine visual inspection of the mesh screens or temperature monitoring of the outlet duct air~~ as required and specified by Technical Specifications in Appendix A to the CoC~~. The frequency of inspection is based on an assumed complete blockage of all four air inlet ducts. There is no inspection requirement as a result of the postulated two inlet duct blockage, because the complete blockage of all four air inlet ducts is bounding.

11.1.4.3 Analysis of Effects and Consequences of Partial Blockage of Air Inlets

~~Evaluations for~~*The* two inlet ducts ~~and three inlet ducts~~ blocked ~~are~~*condition is* evaluated for the *hottest MPC-68*~~ MPC-32 at its maximum decay heat load. Only the MPC-32 is evaluated because it has the highest decay heat load of all MPC designs (Table 1.2.1)~~. The largest temperature rise of the MPC or its contents as a result of the blockage of two air inlet ducts is ~~25~~ *18*°F, for the *fuel cladding. The results for the HI-STORM System are provided in Table 11.1.2.*~~ MPC shell The largest temperature rise of the MPC or its contents as a result of the blockage of three air inlet ducts (performed as a parametric study of incremental duct blockage only) is 81°F, also for the MPC shell. Conservatively adding the largest component temperature rise to all cask system component temperatures, the resultant bounding temperatures for the complete blockage of two air inlet ducts are provided in Table 11.1.2. Following this same procedure of adding the largest component temperature rise to all cask system component temperatures, the resultant bounding temperatures for the complete blockage of three air inlet ducts are included in the same table for comparison purposes. These values are based on full insolation and an ambient temperature of 80°F. The analysis method~~

~~for the blockage of two and three of the air inlet ducts is conservative with respect to the analysis method for the normal condition. As a result of the air inlet duct blockages, the head loss is increased and the airflow is decreased thereby increasing component temperatures.~~

As stated above, the largest temperature rise of the MPC or its contents as a result of the blockage of two air inlet ducts is ~~*2518*~~°F, for the *fuel cladding.*~~MPC shell~~ *This is bounded by the 20°F temperature rise in Subsection 11.1.1, so the* ~~A bounding~~ MPC internal pressure *calculated therein bounds the partial vents blockage condition as well.*
~~as a result of this calculated temperature increase is computed, based on initial conditions presented previously in Subsection 11.1.1.3, as follows:~~

$$\cancel{P_2 = P_1 \frac{T_1 + \Delta T}{T_1}}$$

~~where:~~

~~P_2 = Bounding MPC Cavity Pressure (psia)~~
~~P_1 = Initial MPC Cavity Pressure (89.7 psia)~~
~~T_1 = Initial MPC Cavity Average Temperature (513.6°K)~~
~~ΔT = Bounding MPC Temperature Rise (25°F or 19.9°K)~~

~~Substituting these values into the equation above, the bounding MPC internal pressure is obtained as:~~

$$\cancel{P_2 = 89.7 \times \frac{513.6 + 13.9}{513.6} = 92.1 psia = 77.4 psig}$$

~~The off-normal MPC internal design pressure of 100 psig (Table 2.2.1) has been established to bounds this partial inlet duct blockage condition.~~

~~Although it is a beyond the design basis condition, the bounding pressure rise for the three blocked air inlet ducts condition can be determine in the same manner. As stated above, the bounding temperature rise for this condition is 81°F (44.9°K), and the corresponding bounding MPC internal pressure is 97.5 psia (*82.8* psig). This parametric evaluation demonstrates the insensitivity of the MPC internal pressure to incremental duct blockage, as the relatively large incremental flow area reduction increases the pressure by only *5.4* psi.~~

Structural

There are no structural consequences as a result of this off-normal event.

Thermal

Using the methodology and model discussed in Section 4.4, the thermal analysis for the two air inlet ducts blocked off-normal condition is performed. The analysis demonstrates that under steady-state conditions, no system components exceed the short-term allowable temperatures in Table 2.2.3.

~~The parametric study of incremental duct blockage, performed by evaluating a three air inlet ducts blocked condition, demonstrates the insensitivity of the system to relatively large incremental flow area reductions. This beyond the design basis condition results in relatively small temperature increases and temperatures well below the short-term allowable temperatures in Table 2.2.3, even though no such requirement exists.~~

Shielding

There is no effect on the shielding performance of the system as a result of this off-normal event.

Criticality

There is no effect on the criticality control features of the system as a result of this off-normal event.

Confinement

There is no effect on the confinement function of the MPC as a result of this off-normal event.

Radiation Protection

Since there is no degradation in shielding or confinement capabilities as discussed above, there is no effect on occupational or public exposures as a result of this off-normal event.

Based on this evaluation, it is concluded that the specified off-normal partial blockage of air inlet ducts event does not affect the safe operation of the HI-STORM 100 System.

11.1.4.4 Corrective Action for Partial Blockage of Air Inlets

The corrective action for the partial blockage of air inlet ducts is the removal, cleaning, and replacement of the affected mesh screens. After clearing of the blockage, the storage module temperatures will return to the normal temperatures reported in Chapter 4. Partial blockage of air inlet ducts does not affect the HI-STORM 100 System's ability to operate safely.

Periodic i~~I~~nspection of the HI-STORM overpack air duct screen covers is required~~ with the frequency specified by Technical Specifications in Appendix A to the CoC,~~. ~~a~~*A*lternatively, the outlet duct air temperature is monitored. The frequency of inspection is based on an assumed blockage of all four air inlet ducts analyzed in Subsection 11.2.

11.1.4.5 Radiological Impact of Partial Blockage of Air Inlets

The off-normal event of partial blockage of the air inlet ducts has no radiological impact because the confinement barrier is not breached and shielding is not affected.

11.1.5 Off-Normal Handling of HI-TRAC

During upending and/or downending of the HI-TRAC transfer cask, the total lifted weight is distributed among both the upper lifting trunnions and the lower pocket trunnions. Each of the four trunnions on the HI-TRAC therefore supports approximately one-quarter of the total weight. This even distribution of the load would continue during the entire rotation operation.

If the lifting device is allowed to "go slack", the total weight would be applied to the lower pocket trunnions only. Under this off-normal condition, the pocket trunnions would each be required to support one-half of the total weight, doubling the load per trunnion. This condition is analyzed to demonstrate that the pocket trunnions possess sufficient strength to support the increased load under this off-normal condition.

This off-normal condition does not apply to the HI-TRAC 125D, which does not have lower pocket trunnions. Upending and downending of the HI-TRAC 125D is performed using an L-frame.

11.1.5.1 Postulated Cause of Off-Normal Handling of HI-TRAC

If the cable of the crane handling the HI-TRAC is inclined from the vertical, it would possible to unload the upper lifting trunnions such that the lower pocket trunnions are supporting the total cask weight and the lifting trunnions are only preventing cask rotation.

11.1.5.2 Detection of Off-Normal Handling of HI-TRAC

Handling procedures and standard rigging practice call for maintaining the crane cable in a vertical position by keeping the crane trolley centered over the lifting trunnions. In such an orientation it is not possible to completely unload the lifting trunnions without inducing rotation. If the crane cable were inclined from the vertical, however, the possibility of unloading the lifting trunnions would exist. It is therefore possible to detect the potential for this off-normal condition by monitoring the incline of the crane cable with respect to the vertical.

11.1.5.3 Analysis of Effects and Consequences of Off-Normal Handling of HI-TRAC

If the upper lifting trunnions are unloaded, the lower pocket trunnions will support the total weight of the loaded HI-TRAC. The analysis of the pocket trunnions to support the applied load of one-half of the total weight is provided in ~~Appendices 3.AA and 3.AI~~ *Subsection 3.4.4.3.3.1* of this FSAR. The consequence of off-normal handling of the HI-TRAC is that the pocket trunnions safely support the applied load.

Structural

The stress evaluations of the lower pocket trunnions are discussed in *Subsection 3.4.4.3.3.1 of this FSAR* ~~Appendices 3.AA and 3.AI~~. All stresses are within the allowable values.

Thermal

There is no effect on the thermal performance of the system as a result of this off-normal event.

Shielding

There is no effect on the shielding performance of the system as a result of this off-normal event.

Criticality

There is no effect on the criticality control features of the system as a result of this off-normal event.

Confinement

There is no effect on the confinement function of the MPC as a result of this off-normal event.

Radiation Protection

Since there is no degradation in shielding or confinement capabilities as discussed above, there is no effect on occupational or public exposures as a result of this off-normal event.

Based on this evaluation, it is concluded that the specified off-normal handling of the HI-TRAC does not affect the safe operation of the system.

11.1.5.4 Corrective Action for Off-Normal Handling of HI-TRAC

The HI-TRAC transfer casks are designed to withstand the off-normal handling condition without any adverse effects. There are no corrective actions required for off-normal handling of HI-TRAC other than to attempt to maintain the crane cable vertical during HI-TRAC upending or downending.

11.1.5.5 Radiological Consequences of Off-Normal Handling of HI-TRAC

The off-normal event of off-normal handling of HI-TRAC has no radiological impact because the confinement barrier is not breached and shielding is not affected.

11.1.6 Off-Normal Load Combinations

Load combinations for off-normal conditions are provided in Table 2.2.14. The load combinations include normal loads with the off-normal loads. The load combination results are shown in Section 3.4 to meet all allowable values.

Table 11.1.1

MAXIMUM TEMPERATURES CAUSED BY OFF-NORMAL ENVIRONMENTAL TEMPERATURES

Location	**Temperature [°F]**	~~**Design Basis Limits [°F]**~~
Fuel Cladding	*751* ~~*711* (PWR)~~ ~~*760* (BWR)~~	~~1058 short-term~~
MPC Basket	~~*740*~~*726*	~~950 short-term~~
MPC Shell	~~*371*~~ *490*	~~775 short-term~~
~~Overpack Air Outlet~~	~~*226*~~	~~N/A~~
Overpack Inner Shell	~~*219*~~ *263*	~~350 short-term (overpack concrete)~~
Overpack Outer Shell	~~*165*~~*203*	~~350 short-term (overpack concrete)~~

Table 11.1.2

BOUNDING[‡] TEMPERATURES CAUSED BY PARTIAL BLOCKAGE OF AIR INLET DUCTS [°F]

Temperature Location	**No Blockage of Inlet Ducts**	**Partial Blockage of Inlet Ducts**		~~**Off-Normal Design Basis**~~
		2 Ducts Blocked	~~**3 Ducts Blocked**~~	
Fuel Cladding	~~*740*~~ *731*	~~*765*~~ *749*	~~*821*~~	~~1058 short-term~~
MPC Basket	~~*720*~~ *706*	~~*745*~~ *723*	~~*801*~~	~~950 short-term~~
MPC Shell	~~*351*~~ *470*	~~*376*~~ *486*	~~*432*~~	~~775 short-term~~
Overpack Air Outlet *(mass flow averaged)*	~~*206*~~ *210*	~~*231*~~ *214*	~~*287*~~	~~N/A~~
Overpack Inner Shell	~~*199*~~ *243*	~~*224*~~ *266*	~~*280*~~	~~*400*~~ ~~350 short-term (overpack concrete)~~
Overpack Outer Shell	~~*145*~~ *177*	~~*170*~~ *182*	~~*226*~~	~~*600*~~ ~~350 short-term (overpack concrete)~~

~~‡ *The bounding temperatures presented in this table are obtained by adding the maximum temperature rise of any cask component to the normal condition temperatures of every cask component.*~~

11.2 ACCIDENTS

Accidents, in accordance with ANSI/ANS-57.9, are either infrequent events that could reasonably be expected to occur during the lifetime of the HI-STORM 100 System or events postulated because their consequences may affect the public health and safety. Section 2.2.3 defines the design basis accidents considered. By analyzing for these design basis events, safety margins inherently provided in the HI-STORM 100 System design can be quantified.

The results of the evaluations performed herein demonstrate that the HI-STORM 100 System can withstand the effects of all credible and hypothetical accident conditions and natural phenomena without affecting safety function, and are in compliance with the acceptable criteria. The following sections present the evaluation of the design basis postulated accident conditions and natural phenomena which demonstrate that the requirements of 10CFR72.122 are satisfied, and that the corresponding radiation doses satisfy the requirements of 10CFR72.106(b) and 10CFR20.

The load combinations evaluated for postulated accident conditions are defined in Table 2.2.14. The load combinations include normal loads with the accident loads. The accident load combination evaluations are provided in Section 3.4.

11.2.1 HI-TRAC Transfer Cask Handling Accident

11.2.1.1 Cause of HI-TRAC Transfer Cask Handling Accident

During the operation of the HI-STORM 100 System, the loaded HI-TRAC transfer cask can be transported to the ISFSI in the vertical or horizontal position. The loaded HI-TRAC transfer cask is typically transported by a heavy-haul vehicle that cradles the HI-TRAC horizontally or by a device with redundant drop protection that holds the HI-TRAC vertically. The height of the loaded overpack above the ground shall be limited to below the horizontal handling height limit determined in Chapter 3 ~~and specified by the Technical Specifications in Appendix A to the CoC~~ to limit the inertia loading on the cask in a horizontal drop to less than 45g's. Although a handling accident is remote, a cask drop from the horizontal handling height limit is a credible accident. A vertical drop of the loaded HI-TRAC transfer cask is not a credible accident as the loaded HI-TRAC shall be transported and handled in the vertical orientation by devices designed in accordance with the criteria specified in Subsection 2.3.3.1 as required by the Technical Specification.

11.2.1.2 HI-TRAC Transfer Cask Handling Accident Analysis

The handling accident analysis evaluates the effects of dropping the loaded HI-TRAC in the horizontal position. The analysis of the handling accident is provided in Chapter 3. The analysis shows that the ~~HI-STORM 100 System~~*HI-TRAC* meets all structural requirements and there is no adverse effect on the confinement, thermal or subcriticality performance of the contained MPC. Limited localized damage to the HI-TRAC water jacket shell and loss of the water in the water jacket may occur as a result of the handling accident. The HI-TRAC top lid and transfer lid housing (pool

lid for the HI-TRAC 125D) are demonstrated to remain attached by withstanding the maximum deceleration. The transfer lid doors (not applicable to HI-TRAC 125D) are also shown to remain closed during the drop. Limiting the inertia loading to 60g's or less ensures the fuel cladding remains intact based on dynamic impact effects on spent fuel assemblies in the literature [11.2.1]. Therefore, demonstrating that the 45g limit for the HI-TRAC transfer cask is met ensures that the fuel cladding remains intact.

Structural

The structural evaluation of the MPC for 45g's is provided in Section 3.4. As discussed in Section 3.4, the MPC stresses as a result of the HI-TRAC side drop, 45g loading, are all within allowable values.

As discussed above, the water jacket enclosure shell could be punctured which results in a loss of the water within the water jacket. Additionally, the HI-TRAC top lid, transfer lid (pool lid for the HI-TRAC 125D), and transfer lid doors (not applicable to HI-TRAC 125D) are shown to remain in position under the 45g loading. Analysis of the lead in the HI-TRAC is performed in Appendix 3.F and it is shown that there is no appreciable change in the lead shielding.

Thermal

The loss of the water in the water jacket causes the temperatures to increase slightly due to a reduction in the thermal conductivity through the HI-TRAC water jacket. The temperatures of the MPC in the HI-TRAC transfer cask as a result of the loss of water in the water jacket are presented in Table 11.2.8. As can be seen from the values in the table, the temperatures are ~~well~~ below the short-term allowable fuel cladding and material temperatures provided in Table 2.2.3 for accident conditions.

Shielding

The loss of the water in the water jacket results in an increase in the radiation dose rates at locations adjacent to the water jacket. The shielding analysis results presented in Section 5.1.2 demonstrate that the requirements of 10CFR72.106 are not exceeded. As the structural analysis demonstrates that the HI-TRAC top lid, transfer lid (pool lid for the HI-TRAC 125D), and transfer lid doors (not applicable to HI-TRAC 125D) remain in place, there is no change in the dose rates at the top and bottom of the HI-TRAC.

Criticality

There is no effect on the criticality control features of the system as a result of this accident event.

Confinement

There is no effect on the confinement function of the MPC as a result of this accident event. As discussed in the structural evaluation above, all stresses remain within allowable values, assuring confinement boundary integrity.

Radiation Protection

There is no degradation in the confinement capabilities of the MPC, as discussed above. There are increases in the local dose rates adjacent to the water jacket. The dose rate at 1 meter from the water jacket after the water is lost is calculated in Table 5.1.10. Immediately after the drop accident a radiological inspection of the HI-TRAC will be performed and temporary shielding shall be installed to limit the exposure to the public. Based on a minimum distance to the controlled area boundary of 100 meters, the *10CFR72.106* dose rate *requirements* at the controlled area boundary *(5 Rem limit)* ~~will be approximately 1.48 mrem/hr (Section 5.1.2). Therefore, it is evident, based on the short duration of the accident, that the requirements of 10CFR72.106 (5 Rem) will not be~~ *are not* exceeded *(Section 5.1.2).*

11.2.1.3 HI-TRAC Transfer Cask Handling Accident Dose Calculations

The handling accident could cause localized damage to the HI-TRAC water jacket shell and loss of the water in the water jacket as the neutron shield impacts the ground.

When the water jacket is impacted, the HI-TRAC transfer cask surface dose rate could increase. The HI-TRAC's post-accident shielding analysis presented in Section 5.1.2 assumes complete loss of the water in the water jacket and bounds the dose rates anticipated for the handling accident.

If the water jacket of the loaded HI-TRAC is damaged beyond immediate repair and the MPC is not damaged, the loaded HI-TRAC may be unloaded into a HI-STORM overpack, a HI-STAR overpack, or simply unloaded in the fuel pool. If the MPC is damaged, the loaded HI-TRAC must be returned to the fuel pool for unloading. Depending on the damage to the HI-TRAC and the current location in the loading or unloading sequence, less personnel exposure may be received by continuing to load the MPC into a HI-STORM or HI-STAR overpack. Once the MPC is placed in the HI-STORM or HI-STAR overpack, the dose rates are greatly reduced. The highest personnel exposure will result from returning the loaded HI-TRAC to the fuel pool to unload the MPC.

As a result of the loss of water from the water jacket, the dose rates at 1 meter adjacent to the water jacket mid-height increase (Table 5.1.10). Increasing the personnel exposure for each task affected by the increased dose rate adjacent to the water jacket by the ratio of the one meter dose rate increase results in a cumulative dose of less than *15*.0 person-rem, for the 125-ton HI-TRAC or 100-ton HI-TRAC. Using the ratio of the water jacket mid-height dose rates at one meter is very conservative. Dose rate at the top and bottom of the HI-TRAC water jacket would not increase as much as the peak

mid-height dose rates. In the determination of the personnel exposure, dose rates at the top and bottom of the loaded HI-TRAC are assumed to remain constant.

The analysis of the handling accident presented in Section 3.4 shows that the MPC confinement barrier will not be compromised and, therefore, there will be no release of radioactive material from the confinement vessel. Any possible rupture of the fuel cladding will have no effect on the site boundary dose rates because the magnitude of the radiation source has not changed.

11.2.1.4 HI-TRAC Transfer Cask Handling Accident Corrective Action

Following a handling accident, the ISFSI operator shall first perform a radiological and visual inspection to determine the extent of the damage to the HI-TRAC transfer cask and MPC to the maximum practical extent. As appropriate, place temporary shielding around the HI-TRAC to reduce radiation dose rates. Special handling procedures will be developed and approved by the ISFSI operator to lift and upright the HI-TRAC. Upon uprighting, the portion of the overpack not previously accessible shall be radiologically and visually inspected. If damage to the water jacket is limited to a local penetration or crushing, local repairs can be performed to the shell and the water replaced. If damage to the water jacket is extensive, the damage shall be repaired and re-tested in accordance with Chapter 9, following removal of the MPC.

If upon inspection of the damaged HI-TRAC transfer cask and MPC, damage of the MPC is observed, the loaded HI-TRAC transfer cask will be returned to the facility for fuel unloading in accordance with Chapter 8. The handling accident will not affect the ability to unload the MPC using normal means as the structural analysis of the 60g loading (HI-STAR Docket Numbers 71-9261 and 72-1008) shows that there will be no gross deformation of the MPC basket. After unloading, the structural damage of the HI-TRAC and MPC shall be assessed and a determination shall be made if repairs will enable the equipment to return to service. Subsequent to the repairs, the equipment shall be inspected and appropriate tests shall be performed to certify the equipment for service. If the equipment cannot be repaired and returned to service, the equipment shall be disposed of in accordance with the appropriate regulations.

11.2.2 HI-STORM Overpack Handling Accident

11.2.2.1 Cause of HI-STORM Overpack Handling Accident

During the operation of the HI-STORM 100 System, the loaded HI-STORM overpack is lifted in the vertical orientation. The height of the loaded overpack above the ground shall be limited to below the vertical handling height limit determined in Chapter 3 ~~and specified by the Technical Specifications in Appendix A to the CoC~~. This vertical handling height limit will maintain the inertial loading on the cask in a vertical drop to 45g's or less. Although a handling accident is remote, a drop from the vertical handling height limit is a credible accident.

11.2.2.2 HI-STORM Overpack Handling Accident Analysis

The handling accident analysis evaluates the effects of dropping the loaded overpack in the vertical orientation. The analysis of the handling accident is provided in Chapter 3. The analysis shows that the HI-STORM 100 System meets all structural requirements and there are no adverse effects on the structural, confinement, thermal or subcriticality performance of the HI-STORM 100 System. Limiting the inertia loading to 60g's or less ensures the fuel cladding remains intact based on dynamic impact effects on spent fuel assemblies in the literature [11.2.1].

Structural

The structural evaluation of the MPC under a 60g vertical load is presented in the HI-STAR TSAR and SAR [11.2.6 and 11.2.7] and it is demonstrated therein that the stresses are within allowable limits. The structural analysis of the HI-STORM overpack is presented in Section 3.4. The structural analysis of the overpack shows that the concrete shield attached to the underside of the overpack lid remains attached and air inlet ducts do not collapse.

Thermal

As the structural analysis demonstrates that there is no change in the MPC or overpack, there is no effect on the thermal performance of the system as a result of this event.

Shielding

As the structural analysis demonstrates that there is no change in the MPC or overpack, there is no effect on the shielding performance of the system as a result of this event.

Criticality

There is no effect on the criticality control features of the system as a result of this event.

Confinement

There is no effect on the confinement function of the MPC as a result of this event. As discussed in the structural evaluation above, all stresses remain within allowable values, assuring confinement boundary integrity.

Radiation Protection

Since there is no degradation in shielding or confinement capabilities as discussed above, there is no effect on occupational or public exposures as a result of this event.

Based on this evaluation, it is concluded that the vertical drop of the HI-STORM Overpack with the MPC inside does not affect the safe operation of the HI-STORM 100 System.

11.2.2.3 HI-STORM Overpack Handling Accident Dose Calculations

The vertical drop handling accident of the loaded HI-STORM overpack will not cause any change of the shielding or breach of the MPC confinement boundary. Any possible rupture of the fuel cladding will have no affect on the site boundary dose rates because the magnitude of the radiation source has not changed. Therefore, the dose calculations are equivalent to the normal condition dose rates.

11.2.2.4 HI-STORM Overpack Handling Accident Corrective Action

Following a handling accident, the ISFSI operator shall first perform a radiological and visual inspection to determine the extent of the damage to the overpack. Special handling procedures, as required, will be developed and approved by the ISFSI operator.

If upon inspection of the MPC, structural damage of the MPC is observed, the MPC is to be returned to the facility for fuel unloading in accordance with Chapter 8. After unloading, the structural damage of the MPC shall be assessed and a determination shall be made if repairs will enable the MPC to return to service. Likewise, the HI-STORM overpack shall be thoroughly inspected and a determination shall be made if repairs will enable the HI-STORM overpack to return to service. Subsequent to the repairs, the equipment shall be inspected and appropriate tests shall be performed to certify the HI-STORM 100 System for service. If the equipment cannot be repaired and returned to service, the equipment shall be disposed of in accordance with the appropriate regulations.

11.2.3 Tip-Over

11.2.3.1 Cause of Tip-Over

The analysis of the HI-STORM 100 System has shown that the overpack does not tip over as a result of the accidents (i.e., tornado missiles, flood water velocity, and seismic activity) analyzed in this section. It is highly unlikely that the overpack will tip-over during on-site movement because of the low handling height limit. The tip-over accident is stipulated as a non-mechanistic accident.

For the anchored HI-STORM designs (HI-STORM 100A and 100SA), a tip-over accident is not possible. As described in Chapter 2 of this FSAR, these system designs are not evaluated for the hypothetical tip-over. As such, the remainder of this accident discussion applies only to the non-anchored designs (i.e., the 100 and 100S designs only).

11.2.3.2 Tip-Over Analysis

The tip-over accident analysis evaluates the effects of the loaded overpack tipping-over onto a reinforced concrete pad. The tip-over analysis is provided in Section 3.4. The structural analysis

provided in Appendix 3.A demonstrates that the resultant deceleration loading on the MPC as a result of the tip-over accident is less than the design basis 45g's. The analysis shows that the HI-STORM 100 System meets all structural requirements and there is no adverse effect on the structural, confinement, thermal, or subcriticality performance of the MPC. However, the side impact will cause some localized damage to the concrete and outer shell of the overpack in the radial area of impact.

Structural

The structural evaluation of the MPC presented in Section 3.4 demonstrates that under a 45g loading the stresses are well within the allowable values. Analysis presented in Chapter 3 shows that the concrete shields attached to the underside and top of the overpack lid remains attached. As a result of the tip-over accident there will be localized crushing of the concrete in the area of impact.

Thermal

The thermal analysis of the overpack and MPC is based on vertical storage. The thermal consequences of this accident while the overpack is in the horizontal orientation are bounded by the burial under debris accident evaluated in Subsection 11.2.14. Damage to the overpack will be limited as discussed above. As the structural analysis demonstrates that there is no significant change in the MPC or overpack, once the overpack and MPC are returned to their vertical orientation there is no effect on the thermal performance of the system.

Shielding

The effect on the shielding performance of the system as a result of this event is limited to a localized decrease in the shielding thickness of the concrete.

Criticality

There is no effect on the criticality control features of the system as a result of this event.

Confinement

There is no effect on the confinement function of the MPC as a result of this event. As discussed in the structural evaluation above, all stresses remain within allowable values, assuring confinement boundary integrity.

Radiation Protection

Since there is a very localized reduction in shielding and no effect on the confinement capabilities as discussed above, there is no effect on occupational or public exposures as a result of this accident event.

Based on this evaluation, it is concluded that the accident pressure does not affect the safe operation of the HI-STORM 100 System.

11.2.3.3 Tip-Over Dose Calculations

The tip-over accident could cause localized damage to the radial concrete shield and outer steel shell where the overpack impacts the surface. The overpack surface dose rate in the affected area could increase due to the damage. However, there should be no noticeable increase in the ISFSI site or boundary dose rate, because the affected areas will be small and localized. The analysis of the tip-over accident has shown that the MPC confinement barrier will not be compromised and, therefore, there will be no release of radioactivity or increase in site-boundary dose rates.

11.2.3.4 Tip-Over Accident Corrective Action

Following a tip-over accident, the ISFSI operator shall first perform a radiological and visual inspection to determine the extent of the damage to the overpack. Special handling procedures will be developed and approved by the ISFSI operator.

If upon inspection of the MPC, structural damage of the MPC is observed, the MPC shall be returned to the facility for fuel unloading in accordance with Chapter 8. After unloading, the structural damage of the MPC shall be assessed and a determination shall be made if repairs will enable the MPC to return to service. Likewise, the HI-STORM overpack shall be thoroughly inspected and a determination shall be made if repairs are required and will enable the HI-STORM overpack to return to service. Subsequent to the repairs, the equipment shall be inspected and appropriate tests shall be performed to certify the HI-STORM 100 System for service. If the equipment cannot be repaired and returned to service, the equipment shall be disposed of in accordance with the appropriate regulations.

11.2.4 Fire Accident

11.2.4.1 Cause of Fire

Although the probability of a fire accident affecting a HI-STORM 100 System during storage operations is low due to the lack of combustible materials at the ISFSI, a conservative fire has been assumed and analyzed. The analysis shows that the HI-STORM 100 System continues to perform its structural, confinement, thermal, and subcriticality functions.

11.2.4.2 Fire Analysis

11.2.4.2.1 Fire Analysis for HI-STORM Overpack

The possibility of a fire accident near an ISFSI is considered to be extremely remote due to an absence of combustible materials within the ISFSI and adjacent to the overpacks. The only credible concern is related to a transport vehicle fuel tank fire, causing the outer layers of the storage overpack to be heated by the incident thermal radiation and forced convection heat fluxes. The amount of combustible fuel in the on-site transporter is limited to a volume of 50 gallons ~~based on a Technical Specification in Appendix A to the CoC~~.

With respect to fire accident thermal analysis, NUREG-1536 (4.0,V,5.b) states:

> "Fire parameters included in 10 CFR 71.73 have been accepted for characterizing the heat transfer during the in-storage fire. However, a bounding analysis that limits the fuel source thus limits the length of the fire (e.g., by limiting the source of the fuel in the transporter) has also been accepted."

Based on this NUREG-1536 guidance, the fire accident thermal analysis is performed using the 10 CFR 71.73 parameters and the fire duration is determined from the limited fuel volume of 50 gallons. The entire transient evaluation of the storage fire accident consists of three parts: (1) a bounding steady-state initial condition, (2) the short-duration fire event, and (3) the post-fire temperature relaxation period.

As stated above, the fire parameters from 10 CFR 71.73 are applied to the HI-STORM fire accident evaluation. 10 CFR 71 requirements for thermal evaluation of hypothetical accident conditions specifically define pre- and post-fire ambient conditions, specifically:

> "the ambient air temperature before and after the test must remain constant at that value between -29°C (-20°F) and +38°C (100°F) which is most unfavorable for the feature under consideration."

The ambient air temperature is therefore set to 100°F both before (bounding steady state) and after (post-fire temperature relaxation period) the short-duration fire event.

During the short-duration fire event, the following parameters from 10CFR71.71(c)(4), *also from Reference [11.2.3]*, are applied:

1. Except for a simple support system, the cask must be fully engulfed. The ISFSI pad is a simple support system, so the fire environment is not applied to the overpack baseplate. By fully engulfing the overpack, additional heat transfer surface area is conservatively exposed to the elevated fire temperatures.

2. The average emissivity coefficient must be at least 0.9. During the entire duration of the fire, the painted outer surfaces of the overpack are assumed to remain intact, with an emissivity of 0.85. It is conservative to assume that the flame emissivity is 1.0, the limiting maximum value corresponding to a perfect blackbody emitter. With a flame emissivity conservatively assumed to be 1.0 and a painted surface emissivity of 0.85, the effective emissivity coefficient is 0.85. Because the minimum required value of 0.9 is greater than the actual value of 0.85, use of an average emissivity coefficient of 0.9 is conservative.

3. The average flame temperature must be at least 800°C (1475°F). Open pool fires typically involve the entrainment of large amounts of air, resulting in lower average flame temperatures. Additionally, the same temperature is applied to all exposed cask surfaces, which is very conservative considering the size of the HI-STORM cask. It is therefore conservative to use the 1475°F temperature.

4. The fuel source must extend horizontally at least 1 m (40 in), but may not extend more than 3 m (10 ft), beyond the external surface of the cask. Use of the minimum ring width of 1 meter yields a deeper pool for a fixed quantity of combustible fuel, thereby conservatively maximizing the fire duration.

5. The convection coefficient must be that value which may be demonstrated to exist if the cask were exposed to the fire specified. Based upon results of large pool fire thermal measurements [11.2.2], a conservative forced convection heat transfer coefficient of 4.5 Btu/(hr×ft^2×°F) is applied to exposed overpack surfaces during the short-duration fire.

Due to the severity of the fire condition radiative heat flux, heat flux from incident solar radiation is negligible and is not included. Furthermore, the smoke plume from the fire would block most of the solar radiation.

Based on the 50 gallon fuel volume, the overpack outer diameter and the 1 m fuel ring width, the fuel ring surrounding the overpack covers 147.6 ft^2 and has a depth of 0.54 in. From this depth and a linear fuel consumption rate of 0.15 in/min, the fire duration is calculated to be 3.622 minutes (217 seconds). The linear fuel consumption rate of 0.15 in/min is the smallest value given in a Sandia Report on large pool fire thermal testing [11.2.2]. Use of the minimum linear consumption rate conservatively maximizes the duration of the fire.

It is recognized that the ventilation air in contact with the inner surface of the HI-STORM overpack with design-basis decay heat under maximum normal ambient temperature conditions varies between 80°F at the bottom and ~~206~~*243*°F at the top of the overpack. It is further recognized that the inlet and outlet ducts occupy only 1.25% of area of the cylindrical surface of the massive HI-STORM overpack. Due to the short duration of the fire event and the relative isolation of the ventilation passages from the outside environment, the ventilation air is expected to experience little intrusion of the fire combustion products. As a result of these considerations, it is conservative to assume that the

air in the HI-STORM overpack ventilation passages is held constant at a substantially elevated temperature of 300°F during the entire duration of the fire event.

The thermal transient response of the storage overpack is determined using the ANSYS finite element program. Time-histories for points in the storage overpack are monitored for the duration of the fire and the subsequent post-fire equilibrium phase.

Heat input to the HI-STORM overpack while it is subjected to the fire is from a combination of an incident radiation and convective heat fluxes to all external surfaces. This can be expressed by the following equation:

$$q_F = h_{fc}(T_A - T_S) + 0.1714 \times 10^8 \varepsilon [(T_A + 460)^4 - (T_S + 460)^4]$$

where:

q_F =Surface Heat Input Flux (Btu/ft^2-hr)
h_{fc} = Forced Convection Heat Transfer Coefficient (4.5 Btu/ft^2-hr-°F)
T_A = Fire Condition Temperature (1475°F)
T_S = Transient Surface Temperature (°F)
ε = Average Emissivity (0.90 per 10 CFR 71.73)

The forced convection heat transfer coefficient is based on the results of large pool fire thermal measurements [11.2.2].

After the fire event, the ambient temperature is restored to 100°F and the storage overpack cools down (post-fire temperature relaxation). Heat loss from the outer surfaces of the storage overpack is determined by the following equation:

$$q_S = h_S(T_S - T_A) + 0.1714 \times 10^8 \varepsilon [(T_S + 460)^4 - (T_A + 460)^4]$$

where:

q_S =Surface Heat Loss Flux (Btu/ft^2-hr)
h_S = Natural Convection Heat Transfer Coefficient (Btu/ft^2-hr-°F)
T_S = Transient Surface Temperature (°F)
T_A = Ambient Temperature (°F)
ε = Surface Emissivity

In the post-fire temperature relaxation phase, the surface heat transfer coefficient (h_S) is determined by the following equation:

$$h_S = 0.19 \times (T_A - T_S)^{1/3}$$

where:

h_S = Natural Convection Heat Transfer Coefficient (Btu/ft^2-hr-°F)
T_A = External Air Temperature (°F)
T_S = Transient Surface Temperature (°F)

As discussed in Subsection 4.5.1.1.2, this equation is appropriate for turbulent natural convection from vertical surfaces. For the same conservative value of the Z parameter assumed earlier (2.6×10^5) and the HI-STORM overpack height of approximately 19 feet, the surface-to-ambient temperature difference required to ensure turbulence is 0.56 °F.

A two-dimensional, axisymmetric model was developed for this analysis. Material thermal properties used were taken from Section 4.2. An element plot of the 2-D axisymmetric ANSYS model is shown in Figure 11.2.1. The outer surface and top surface of the overpack are exposed to the ambient conditions (fire and post-fire), and the base of the overpack is insulated. The transient study is conducted for a period of 5 hours, which is sufficient to allow temperatures in the overpack to reach their maximum values and begin to recede.

Based on the results of the analysis, the maximum temperatures ~~increases~~ at several points near the overpack mid-height are summarized in Table 11.2.2 along with the corresponding peak temperatures *in the MPC*. ~~Temperature profiles through the storage overpack wall thickness near the mid-height of the cask are included in Figures 11.2.2 through 11.2.4. A plot of temperature versus time is shown in Figure 11.2.5 for several points through the overpack wall, near the mid-height of the cask. The temperature profile plots (Figures 11.2.2 through 11.2.4) each contain profiles corresponding to time "snapshots". Profiles are presented at the following times: 1 minute (60 seconds), 2 minutes (120 seconds), 3.622 minutes (217 seconds – end of fire), 10 minutes (600 seconds), 20 minutes (1200 seconds), 40 minutes and 90 minutes.~~

The primary shielding material in the storage overpack is concrete, which can suffer a reduction in neutron shielding capability at sustained high temperatures due to a loss of water. ~~As shown in Figure 11.2.5, l~~Less than 1 inch of the concrete near the outer overpack surface exceeds the material short-term temperature limit. This condition is addressed specifically in NUREG-1536 (4.0,V,5.b), which states:

> "The NRC accepts that concrete temperatures may exceed the temperature criteria of ACI 349 for accidents if the temperatures result from a fire."

These results demonstrate that the fire accident event does not substantially affect the HI-STORM overpack. Only localized regions of concrete are exposed to temperatures in excess of the allowable short-term temperature limit. No portions of the steel structure exceed the allowable temperature limits.

Having evaluated the effects of the fire on the overpack, we must now evaluate the effects on the MPC and contained fuel assemblies. Guidance for the evaluation of the MPC and its internals during a fire event is provided by NUREG-1536 (4.0,V,5.b), which states:

> "For a fire of very short duration (i.e., less than 10 percent of the thermal time constant of the cask body), the NRC finds it acceptable to calculate the fuel temperature increase by assuming that the cask inner wall is adiabatic. The fuel temperature increase should then be determined by dividing the decay energy released during the fire by the thermal capacity of the basket-fuel assembly combination."

The time constant of the cask body (i.e., the overpack) can be determined using the formula:

$$\tau = \frac{c_p \times \rho \times L_c^2}{k}$$

where:

c_p = Overpack Specific Heat Capacity (Btu/lb-°F)
ρ = Overpack Density (lb/ft^3)
L_c = Overpack Characteristic Length (ft)
k = Overpack Thermal Conductivity (Btu/ft-hr-°F)

The concrete contributes the majority of the overpack mass and volume, so we will use the specific heat capacity (0.156 Btu/lb-°F), density (142 lb/ft^3) and thermal conductivity (1.05 Btu/ft-hr-°F) of concrete for the time constant calculation. The characteristic length of a hollow cylinder is its wall thickness. The characteristic length for the HI-STORM overpack is therefore 29.5 in, or approximately 2.46 ft. Substituting into the equation, the overpack time constant is determined as:

$$\tau = \frac{0.156 \times 142 \times 2.46^2}{1.05} = 127.7 hrs$$

One-tenth of this time constant is approximately 12.8 hours (766 minutes), substantially longer than the fire duration of 3.622 minutes, so the MPC is evaluated by considering the MPC canister as an adiabatic boundary. The temperature of the MPC is therefore increased by the contained decay heat only.

Table 4.5.5 lists lower-bound thermal inertia values for the MPC and the contained fuel assemblies of 4680 Btu/°F and 2240 Btu/°F, respectively. Applying an upper-bound decay heat load of ~~28.74~~ *38* kW (~~98,090~~ *1.3×10^5* Btu/hr) for the 3.622 minute (0.0604 hours) fire duration results in the contained fuel assemblies heating up by only:

$$\Delta T_{fuel} = \frac{1.3 \times 10^5 \times 0.0604}{4680 + 2240} = 1.1°F$$

This is a negligible increase in the fuel temperature. Consequently, the impact on the MPC internal helium pressure will be negligible as well. Based on a conservative analysis of the HI-STORM 100 System response to a hypothetical fire event, it is concluded that the fire event does not significantly affect the temperature of the MPC or contained fuel. Furthermore, the ability of the HI-STORM 100 System to cool the spent nuclear fuel within design temperature limits during post-fire temperature relaxation is not compromised.

Structural

As discussed above, there are no structural consequences as a result of the fire accident condition.

Thermal

As discussed above, the MPC internal pressure increases a negligible amount and is bounded by the 100% fuel rod rupture accident in Section 11.2.9. As shown in Table 11.2.2, the peak fuel cladding and material temperatures are well below short-term accident condition allowable temperatures of Table 2.2.3.

Shielding

With respect to concrete damage from a fire, NUREG-1536 (4.0,V,5.b) states: "the loss of a small amount of shielding material is not expected to cause a storage system to exceed the regulatory requirements in 10 CFR 72.106 and, therefore, need not be estimated or evaluated in the SAR." Less than one-inch of the concrete (less than 4% of the total overpack radial concrete section) exceeds the short-term temperature limit.

Criticality

There is no effect on the criticality control features of the system as a result of this event.

Confinement

There is no effect on the confinement function of the MPC as a result of this event.

Radiation Protection

Since there is a very localized reduction in shielding and no effect on the confinement capabilities as discussed above, there is no effect on occupational or public exposures as a result of this accident event.

Based on this evaluation, it is concluded that the overpack fire accident does not affect the safe operation of the HI-STORM 100 System.

11.2.4.2.2 Fire Analysis for HI-TRAC Transfer Cask

To demonstrate the fuel cladding and MPC pressure boundary integrity under an exposure to a hypothetical short duration fire event during on-site handling operations, a fire accident analysis of the loaded 100-ton HI-TRAC is performed. This analysis, because of the lower mass of the 100-ton HI-TRAC, bounds the effects for the 125-ton HI-TRAC. In this analysis, the contents of the HI-TRAC are conservatively postulated to undergo a transient heat-up as a lumped mass from the decay heat input and heat input from the short duration fire. The rate of temperature rise of the HI-TRAC depends on the thermal inertia of the cask, the cask initial conditions, the spent nuclear fuel decay heat generation, and the fire heat flux. All of these parameters are conservatively bounded by the values in Table 11.2.3, which are used for the fire transient analysis.

Using the values stated in Table 11.2.3, a bounding cask temperature rise of ~~5.509~~*5.56*°F per minute is determined from the combined radiant and forced convection fire and decay heat inputs to the cask. During the handling of the HI-TRAC transfer cask, the transporter is limited to a maximum of 50 gallons~~, in accordance with a Technical Specification in Appendix A to the CoC~~. The duration of the 50-gallon fire is 4.775 minutes. Therefore, *the temperature rise computed as the product of the rate of temperature rise and the fire duration is 26.6~~36.8~~°F. Because the cladding temperature at the start of fire is substantially below the accident temperature limit (approximately 300°F lower)* the fuel cladding *temperature limit is not exceeded.* ~~will not exceed the short term fuel cladding temperature limit (see Table 11.2.5).~~

The elevated temperatures as a result of the fire accident will cause the pressure in the water jacket to increase and cause the overpressure relief valves to vent steam to the atmosphere. Based on the fire heat input to the water jacket, less than 11% of the water in the water jacket can be boiled off. However, it is conservatively assumed, for dose calculations, that all the water in the water jacket is lost. In the 125-ton HI-TRAC, which uses Holtite in the lids for neutron shielding, the elevated fire temperatures would cause the Holtite to exceed its design accident temperature limits. It is conservatively assumed, for dose calculations, that all the Holtite in the 125-ton HI-TRAC is lost.

Due to the increased temperatures the MPC experiences as a result of the fire accident in the HI-TRAC transfer cask, the MPC internal pressure increases. Table 11.2.4 provides the MPC maximum internal pressures, as a result of the HI-TRAC fire accident, *for a conservatively bounding initial steady state condition of the highest normal operating pressure and minimum cavity average temperature. The computed accident pressure is substantially below the accident design pressure (Table 2.2.1).* ~~The values presented in Table 11.2.4 are determined using a bounding temperature rise of 43.2°F, instead of the calculated 26.3°F temperature rise, and are therefore conservative.~~ Table 11.2.5 provides a summary of the loaded HI-TRAC bounding maximum temperatures for the hypothetical fire accident condition.

Structural

As discussed above, there are no structural consequences as a result of the fire accident condition.

Thermal

As discussed above, the MPC internal pressure *and fuel temperature* increases as a result of the fire accident. ~~, but the~~ *The fire accident MPC* internal pressure *and peak fuel cladding temperature*~~, conservatively including a non-mechanistic 100% fuel rod rupture, is shown in Table 11.2.4 to be~~ *are substantially* less than the ~~accident~~ *accident limits for MPC internal pressure and maximum cladding temperature (Tables 2.2.1 and 2.2.3).* ~~accident condition MPC internal design pressure of 200 psig (Table 2.2.1). As shown in Table 11.2.5, the peak fuel cladding and material temperatures are well below short-term accident condition allowable temperatures of Table 2.2.3.~~

The loss of the water in the water jacket causes the temperatures to increase ~~slightly~~ due to a reduction in the thermal conductivity through the HI-TRAC water jacket. The temperatures of the MPC in the HI-TRAC transfer cask as a result of the loss of water in the water jacket are presented in Table 11.2.8 based on an assumed start at normal on-site transport conditions *and assuming that a steady state is reached.* As can be seen from the values in the table, *the temperatures are below the accident temperature limits.* ~~the temperatures increase by less than 20°F. Therefore, if the temperatures presented in Table 11.2.5 were increased by 20°F to account for the decrease in conductivity of the water jacket, the resultant temperatures will still be well below the short-term allowable fuel cladding and material temperatures provided in Table 2.2.3 for accident conditions.~~

Shielding

The assumed loss of all the water in the water jacket results in an increase in the radiation dose rates at locations adjacent to the water jacket. The assumed loss of all the Holtite in the 125-ton HI-TRAC lids results in an increase in the radiation dose rates at locations adjacent to the lids. The shielding analysis results presented in Section 5.1.2 demonstrate that the requirements of 10CFR72.106 are not exceeded.

Criticality

There is no effect on the criticality control features of the system as a result of this event.

Confinement

There is no effect on the confinement function of the MPC as a result of this event, since the internal pressure does not exceed the accident condition design pressure and the MPC confinement boundary temperatures do not exceed the short-term allowable temperature limits.

Radiation Protection

There is no degradation in confinement capabilities of the MPC, as discussed above. There are increases in the local dose rates adjacent *to the* water jacket. HI-TRAC dose rates at 1 meter and 100 meters from the water jacket, after the water is lost, have already been reported in Subsection 11.2.1.2. Immediately after the fire accident a radiological inspection of the HI-TRAC will be performed and temporary shielding shall be installed to limit the exposure to the public.

11.2.4.3 Fire Dose Calculations

The complete loss of the HI-TRAC neutron shield along with the water jacket shell is assumed in the shielding analysis for the post-accident analysis of the loaded HI-TRAC in Chapter 5 and bounds the determined fire accident consequences. The loaded HI-TRAC following a fire accident meets the accident dose rate requirement of 10CFR72.106.

The elevated temperatures experienced by the HI-STORM overpack concrete shield is limited to the outermost layer. Therefore, any corresponding reduction in neutron shielding capabilities is limited to the outermost layer. The slight increase in the neutron dose rate as a result of the concrete in the outer inch reaching elevated temperatures will not significantly increase the site boundary dose rate, due to the limited amount of the concrete shielding with reduced effectiveness and the negligible neutron dose rate calculated for normal conditions at the site boundary. The loaded HI-STORM overpack following a fire accident meets the accident dose rate requirement of 10CFR72.106.

The analysis of the fire accident shows that the MPC confinement boundary is not compromised and therefore, there is no release of airborne radioactive materials.

11.2.4.4 Fire Accident Corrective Actions

Upon detection of a fire adjacent to a loaded HI-TRAC or HI-STORM overpack, the ISFSI operator shall take the appropriate immediate actions necessary to extinguish the fire. Fire fighting personnel should take appropriate radiological precautions, particularly with the HI-TRAC as the pressure relief valves may have opened and water loss from the water jacket may have occurred resulting in an increase in radiation doses. Following the termination of the fire, a visual and radiological inspection of the equipment shall be performed.

As appropriate, install temporary shielding around the HI-TRAC. Specific attention shall be taken during the inspection of the water jacket of the HI-TRAC. If damage to the HI-TRAC is limited to the loss of water in the water jacket due to the pressure increase, the water may be replaced by adding water at pressure. If damage to the HI-TRAC water jacket or HI-TRAC body is widespread and/or radiological conditions require, the HI-TRAC shall be unloaded in accordance with Chapter 8, prior to repair.

If damage to the HI-STORM storage overpack as the result of a fire event is widespread and/or as radiological conditions require, the MPC shall be removed from the HI-STORM overpack in accordance with Chapter 8. However, the thermal analysis described herein demonstrates that only the outermost layer of the radial concrete exceeds its design temperature. The HI-STORM overpack may be returned to service if there is no increase in the measured dose rates (i.e., the overpack's shielding effectiveness is confirmed) and if the visual inspection is satisfactory.

11.2.5 Partial Blockage of MPC Basket Vent Holes

Each MPC basket fuel cell wall has elongated vent holes at the bottom and top. The partial blockage of the MPC basket vent holes analyzes the effects on the HI-STORM 100 System due to the restriction of the vent openings.

11.2.5.1 Cause of Partial Blockage of MPC Basket Vent Holes

After the MPC is loaded with spent nuclear fuel, the MPC cavity is drained, vacuum dried, and backfilled with helium. There are only two possible sources of material that could block the MPC basket vent holes. These are the fuel cladding/fuel pellets and crud. Due to the maintenance of relatively low cladding temperatures during storage, it is not credible that the fuel cladding would rupture, and that fuel cladding and fuel pellets would fall to block the basket vent holes. It is conceivable that a percentage of the crud deposited on the fuel rods may fall off of the fuel assembly and deposit at the bottom of the MPC.

Helium in the MPC cavity provides an inert atmosphere for storage of the fuel. The HI-STORM 100 System maintains the peak fuel cladding temperature below the required long-term storage limits. All credible accidents do not cause the fuel assembly to experience an inertia loading greater than 60g's. Therefore, there is no mechanism for the extensive rupture of spent fuel rod cladding.

Crud can be made up of two types of layers, loosely adherent and tightly adherent. The SNF assembly movement from the fuel racks to the MPC may cause a portion of the loosely adherent crud to fall away. The tightly adherent crud is not removed during ordinary fuel handling operations. The MPC vent holes that act as the bottom plenum for the MPC internal thermosiphon are of an elongated, semi-circular design to ensure that the flow passages will remain open under a hypothetical shedding of the crud on the fuel rods. For conservatism, only the minimum semi-circular hole area is credited in the thermal models (i.e., the elongated portion of the hole is completely neglected).

The amount of crud on fuel assemblies varies greatly from plant to plant. Typically, BWR plants have more crud than PWR plants. Based on the maximum expected crud volume per fuel assembly provided in reference [11.2.5], and the area at the base of the MPC basket fuel storage cell, the maximum depth of crud at the bottom of the MPC-68 was determined. For the PWR-style MPC designs (see Table 1.2.1), 90% of the maximum crud volume was used to determine the crud depth. The maximum crud depths calculated for each of the MPCs is listed in Table 2.2.8. The maximum

amount of crud was assumed to be present on all fuel assemblies within the MPC. Both the tightly and loosely adherent crud was conservatively assumed to fall off of the fuel assembly. As can be seen by the values listed in the table, the maximum amount of crud depth does not totally block any of the MPC basket vent holes as the crud accumulation depth is less than the elongation of the vent holes. Therefore, the available vent holes area is greater than that used in the thermal models.

11.2.5.2 Partial Blockage of MPC Basket Vent Hole Analysis

The partial blockage of the MPC basket vent holes has no affect on the structural, confinement and thermal analysis of the MPC. There is no affect on the shielding analysis other than a slight increase of the gamma radiation dose rate at the base of the MPC due to the accumulation of crud. As the MPC basket vent holes are not completely blocked, preferential flooding of the MPC fuel basket is not possible, and, therefore, the criticality analyses are not affected.

Structural

There are no structural consequences as a result of this event.

Thermal

There is no effect on the thermal performance of the system as a result of this event.

Shielding

There is no effect on the shielding performance of the system as a result of this accident event.

Criticality

There is no effect on the criticality control features of the system as a result of this accident event.

Confinement

There is no effect on the confinement function of the MPC as a result of this accident event.

Radiation Protection

Since there is no degradation in shielding or confinement capabilities as discussed above, there is no effect on occupational or public exposures as a result of this accident event.

Based on this evaluation, it is concluded that the partial blockage of MPC vent holes does not affect the safe operation of the HI-STORM 100 System.

11.2.5.3 Partial Blockage of MPC Basket Vent Holes Dose Calculations

Partial blockage of basket vent holes will not result in a compromise of the confinement boundary. Therefore, there will be no effect on the site boundary dose rates because the magnitude of the radiation source has not changed. There will be no radioactive material release.

11.2.5.4 Partial Blockage of MPC Basket Vent Holes Corrective Action

There are no consequences that exceed normal storage conditions. No corrective action is required for the partial blockage of the MPC basket vent holes.

11.2.6 Tornado

11.2.6.1 Cause of Tornado

The HI-STORM 100 System will be stored on an unsheltered ISFSI concrete pad and subject to environmental conditions. Additionally, the transfer of the MPC from the HI-TRAC transfer cask to the overpack may be performed at the unsheltered ISFSI concrete pad. It is possible that the HI-STORM System (storage overpack and HI-TRAC transfer cask) may experience the extreme environmental conditions of a tornado.

11.2.6.2 Tornado Analysis

The tornado accident has two effects on the HI-STORM 100 System. The tornado winds and/or tornado missile attempt to tip-over the loaded overpack or HI-TRAC transfer cask. The pressure loading of the high velocity winds and/or the impact of the large tornado missiles act to apply an overturning moment. The second effect is tornado missiles propelled by high velocity winds which attempt to penetrate the storage overpack or HI-TRAC transfer cask.

During handling operations at the ISFSI pad, the loaded HI-TRAC transfer cask, while in the vertical orientation, shall be attached to a lifting device designed in accordance with the requirements specified in Subsection 2.3.3.1. Therefore, it is not credible that the tornado missile and/or wind could tip-over the loaded HI-TRAC while being handled in the vertical orientation. During handling of the loaded HI-TRAC in the horizontal orientation, it is possible that the tornado missile and/or wind may cause the rollover of the loaded HI-TRAC on the transport vehicle. The horizontal drop handling accident for the loaded HI-TRAC, Subsection 11.2.1, evaluates the consequences of the loaded HI-TRAC falling from the horizontal handling height limit and consequently this bounds the effect of the roll-over of the loaded HI-TRAC on the transport vehicle.

Structural

Section 3.4 provides the analysis of the pressure loading which attempts to tip-over the storage overpack and the analysis of the effects of the different types of tornado missiles. These analyses

show that the loaded storage overpack does not tip-over as a result of the tornado winds and/or tornado missiles.

Analyses provided in Section 3.4 also shows that the tornado missiles do not penetrate the storage overpack or HI-TRAC transfer cask to impact the MPC. The result of the tornado missile impact on the storage overpack or HI-TRAC transfer cask is limited to damage of the shielding.

Thermal

The loss of the water in the water jacket causes the temperatures to increase slightly due to a reduction in the thermal conductivity through the HI-TRAC water jacket. The temperatures of the MPC in the HI-TRAC transfer cask as a result of the loss of water in the water jacket are presented in Table 11.2.8. As can be seen from the values in the table, the temperatures are well below the short-term allowable fuel cladding and material temperatures provided in Table 2.2.3 for accident conditions.

Shielding

The loss of the water in the water jacket results in an increase in the radiation dose rates at locations adjacent to the water jacket. The shielding analysis results presented in Section 5.1.2 demonstrate that the requirements of 10CFR72.106 are not exceeded.

Criticality

There is no effect on the criticality control features of the system as a result of this event.

Confinement

There is no effect on the confinement function of the MPC as a result of this event.

Radiation Protection

There is no degradation in confinement capabilities of the MPC, since the tornado missiles do not impact the MPC, as discussed above. There are increases in the local dose rates adjacent water jacket as a result of the loss of water in the HI-TRAC water jacket. HI-TRAC dose rates at 1 meter and 100 meters from the water jacket, after the water is lost, have already been *discussed*~~reported~~ in Subsection 11.2.1.2. Immediately after the tornado accident a radiological inspection of the HI-TRAC will be performed and temporary shielding shall be installed to limit the exposure to the public.

11.2.6.3 Tornado Dose Calculations

The tornado winds do not tip-over the loaded storage overpack; damage the shielding materials of the overpack or HI-TRAC; or damage the MPC confinement boundary. There is no affect on the radiation dose as a result of the tornado winds. A tornado missile may cause localized damage in the concrete radial shielding of the storage overpack. However, the damage will have a negligible effect on the site boundary dose. A tornado missile may penetrate the HI-TRAC water jacket shell causing the loss of the neutron shielding (water). The effects of the tornado missile damage on the loaded HI-TRAC transfer cask is bounded by the post-accident dose assessment performed in Chapter 5, which conservatively assumes complete loss of the water in the water jacket and the water jacket shell.

11.2.6.4 Tornado Accident Corrective Action

Following exposure of the HI-STORM 100 System to a tornado, the ISFSI operator shall perform a visual and radiological inspection of the overpack and/or HI-TRAC transfer cask. Damage sustained by the overpack outer shell, concrete, or vent screens shall be inspected and repaired. Damage sustained by the HI-TRAC shall be inspected and repaired.

11.2.7 Flood

11.2.7.1 Cause of Flood

The HI-STORM 100 System will be located on an unsheltered ISFSI concrete pad. Therefore, it is possible for the storage area to be flooded. The potential sources for the flood water could be unusually high water from a river or stream, a dam break, a seismic event, or a hurricane.

11.2.7.2 Flood Analysis

The flood accident affects the HI-STORM 100 overpack structural analysis in two ways. The flood water velocity acts to apply an overturning moment, which attempts to tip-over the loaded overpack. The flood affects the MPC by applying an external pressure.

Structural

Section 3.4 provides the analysis of the flood water applying an overturning moment. The results of the analysis show that the loaded overpack does not tip over if the flood velocity does not exceed the value stated in Table 2.2.8.

The structural evaluation of the MPC for the accident condition external pressure (Table 2.2.1) is presented in Section 3.4 and the resulting stresses from this event are shown to be well within the allowable values.

Thermal

For a flood of sufficient magnitude to allow the water to come into contact with the MPC, there is no adverse effect on the thermal performance of the system. The thermal consequence of such a flood is an increase in the rejection of the decay heat. Because the storage overpack is ventilated, water from a large flood will enter the annulus between the MPC and the overpack. The water would actually provide cooling that exceeds that available in the air filled annulus, due to water's higher thermal conductivity, density and heat capacity, and the forced convection coefficient associated with flowing water. Since the flood water temperature will be within the off-normal temperature range specified in Table 2.2.2, the thermal transient associated with the initial contact of the floodwater will be bounded by the off-normal operation conditions.

For a smaller flood that blocks the air inlet ducts but is not sufficient to allow water to come into contact with the MPC, a thermal analysis is included in Subsection 11.2.13 of this FSAR.

Shielding

There is no effect on the shielding performance of the system as a result of this event. The flood water acts as a radiation shield and will reduce the radiation doses.

Criticality

There is no effect on the criticality control features of the system as a result of this event. The criticality analysis is unaffected because under the flooding condition water does not enter the MPC cavity and therefore the reactivity would be less than the loading condition in the fuel pool which is presented in Section 6.1.

Confinement

There is no effect on the confinement function of the MPC as a result of this event. As discussed in the structural evaluation above, all stresses remain within allowable values, assuring confinement boundary integrity.

Radiation Protection

Since there is no degradation in shielding or confinement capabilities as discussed above, there is no effect on occupational or public exposures as a result of this event.

Based on this evaluation, it is concluded that the flood accident does not affect the safe operation of the HI-STORM 100 System.

11.2.7.3 Flood Dose Calculations

Since the flood accident produces no leakage of radioactive material and no reduction in shielding effectiveness, there are no adverse radiological consequences.

11.2.7.4 Flood Accident Corrective Action

As shown in the analysis of the flood accident, the HI-STORM 100 System sustains no damage as a result of the flood. At the completion of the flood, *surfaces wetted by floodwater shall be cleared of debris and cleaned of adherent foreign matter.* ~~the exterior and interior of the overpack, and the exterior of the MPC shall be cleaned to maintain the proper air flow and emissivity.~~

11.2.8 Earthquake

11.2.8.1 Cause of Earthquake

The HI-STORM 100 System may be employed at any reactor or ISFSI facility in the United States. It is possible that during the use of the HI-STORM 100 System, the ISFSI may experience an earthquake.

11.2.8.2 Earthquake Analysis

The earthquake accident analysis evaluates the effects of a seismic event on the loaded HI-STORM 100 System. The objective is to determine the stability limits of the HI-STORM 100 System. Based on a static stability criteria, it is shown in Chapter 3 that the HI-STORM 100 System is qualified to seismic activity less than or equal to the values specified in Table 2.2.8. The analyses in Chapter 3 show that the HI-STORM 100 System will not tip over under the conditions evaluated. The seismic activity has no adverse thermal, criticality, confinement, or shielding consequences.

Some ISFSI sites will have earthquakes that exceed the seismic activity specified in Table 2.2.8. For these high-seismic sites, anchored HI-STORM designs (the HI-STORM 100A and 100SA) have been developed. The design of these anchored systems is such that seismic loads cannot result in tip-over or lateral displacement. Chapter 3 provides a detailed discussion of the anchored systems design.

Structural

The sole structural effect of the earthquake is an inertial loading of less than 1g. This loading is bounded by the tip-over analysis presented in Section 11.2.3, which analyzes a deceleration of 45g's and demonstrates that the MPC allowable stress criteria are met.

Thermal

There is no effect on the thermal performance of the system as a result of this event.

Shielding

There is no effect on the shielding performance of the system as a result of this event.

Criticality

There is no effect on the criticality control features of the system as a result of this event.

Confinement

There is no effect on the confinement function of the MPC as a result of this event.

Radiation Protection

Since there is no degradation in shielding or confinement capabilities as discussed above, there is no effect on occupational or public exposures as a result of this event.

Based on this evaluation, it is concluded that the earthquake does not affect the safe operation of the HI-STORM 100 System.

11.2.8.3 Earthquake Dose Calculations

Structural analysis of the earthquake accident shows that the loaded overpack will not tip over as a result of the specified seismic activity. If the overpack were to tip over, the resultant damage would be equal to that experienced by the tip-over accident analyzed in Subsection 11.2.3. Since the loaded overpack does not tip-over, there is no increase in radiation dose rates or release of radioactivity.

11.2.8.4 Earthquake Accident Corrective Action

Following the earthquake accident, the ISFSI operator shall perform a visual and radiological inspection of the overpacks in storage to determine if any of the overpacks have tipped-over. In the unlikely event of a tip-over, the corrective actions shall be in accordance with Subsection 11.2.3.4.

11.2.9 100% Fuel Rod Rupture

This accident event postulates that all the fuel rods rupture and that the appropriate quantities of fission product gases and fill gas are released from the fuel rods into the MPC cavity.

11.2.9.1 Cause of 100% Fuel Rod Rupture

Through all credible accident conditions, the HI-STORM 100 System maintains the spent nuclear fuel in an inert environment while maintaining the peak fuel cladding temperature below the required

short-term temperature limits, thereby providing assurance of fuel cladding integrity. There is no credible cause for 100% fuel rod rupture. This accident is postulated to evaluate the MPC confinement barrier for the maximum possible internal pressure based on the non-mechanistic failure of 100% of the fuel rods.

11.2.9.2 100% Fuel Rod Rupture Analysis

The 100% fuel rod rupture accident has no thermal, structural, criticality or shielding consequences. The event does not change the reactivity of the stored fuel, the magnitude of the radiation source which is being shielded, the shielding capability, or the criticality control features of the HI-STORM 100 System. The determination of the maximum accident pressure is provided in Chapter 4. The MPC design basis internal pressure bounds the pressure developed assuming 100% fuel rod rupture. The structural analysis provided in Chapter 3 evaluates the MPC confinement boundary under the accident condition internal pressure.

Structural

The structural evaluation of the MPC for the accident condition internal pressure presented in Section 3.4 demonstrates that the MPC stresses are well within the allowable values.

Thermal

The MPC internal pressure for the 100% fuel rod rupture condition is presented in Table 4.4.14. As can be seen from the values, the ~~200 psig~~ design basis accident condition MPC internal pressure *(Table 2.2.1)* used in the structural evaluation bounds the calculated value.

Shielding

There is no effect on the shielding performance of the system as a result of this event.

Criticality

There is no effect on the criticality control features of the system as a result of this event.

Confinement

There is no effect on the confinement function of the MPC as a result of this event. As discussed in the structural evaluation above, all stresses remain within allowable values, assuring confinement boundary integrity.

Radiation Protection

Since there is no degradation in shielding or confinement capabilities as discussed above, there is no effect on occupational or public exposures as a result of this event.

Based on this evaluation, it is concluded that the non-mechanistic 100% fuel rod rupture accident does not affect the safe operation of the HI-STORM 100 System.

11.2.9.3 100% Fuel Rod Rupture Dose Calculations

The MPC confinement boundary maintains its integrity. There is no effect on the shielding effectiveness, and the magnitude of the radiation source is unchanged. However, the radiation source could redistribute within the sealed MPC cavity causing a slight change in the radiation dose rates at certain locations. Therefore, there is no release of radioactive material or significant increase in radiation dose rates.

11.2.9.4 100% Fuel Rod Rupture Accident Corrective Action

As shown in the analysis of the 100% fuel rod rupture accident, the MPC confinement boundary is not damaged. The HI-STORM 100 System is designed to withstand this accident and continue performing the safe storage of spent nuclear fuel under normal storage conditions. No corrective actions are required.

11.2.10 Confinement Boundary Leakage

The MPC uses redundant confinement closures to assure that there is no release of radioactive materials for postulated storage accident conditions. The analyses presented in Chapter 3 and this chapter demonstrate that the MPC remains intact during all postulated accident conditions. The discussion contained in Chapter 7 demonstrates that MPC is designed, welded, tested and inspected to meet the guidance of ISG-18 such that leakage from the confinement boundary is considered non-credible. ~~The confinement boundary leakage accident assumes simultaneous rupture of 100% of the fuel rods and the release of the available radioactive gas inventory to the environment at a rate based on 150% of the maximum leak rate under reference conditions.~~

11.2.10.1 Cause of Confinement Boundary Leakage

There is no credible cause for confinement boundary leakage. The accidents analyzed in this chapter show that the MPC confinement boundary withstands all credible accidents. There are no man-made or natural phenomena that could cause failure of the confinement boundary restricting radioactive material release. *Additionally, because the MPC satisfies the criteria specified in Interim Staff Guidance (ISG) 18, there is no credible leakage that would occur from the confinement boundary.* ~~The release is analyzed to demonstrate the safety of the HI-STORM 100 System~~

~~11.2.10.2 Confinement Boundary Leakage Analysis~~

~~The following is the basis for the conservative analysis of the confinement boundary leakage accident.~~

1. ~~All the fuel stored in the MPC has been cooled for 5 years. The PWR fuel type is the B&W 15×15 at 4.8% *5.0%* enrichment with a burnup of 70,000 *75,000* MWD/MTU. The BWR fuel type is the GE 7×7 at 4.84% enrichment with a burnup of 60,000 *70,000* MWD/MTU. These fuel characteristics bound the design basis fuel for the HI-STORM 100 System.~~

2. ~~One hundred percent of all the fuel rods are assumed to rupture.~~

3. ~~The releasable source term and release fractions are in accordance with NUREG-1536, ISG-5 and ISG-11.~~

~~The maximum possible leakage rate of radionuclides to the environment is based on the helium leak rate under reference test conditions from the Technical Specification in Appendix A to the CoC.~~

~~*Credit is taken for the gravitational settling of fines, volatiles and cruds.*~~

~~Chapter 7 presents an evaluation of the consequences of a non-mechanistic postulated ground-level breach of the MPC confinement boundary under hypothetical accident conditions of storage. The resulting Total Effective Dose Equivalent (TEDE) and other dose equivalents at a downstream distance of 100 meters are evaluated for each MPC type.~~

~~Structural~~

~~There are no structural consequences of the loss of confinement accident.~~

~~Thermal~~

~~Since this event is a non-mechanistic assumption, there are no realistic thermal consequences. As discussed in the Technical Specifications in Appendix A to the CoC, the leak test rate would result in a negligible loss of helium fill gas over the design life of the MPC, which would have an inconsequential effect on thermal performance.~~

~~Shielding~~

~~There is no effect on the shielding performance of the system as a result of this event.~~

~~Criticality~~

~~There is no effect on the criticality control features of the system as a result of this event.~~

~~Confinement~~

~~This event is based upon an assumed instantaneous breach of the confinement.~~

~~Radiation Protection~~

~~The postulated release will result in an increase in dose to the public. The analysis of this event is provided in Section 7.3. As shown therein, the postulated breach results in dose rates to the public less than the limit established by 10CFR72.106(b) for the site boundary.~~

~~11.2.10.3 Confinement Boundary Leakage Dose Calculations~~

~~10CFR72.106 requires that any individual located at or beyond the nearest controlled area boundary must not receive a dose greater than 5 Rem to the whole body or any organ from any design basis accident. The maximum whole body dose contribution as a result of the instantaneous leak accident is calculated in Chapter 7 (Table 7.3.8). The maximum doses as a result of the confinement boundary leak accident is calculated in Chapter 7 (Table 7.3.8). Both values are well below the regulatory limit of 5 Rem.~~

11.2.10.34 Confinement Boundary Leakage Accident Corrective Action

The HI-STORM 100 System is designed to withstand this accident and continue performing the safe storage of spent nuclear fuel. No corrective actions are required.

~~A detected breached MPC will need to be repaired or the fuel removed and placed into a new MPC. First, the breached MPC must be returned to the facility in accordance with the procedures provided in Chapter 8. If the leak can be detected and repaired, and testing can be performed to verify the integrity of the confinement boundary, the MPC may be placed back into service. Otherwise, the MPC should be unloaded in accordance with the procedures provided in Chapter 8.~~

11.2.11 Explosion

11.2.11.1 Cause of Explosion

An explosion within the bounds of an ISFSI is improbable since there are no explosive materials within the site boundary. An explosion as a result of combustion of the fuel contained in cask transport vehicle is possible. The fuel available for the explosion would be limited and therefore, any explosion would be limited in size. Any explosion stipulated to occur beyond the site boundary would have a minimal effect on the HI-STORM 100 System.

11.2.11.2 Explosion Analysis

Any credible explosion accident is bounded by the accident external pressure of 60 psig (Table 2.2.1) analyzed as a result of the flood accident water depth in Subsection 11.2.7 and the tornado missile accident of Subsection 11.2.6, because explosive materials will not be stored within close proximity to the casks. The HI-STORM Overpack does not experience the 60 psi external pressure since it is not a sealed vessel. However, a pressure differential of 10.0 psi (Table 2.2.1) is applied to the overpack. Section 3.4 provides the analysis of the accident external pressure on the MPC and overpack. The analysis shows that the MPC can withstand the effects of the accident condition external pressure, while conservatively neglecting the MPC internal pressure.

Structural

The structural evaluations for the MPC accident condition external pressure and overpack pressure differential are presented in Section 3.4 and demonstrate that all stresses are within allowable values.

Thermal

There is no effect on the thermal performance of the system as a result of this event.

Shielding

There is no effect on the shielding performance of the system as a result of this event.

Criticality

There is no effect on the criticality control features of the system as a result of this event.

Confinement

There is no effect on the confinement function of the MPC as a result of this event. As discussed in the structural evaluation above, all stresses remain within allowable values, assuring confinement boundary integrity.

Radiation Protection

Since there is no degradation in shielding or confinement capabilities as discussed above, there is no effect on occupational or public exposures as a result of this event.

Based on this evaluation, it is concluded that the explosion accident does not affect the safe operation of the HI-STORM 100 System.

11.2.11.3 Explosion Dose Calculations

The bounding external pressure load has no effect on the HI-STORM 100 overpack and MPC. Therefore, no effect on the shielding, criticality, thermal or confinement capabilities of the HI-STORM 100 System is experienced as a result of the explosion pressure load. The effects of explosion generated missiles on the HI-STORM 100 System structure is bounded by the analysis of tornado generated missiles.

11.2.11.4 Explosion Accident Corrective Action

The explosive overpressure caused by the explosion is bounded by the external pressure exerted by the flood accident. The external pressure from the flood is shown not to damage the HI-STORM 100 System. Following an explosion, the ISFSI operator shall perform a visual and radiological inspection of the overpack. If the outer shell or concrete is damaged as a result of explosion generated missiles, the concrete material may be replaced and the outer shell repaired.

11.2.12 Lightning

11.2.12.1 Cause of Lightning

The HI-STORM 100 System will be stored on an unsheltered ISFSI concrete pad. There is the potential for lightning to strike the overpack. This analysis evaluates the effects of lightning striking the overpack.

11.2.12.2 Lightning Analysis

The HI-STORM 100 System is a large metal/concrete cask stored in an unsheltered ISFSI. As such, it may be subject to lightning strikes. When the HI-STORM 100 System is hit with lightning, the lightning will discharge through the steel shell of the overpack to the ground. Lightning strikes have high currents, but their duration is short (i.e., less than a second). The overpack outer shell is composed of conductive carbon steel and, as such, will provide a direct path to ground.

The MPC provides the confinement boundary for the spent nuclear fuel. The effects of a lightning strike will be limited to the overpack. The lightning current will discharge into the overpack and directly into the ground. Therefore, the MPC will be unaffected.

The lightning accident shall have no adverse consequences on thermal, criticality, confinement, shielding, or structural performance of the HI-STORM 100 System.

Structural

There is no structural consequence as a result of this event.

Thermal

There is no effect on the thermal performance of the system as a result of this event.

Shielding

There is no effect on the shielding performance of the system as a result of this event.

Criticality

There is no effect on the criticality control features of the system as a result of this event.

Confinement

There is no effect on the confinement function of the MPC as a result of this event.

Radiation Protection

Since there is no degradation in shielding or confinement capabilities as discussed above, there is no effect on occupational or public exposures as a result of this event.

Based on this evaluation, it is concluded that the lightning accident does not affect the safe operation of the HI-STORM 100 System.

11.2.12.3 Lightning Dose Calculations

An evaluation of lightning strikes demonstrates that the effect of a lightning strike has no effect on the confinement boundary or shielding materials. Therefore, no further analysis is necessary.

11.2.12.4 Lightning Accident Corrective Action

The HI-STORM 100 System will not sustain any damage from the lightning accident. There is no surveillance or corrective action required.

11.2.13 100% Blockage of Air Inlets

11.2.13.1 Cause of 100% Blockage of Air Inlets

This event is defined as a complete blockage of all four bottom inlets. Such blockage of the inlets may be postulated to occur as a result of a flood, blizzard snow accumulation, tornado debris, or volcanic activity.

11.2.13.2 100% Blockage of Air Inlets Analysis

The immediate consequence of a complete blockage of the air inlet ducts is that the normal circulation of air for cooling the MPC is stopped. An amount of heat will continue to be removed by localized air circulation patterns in the overpack annulus and outlet ducts, and the MPC will continue to radiate heat to the relatively cooler storage overpack. As the temperatures of the MPC and its contents rise, the rate of heat rejection will increase correspondingly. Under this condition, the temperatures of the overpack, the MPC and the stored fuel assemblies will rise as a function of time.

As a result of the large mass, and correspondingly large thermal capacity, of the storage overpack (in excess of 170,000 lbs), it is expected that a significant temperature rise is only possible if the completely blocked condition is allowed to persist for a number of days. This accident condition is, however, a short duration event that will be identified and corrected by scheduled periodic surveillance at the ISFSI site. Thus, the worst possible scenario is a complete loss of ventilation air during the scheduled surveillance time interval in effect at the ISFSI site.

It is noted that there is a large thermal margin, between the maximum calculated fuel cladding temperature with design-basis fuel decay heat (Tables 4.4.9, 4.4.10 *and 4.4.26*~~, 4.4.46 and 4.4.27~~) and the short-term fuel cladding temperature limit (*Table 2.2.3*~~1058°F~~), to meet the transient short-term fuel cladding temperature excursion. In other words, the fuel stored in a HI-STORM system can heat up by over 300°F before the short-term peak temperature limit is reached. The concrete in the overpack and the MPC and overpack structural members also have significant, margins between their calculated maximum long-term temperatures and their short-term temperature limits, with which to withstand such extreme hypothetical events.

To rigorously evaluate the minimum time available before the short-term temperature limits of either the concrete, structural members or fuel cladding are exceeded, a transient thermal model of the HI-STORM System is developed. The HI-STORM system transient model with all four air inlet ducts completely blocked is created as an axisymmetric finite-volume (FLUENT) model. With the exceptions of the inlet air duct blockage and the specification of thermal inertia properties (i.e., density and heat capacity), the model is identical to the steady-state models discussed in Chapter 4 of this FSAR. ~~The model includes the lowest MPC thermal inertia of any MPC design.~~ *The MPC-68 design yields the highest normal storage condition fuel cladding temperature. This design has the smallest margin to the accident temperature limit and consequently will have the shortest time to reach this limit and is, therefore, used in performing this evaluation.*

~~In the first step of the transient solution, t~~*T*he decay heat load is set equal to *the design basis maximum value (Table 4.4.39), the inlets are blocked, and a transient solution is performed.*~~22.25 kW. and the MPC internal convection (i.e., thermosiphon) is suppressed.~~ This evaluation provides the ~~peak~~ temperatures of the fuel cladding, the MPC ~~confinement boundary~~ and the ~~concrete~~ overpack ~~shield wall~~, all as a function of time. ~~Because the MPC with the lowest thermal inertia is used in the analysis, the temperature rise results obtained from evaluation of this transient model, therefore, bound the temperature rises for all MPC designs (Table 1.2.1) under this postulated event.~~

The results of the blocked duct thermal transient evaluation are presented in ~~Figure 11.2.7 and~~ Table 11.2.9. ~~Figure 11.2.7 presents the temperature rise as a function of time after complete air inlet duct blockage for the following:~~

~~i. Fuel Cladding at the Location of Initial Maximum Temperature~~
~~ii. MPC Shell at the Location of Initial Maximum Temperature~~
~~iii. Overpack Inner Concrete at the Active Fuel Axial Mid-Height~~
~~iv. Overpack Inner Concrete at the Location of Initial Maximum Temperature~~
~~v. Overpack Outer Concrete at the Active Fuel Axial Mid-Height~~
~~vi. Overpack Outer Concrete at the Location of Initial Maximum Temperature~~

The concrete section average (i.e., through thickness), *fuel cladding, and all MPC and overpack steel component* temperatures remain~~s~~ below the*ir respective* short-term temperature limit*s* through ~~72~~ *24* hours of *continuous full* blockage. ~~Both the fuel cladding and the MPC confinement boundary temperatures remain below their respective short-term temperature limits. at 72 hours, the fuel cladding by over 150°F and the confinement boundary by almost 175°F.~~ Table 11.2.9 summarizes the *maximum* temperatures at several points in the HI-STORM System at ~~33~~*24* hours ~~and 72 hours~~ after complete inlet air duct blockage. These results establish the design-basis minimum surveillance interval ~~(i.e., 24 hours per Technical Specifications in Appendix A to the CoC)~~ for the duct screens. *As soon as at least one duct is cleared and convection flow is reestablished, convective heat dissipation begins and temperatures will begin trending toward the normal condition as the remaining ducts are cleared.*

~~Incorporation of the MPC thermosiphon internal natural convection, as described in Chapter 4, enables the maximum design basis decay heat load to rise to about 29 kW. The thermosiphon effect also shifts the highest temperatures in the MPC enclosure vessel toward the top of the MPC. The peak MPC closure plate outer surface temperature, for example, is computed to be about 450°F in the thermosiphon enabled solution compared to about 210°F in the thermosiphon suppressed solution, with both solutions computing approximately the same peak clad temperature. In the 100% inlet duct blockage condition, the heated MPC closure plate and MPC shell become effective heat dissipaters because of their proximity to the overpack outlet ducts and by virtue of the fact that thermal radiation heat transfer rises at the fourth power of absolute temperature. As a result of this increased heat rejection from the upper region of the MPC, the time limit for reaching the short-term peak fuel cladding temperature limit (72 hours) remains applicable.~~

~~It should be noted that the rupture of 100% of the fuel rods and the subsequent release of the contained rod gases has a significant positive impact on the MPC internal thermosiphon heat transport mechanism. The increase in the MPC internal pressure accelerates the thermosiphon, as does the introduction of higher molecular weight gaseous fission products. The values reported in Table 11.2.9 do not reflect this improved heat transfer and will actually be lower than reported. Crediting the increased MPC internal pressure only and neglecting the higher molecular weights of the gaseous fission products, the MPC bulk average gas temperature will be reduced by approximately 34.5°C (62.1°F).~~

Under the complete air inlet ducts blockage accident condition, it must be demonstrated that the MPC internal pressure does not exceed its design-basis accident limit during this event. Chapter 4 presented the MPC internal pressure calculated at an ambient temperature of 80°F, ~~100% fuel rods ruptured,~~ full insolation, and maximum decay heat. This calculated *bounding* pressure is *97.9 psig* ~~174.8~~ *(112.6* psia*)*, as reported in Table 4.4.14, at an average temperature of ~~513.6~~*528.0*°K. Using this pressure, a*n* ~~bounding~~ increase in the MPC cavity *bulk* temperature of 184°F (102.2°K, ~~maximum of MPC shell or fuel cladding~~*MPC cavity bulk* temperature rise ~~33~~*24* hours after blockage of all four ducts~~, see Table 11.2.9), the reduction in the bulk average gas temperature of 34.5°C,~~ and the ideal gas law, the resultant MPC internal pressure is calculated below.

$$\frac{P_1}{P_2} = \frac{T_1}{T_2}$$

$$P_2 = \frac{P_1 T_2}{T_1}$$

$$P_2 = \frac{(112.6\,\text{psi}\,a)\,(528.0°\text{K} + 102.2°\text{K})}{528.0°\text{K}}$$

$$P_2 = 134.4\ \text{psia or } 119.7\ \text{psig}$$

The accident MPC internal design pressure ~~of 200 psig~~ (Table 2.2.1) bounds the resultant pressure calculated above. Therefore, no additional analysis is required.

Structural

There are no structural consequences as a result of this event.

Thermal

Thermal analysis is performed to determine the time until the ~~concrete section average and peak fuel cladding~~*HI-STORM System components and contents* temperatures approach their short-term temperature limits. At the specified time limit, ~~both the concrete section average and peak fuel cladding~~*all components and contents* temperatures remain below their short-term temperature limits. The MPC internal pressure for this event is calculated as presented above. As can be seen from the value above, the ~~200 psig~~ design basis internal pressure for accident conditions used in the structural evaluation bounds the calculated value above.

~~To demonstrate the robustness of the HI-STORM System design, the results of the parametric study of incremental duct blockage performed in Subsection 11.1.4 are examined again. Even with three air inlet ducts completely blocked, as shown in Table 11.1.2, large steady-state margins against the short-term temperature limits exist for all system components and the fuel cladding of the stored assemblies. Both the peak fuel cladding and overpack concrete section average temperatures, which approach their limiting temperatures under the 100% blockage condition, with a single open duct are~~

~~approximately 240°F and 100°F, respectively, less than their respective short-term temperature limits. These results show that only a relatively small amount of the total air inlet duct area, on the order of 25% or less, must remain open to prevent exceeding system short-term temperature limits under steady-state conditions.~~

Shielding

There is no effect on the shielding performance of the system as a result of this event, since the concrete temperatures do not exceed the short-term condition design temperature provided in Table 2.2.3.

Criticality

There is no effect on the criticality control features of the system as a result of this event.

Confinement

There is no effect on the confinement function of the MPC as a result of this event.

Radiation Protection

Since there is no degradation in shielding or confinement capabilities as discussed above, there is no effect on occupational or public exposures as a result of this event.

Based on this evaluation, it is concluded that the 100% blockage of air inlets accident does not affect the safe operation of the HI-STORM 100 System, if the blockage is removed in the specified time period. ~~The Technical Specifications in Appendix A to the CoC specify the time interval to ensure that the blockage duration cannot exceed the time limit calculated herein.~~

11.2.13.3 100% Blockage of Air Inlets Dose Calculations

As shown in the analysis of the 100% blockage of air inlets accident, the shielding capabilities of the HI-STORM 100 System are unchanged because the peak concrete temperature does not exceed its short-term condition design temperature. The elevated temperatures will not cause the breach of the confinement system and the short term fuel cladding temperature limit is not exceeded. Therefore, there is no radiological impact.

11.2.13.4 100% Blockage of Air Inlets Accident Corrective Action

Analysis of the 100% blockage of air inlet ducts accident shows that the *cask system components and contents*~~overpack concrete section average and fuel cladding peak~~ temperatures ~~remain substantially below their short-term temperature~~ *are within the accident temperature* limits if the blockage is cleared within ~~72~~ *24* hours. Upon detection of the complete blockage of the air inlet ducts, the ISFSI

operator shall assign personnel to clear the blockage with mechanical and manual means as necessary. After clearing the overpack ducts, the overpack shall be visually and radiologically inspected for any damage. ~~Per the Technical Specifications in Appendix A to the CoC, visual inspection of the duct screens is specified on a frequency of 24 hours, or air outlet temperature monitoring is required. Therefore, an undetected blockage event could not exceed 24 hours.~~

If exit air temperature monitoring is performed in lieu of direct visual inspections, the difference between the ambient air temperature and the exit air temperature will be the basis for assurance that the temperature limits are not exceeded. A measured temperature difference between the ambient air and the exit air that exceeds the design-basis maximum air temperature rise, calculated in Section 4.4.2, will indicate blockage of the overpack air ducts.

For an accident event that completely blocks the inlet or outlet air ducts, a site-specific evaluation or analysis may be performed to demonstrate that adequate heat removal is available for the duration of the event. Adequate heat removal is defined as *cask system components and contents*~~overpack concrete section average and fuel cladding~~ temperatures remaining below their short term temperature limits. For those events where an evaluation or analysis is not performed or is not successful in showing that ~~fuel cladding~~ temperatures remain below the*ir* short term temperature limit*s*, the site's emergency plan shall include provisions to address removal of the material blocking the air inlet ducts and to provide alternate means of cooling prior to exceeding the time when the fuel cladding temperature reaches its short-term temperature limit. Alternate means of cooling could include, for example, spraying water into the air outlet ducts using pumps or fire-hoses or blowing air into the air outlet ducts using fans, to directly cool the MPC. Another example of supplemental cooling, for sufficiently low decay heat loads, would be to remove the overpack lid to increase free-surface natural convection.

11.2.14 Burial Under Debris

11.2.14.1 Cause of Burial Under Debris

Burial of the HI-STORM System under debris is not a credible accident. During storage at the ISFSI, there are no structures over the casks. The minimum regulatory distance of 100 meters from the ISFSI to the nearest site boundary and the controlled area around the ISFSI concrete pad precludes the close proximity of substantial amounts of vegetation.

There is no credible mechanism for the HI-STORM System to become completely buried under debris. However, for conservatism, complete burial under debris is considered. Blockage of the HI-STORM overpack air inlet ducts has already been considered in Subsection 11.2.13.

11.2.14.2 Burial Under Debris Analysis

Burial of the HI-STORM System does not impose a condition that would have more severe consequences for criticality, confinement, shielding, and structural analyses than that performed for

the other accidents analyzed. The debris would provide additional shielding to reduce radiation doses. The accident external pressure encountered during the flood bounds any credible pressure loading caused by the burial under debris.

Burial under debris can affect thermal performance because the debris acts as an insulator and heat sink. This will cause the HI-STORM System and fuel cladding temperatures to increase. A thermal analysis has been performed to determine the time for the fuel cladding temperatures to reach the short term accident condition temperature limit during a burial under debris accident.

To demonstrate the inherent safety of the HI-STORM System, a bounding analysis that considers the debris to act as a perfect insulator is considered. Under this scenario, the contents of the HI-STORM System will undergo a transient heat up under adiabatic conditions. The minimum time required for the fuel cladding to reach the short term design fuel cladding temperature limit depends on the amount of thermal inertia of the cask, the cask initial conditions, and the spent nuclear fuel decay heat generation.

As stated in Subsection 11.2.13.2, there is a margin of over 300°F between the maximum calculated fuel cladding temperature and the short-term fuel cladding temperature limit. If a highly conservative 150°F is postulated as the permissible fuel cladding temperature rise for the burial under debris scenario, then a curve representing the relationship between the time required and decay heat load can be constructed. This curve is shown in Figure 11.2.6. In this figure, plots of the burial period at different levels of heat generation in the MPC are shown based on a 150°F rise in fuel cladding temperature resulting from transient heating of the HI-STORM System. Using the values stated in Table 11.2.6, the allowable time before the cladding temperatures meet the short-term fuel cladding temperature limit can be determined using:

$$\Delta t = \frac{m \times c_p \times \Delta T}{Q}$$

where:

Δt = Allowable Burial Time (hrs)
m = Mass of HI-STORM System (lb)
c_p = Specific Heat Capacity (Btu/lb×°F)
ΔT = Permissible Fuel Cladding Temperature Rise (150°F)
Q = Total Decay Heat Load (Btu/hr)

The allowable burial time as a function of total decay heat load (Q) is presented in Figure 11.2.6.

The MPC cavity internal pressure under this accident scenario is bounded by the calculated internal pressure for the hypothetical 100% air inlets blockage previously evaluated in Subsection 11.2.13.2.

Structural

The structural evaluation of the MPC enclosure vessel for accident internal pressure conditions bounds the pressure calculated herein. Therefore, the resulting stresses from this event are well within the allowable values, as demonstrated in Section 3.4.

Thermal

With the cladding temperature rise limited to 150°F, the corresponding pressure rise, bounded by the calculations in Subsection 11.2.13.2, demonstrates large margins of safety for the MPC vessel structural integrity. Consequently, cladding integrity and confinement function of the MPC are not compromised.

Shielding

There is no effect on the shielding performance of the system as a result of this event.

Criticality

There is no effect on the criticality control features of the system as a result of this event.

Confinement

There is no effect on the confinement function of the MPC as a result of this event. As discussed in the structural evaluation above, all stresses remain within allowable values, assuring confinement boundary integrity.

Radiation Protection

Since there is no degradation in shielding or confinement capabilities as discussed above, there is no effect on occupational or public exposures as a result of this event.

Based on this evaluation, it is concluded that the burial under debris accident does not affect the safe operation of the HI-STORM 100 System, if the debris is removed within the specified time (Figure 11.2.6). The 24-hour minimum duct inspection interval ~~specified in the Technical Specification in Appendix A to the CoC~~ ensures that a burial under debris condition will be detected long before the allowable burial time is reached.

11.2.14.3 Burial Under Debris Dose Calculations

As discussed in burial under debris analysis, the shielding is enhanced while the HI-STORM System is covered.

The elevated temperatures will not cause the breach of the confinement system and the short term fuel cladding temperature limit is not exceeded. Therefore, there is no radiological impact.

11.2.14.4 Burial Under Debris Accident Corrective Action

Analysis of the burial under debris accident shows that the fuel cladding peak temperatures will not exceed the short term limit if the debris is removed within ~~45~~*34* hours. Upon detection of the burial under debris accident, the ISFSI operator shall assign personnel to remove the debris with mechanical and manual means as necessary. After uncovering the storage overpack, the storage overpack shall be visually and radiologically inspected for any damage. The loaded MPC shall be removed from the storage overpack with the HI-TRAC transfer cask to allow complete inspection of the overpack air inlets and outlets, and annulus. Removal of obstructions to the air flow path shall be performed prior to the re-insertion of the MPC. The site's emergency action plan shall include provisions for the performance of this corrective action.

11.2.15 Extreme Environmental Temperature

11.2.15.1 Cause of Extreme Environmental Temperature

The extreme environmental temperature is postulated as a constant ambient temperature caused by extreme weather conditions. To determine the effects of the extreme temperature, it is conservatively assumed that the temperature persists for a sufficient duration to allow the HI-STORM 100 System to achieve thermal equilibrium. Because of the large mass of the HI-STORM 100 System, with its corresponding large thermal inertia and the limited duration for the extreme temperature, this assumption is conservative.

11.2.15.2 Extreme Environmental Temperature Analysis

The accident condition considering an environmental temperature of 125°F for a duration sufficient to reach thermal equilibrium is evaluated with respect to accident condition design temperatures listed in Table 2.2.3. The evaluation is performed with design basis fuel with the maximum decay heat and the most restrictive thermal resistance. The 125°F environmental temperature is applied with full solar insolation.

The HI-STORM 100 System maximum temperatures for components close to the design basis temperatures are listed in Section 4.4. These temperatures are conservatively calculated at an environmental temperature of 80°F. The extreme environmental temperature is 125°F, which is an increase of 45°F. Conservatively bounding temperatures for all the MPC designs are obtained and reported in Table 11.2.7. As illustrated by the table, all the temperatures are well below the accident condition design basis temperatures. The extreme environmental temperature is of a short duration (several consecutive days would be highly unlikely) and the resultant temperatures are evaluated against short-term accident condition temperature limits. Therefore, the HI-STORM 100 System extreme environmental temperatures meet the design requirements.

Additionally, the extreme environmental temperature generates a pressure that is bounded by the pressure calculated for the complete inlet duct blockage condition because the duct blockage condition temperatures are much higher than the temperatures that result from the extreme environmental temperature. As shown in Subsection 11.2.13.2, the accident condition pressures are below the accident limit specified in Table 2.2.1.

Structural

The structural evaluation of the MPC enclosure vessel for accident condition internal pressure bounds the pressure resulting from this event. Therefore, the resulting stresses from this event are bounded by that of the accident condition and are well within the allowable values, as discussed in Section 3.4.

Thermal

The resulting temperatures for the system and fuel assembly cladding are provided in Table 11.2.7. As can be seen from this table, all temperatures are within the short-term accident condition allowable values specified in Table 2.2.3.

Shielding

There is no effect on the shielding performance of the system as a result of this event, since the concrete temperature does not exceed the short-term temperature limit specified in Table 2.2.3.

Criticality

There is no effect on the criticality control features of the system as a result of this event.

Confinement

There is no effect on the confinement function of the MPC as a result of this event. As discussed in the structural evaluation above, all stresses remain within allowable values, assuring confinement boundary integrity.

Radiation Protection

Since there is no degradation in shielding or confinement capabilities as discussed above, there is no effect on occupational or public exposures as a result of this event.

Based on this evaluation, it is concluded that the extreme environment temperature accident does not affect the safe operation of the HI-STORM 100 System.

11.2.15.3 Extreme Environmental Temperature Dose Calculations

The extreme environmental temperature will not cause the concrete to exceed its normal design temperature. Therefore, there will be no degradation of the concrete's shielding effectiveness. The elevated temperatures will not cause a breach of the confinement system and the short-term fuel cladding temperature is not exceeded. Therefore, there is no radiological impact on the HI-STORM 100 System for the extreme environmental temperature and the dose calculations are equivalent to the normal condition dose rates.

11.2.15.4 Extreme Environmental Temperature Corrective Action

There are no consequences of this accident that require corrective action.

Table 11.2.1

INTENTIONALLY DELETED

Table 11.2.2

HI-STORM 100 OVERPACK ~~MAXIMUM~~ *BOUNDING* TEMPERATURES
AS A RESULT OF THE HYPOTHETICAL FIRE CONDITION

Material/Component	**Initial† Condition (°F)**	**During Fire (°F)**	**Post-Fire†† Cooldown (°F)**
Fuel Cladding	~~691 (MPC-24)~~ ~~691 (MPC-24E)~~ ~~691 (MPC-32)~~ ~~740 (MPC-68)~~*731*	~~692 (MPC-24)~~ ~~692 (MPC-24E)~~ ~~692 (MPC-32)~~ ~~741 (MPC-68)~~*732*	~~692 (MPC-24)~~ ~~692 (MPC-24E)~~ ~~692 (MPC-32)~~ ~~741 (MPC-68)~~*732*
MPC Fuel Basket	~~650 (MPC-24)~~ ~~650 (MPC-24E)~~ ~~660 (MPC-32)~~ ~~720 (MPC-68)~~*706*	~~651 (MPC-24)~~ ~~651 (MPC-24E)~~ ~~661 (MPC-32)~~ ~~721 (MPC-68)~~*707*	~~651 (MPC-24)~~ ~~651 (MPC-24E)~~ ~~661 (MPC-32)~~ ~~721 (MPC-68)~~*707*
Overpack Inner Shell	~~195~~*250*	300	~~195~~*300*
~~Overpack Radial Concrete Inner Surface~~	~~195~~	~~281~~	~~282~~
Overpack Radial Concrete Mid-Surface	~~173~~*222*	~~173~~*222*	~~184~~*235*
~~Overpack Radial Concrete Outer Surface~~	~~157~~	~~529~~	~~530~~
Overpack Outer Shell	~~157~~*200*	~~570~~*585*	~~570~~*585*

† Bounding ~~195~~*250*°F uniform inner surface and ~~157~~*200*°F uniform outer surface temperatures assumed.

†† Maximum temperature during post-fire cooldown.

Table 11.2.3

SUMMARY OF INPUTS FOR HI-TRAC FIRE ACCIDENT HEAT-UP

Minimum Weight of Loaded HI-TRAC with Pool Lid (lb)	180,436
Lower Heat Capacity of Carbon Steel (Btu/lbm·°R)	0.1
Heat Capacity UO_2 (Btu/lbm·°R)	0.056
Heat Capacity Lead (Btu/lbm·°R)	0.031
Maximum Decay Heat (kW)	~~28.7~~*438*
Total Fuel Assembly Weight (lb)	40,320
Lead Weight (lb)	52,478
Water Weight (lb)	7,595

Table 11.2.4

BOUNDING HI-TRAC HYPOTHETICAL
FIRE CONDITION PRESSURES[‡]

Condition	**Pressure (psig)**			
	~~**MPC-24**~~	~~**MPC-24E**~~	~~**MPC-32**~~	~~**MPC-68**~~
~~Without Fuel Rod Rupture~~*Initial Condition*	~~79.8~~*107.4*	~~79.8~~	~~79.8~~	~~79.8~~
~~With 100% Fuel Rod Rupture~~*Bounding Maximum*	~~158.9~~*110.6*	~~159.3~~	~~191.1~~	~~126.6 (72.48-488)~~

~~‡ The reported pressures are based on temperatures that exceed the calculated maximum temperatures and are therefore slightly conservative.~~

Table 11.2.5

SUMMARY OF BOUNDING MPC PEAK TEMPERATURES
DURING A HYPOTHETICAL HI-TRAC FIRE ACCIDENT CONDITION

Location	**Initial Steady State Temperature [°F]**	**Bounding Temperature Rise [°F]**	**Hottest MPC Cross Section Peak Temperature [°F]**
Fuel Cladding	~~872~~~~600~~~~455~~*745*	~~26.3~~*326.6*	~~898.3~~*771.6*
Basket Periphery	*620*	~~26.3~~*326.6*	~~626.3~~*646.6*
MPC Shell	*433*	~~26.3~~*326.6*	~~481.3~~*459.6*

Table 11.2.6

SUMMARY OF INPUTS FOR ADIABATIC CASK HEAT-UP

Minimum Weight of HI-STORM 100 System (lb) (overpack and MPC)	300,000
Lower Heat Capacity of Carbon Steel (BTU/lb/°F)	0.1
Initial Uniform Temperature of Cask (°F)	740†
~~Bounding Decay Heat (kW)~~	~~28.74~~

† The cask is conservatively assumed to be at a uniform temperature *that conservatively bounds all normal storage temperatures.* ~~equal to the maximum fuel cladding temperature.~~

Table 11.2.7

MAXIMUM TEMPERATURES CAUSED BY EXTREME ENVIRONMENTAL TEMPERATURES[‡] [°F]

Location	Temperature	~~Accident Temperature Limit~~
Fuel Cladding	~~736 (PWR)~~ ~~785 (BWR)~~ *776*	~~1058~~
MPC Basket	~~765~~ *751*	~~950~~
MPC Shell	~~396~~*515*	~~775~~
Overpack Air Exit	~~251~~ *261*	~~N/A~~
Overpack Inner Shell	~~244~~ *288*	~~350 (overpack concrete)~~
Overpack Outer Shell	~~190~~ *228*	~~350 (overpack concrete)~~

~~‡ Conservatively bounding temperatures reported include a hypothetical rupture of 10% of the fuel rods.~~

Table 11.2.8

~~MAXIMUM~~ *BOUNDING MPC* TEMPERATURES CAUSED BY LOSS OF WATER FROM THE HI-TRAC WATER JACKET [°F]

Temperature Location	**Normal**	**Calculated Without Water in Water Jacket**	~~**Accident Condition Design Temperature**~~
Fuel Cladding	~~872~~*745*	~~888~~*761*	~~1058 short-term~~
MPC Basket	~~852~~*728*	~~868~~*745*	~~950 short-term~~
MPC Basket Periphery	~~600~~*620*	~~612~~*630*	~~950 short-term~~
MPC Shell	~~455~~*433*	~~466~~*443*	~~775 short-term~~
~~HI-TRAC Inner Shell~~	~~322~~	~~342~~	~~400 long-term~~ ~~600 short-term~~
~~HI-TRAC Water Jacket Inner Surface~~	~~314~~	~~334~~	~~350 long-term~~
~~HI-TRAC Enclosure Shell Outer Surface~~	~~224~~	~~222~~	~~350 long-term~~
~~Axial Neutron Shield~~~~†~~	~~258~~	~~261~~	~~300 long-term~~

~~Note: Where it can be shown that the temperatures are below the normal long-term condition limits, the calculated temperatures are compared to the normal long-term temperature limits for conservatism. The corresponding short-term temperature limits are higher temperatures as presented in Table 2.2.3.~~

~~† Local maximum section temperature.~~

Table 11.2.9

SUMMARY OF BLOCKED AIR INLET DUCT EVALUATION RESULTS

	Max. Initial Steady-State Temp.[†](°F)	Temperature Rise (°F)		Transient Temperature (°F)		~~Short-Term Temperature Limit (°F)~~
		at ~~33~~*24* hrs	~~at 72 hrs~~	at *24*~~33~~ hrs	~~at 72 hrs~~	
Fuel Cladding	~~740~~*731*	~~101~~*222*	~~160~~	~~841~~*953*	~~900~~	~~1058~~
MPC Shell	~~351~~*470*	~~184~~*123*	~~250~~	~~535~~*593*	~~601~~	~~775~~
Overpack Inner Shell ~~#1~~[††]	~~199~~*243*	~~113~~*156*	~~174~~	~~312~~*399*	~~373~~	~~600~~
~~Overpack Inner Shell #2[†††]~~	~~155~~	~~193~~	~~286~~	~~348~~	~~441~~	~~600~~
~~Overpack Outer Shell~~	~~145~~	~~14~~	~~40~~	~~159~~	~~185~~	~~600~~
Concrete Section Average	~~172~~*183*	~~79~~*53*	~~141~~	~~251~~*236*	~~313~~	~~350~~

~~[†] Conservatively bounding temperatures reported includes a hypothetical rupture of 10% of the fuel rods.~~

~~[††] Coincident with location of initial maximum.~~

~~[†††] Coincident with active fuel axial mid-height.~~

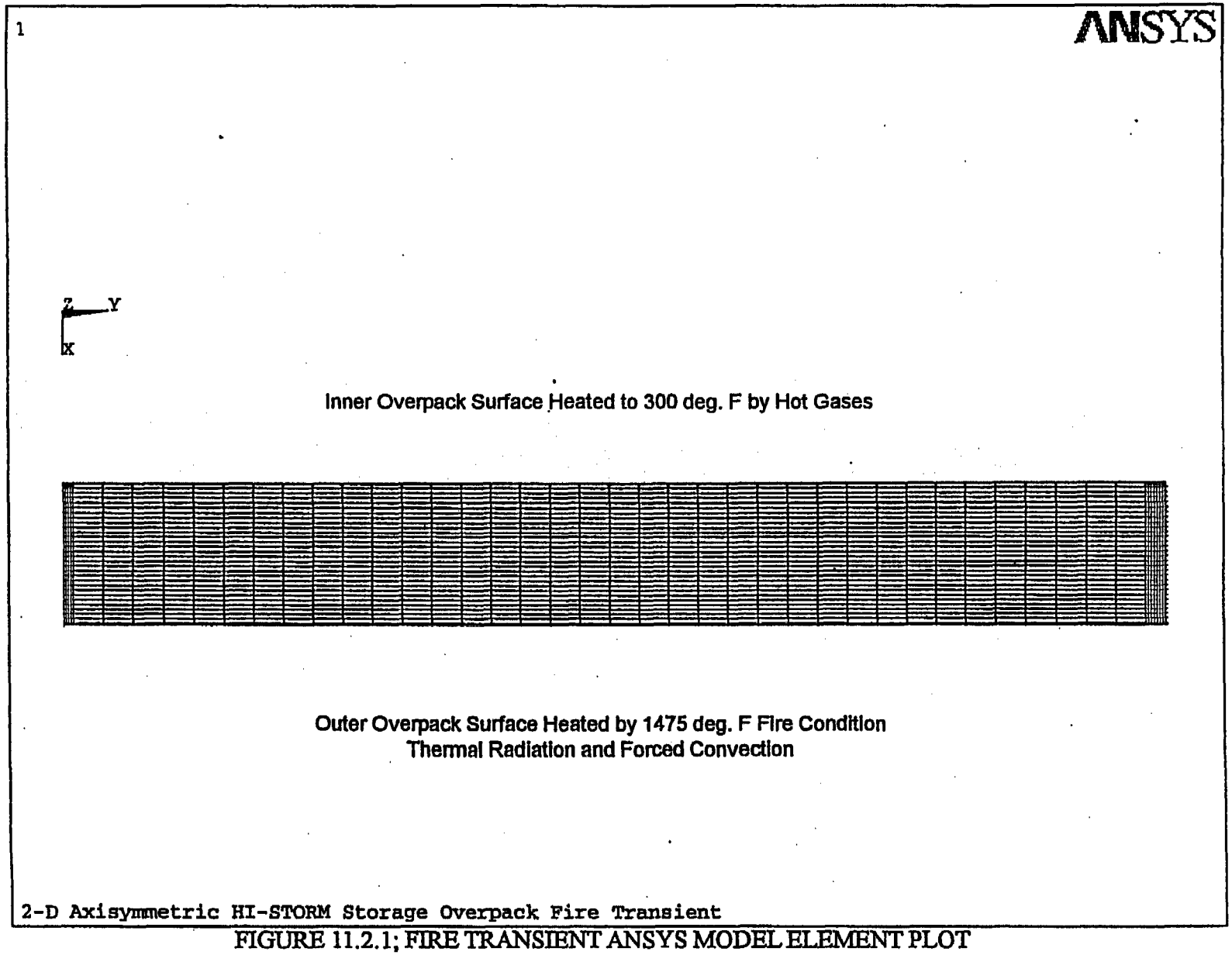


FIGURE 11.2.1; FIRE TRANSIENT ANSYS MODEL ELEMENT PLOT

FIGURES 11.2.2 THROUGH 11.2.5

[INTENTIONALLY DELETED]

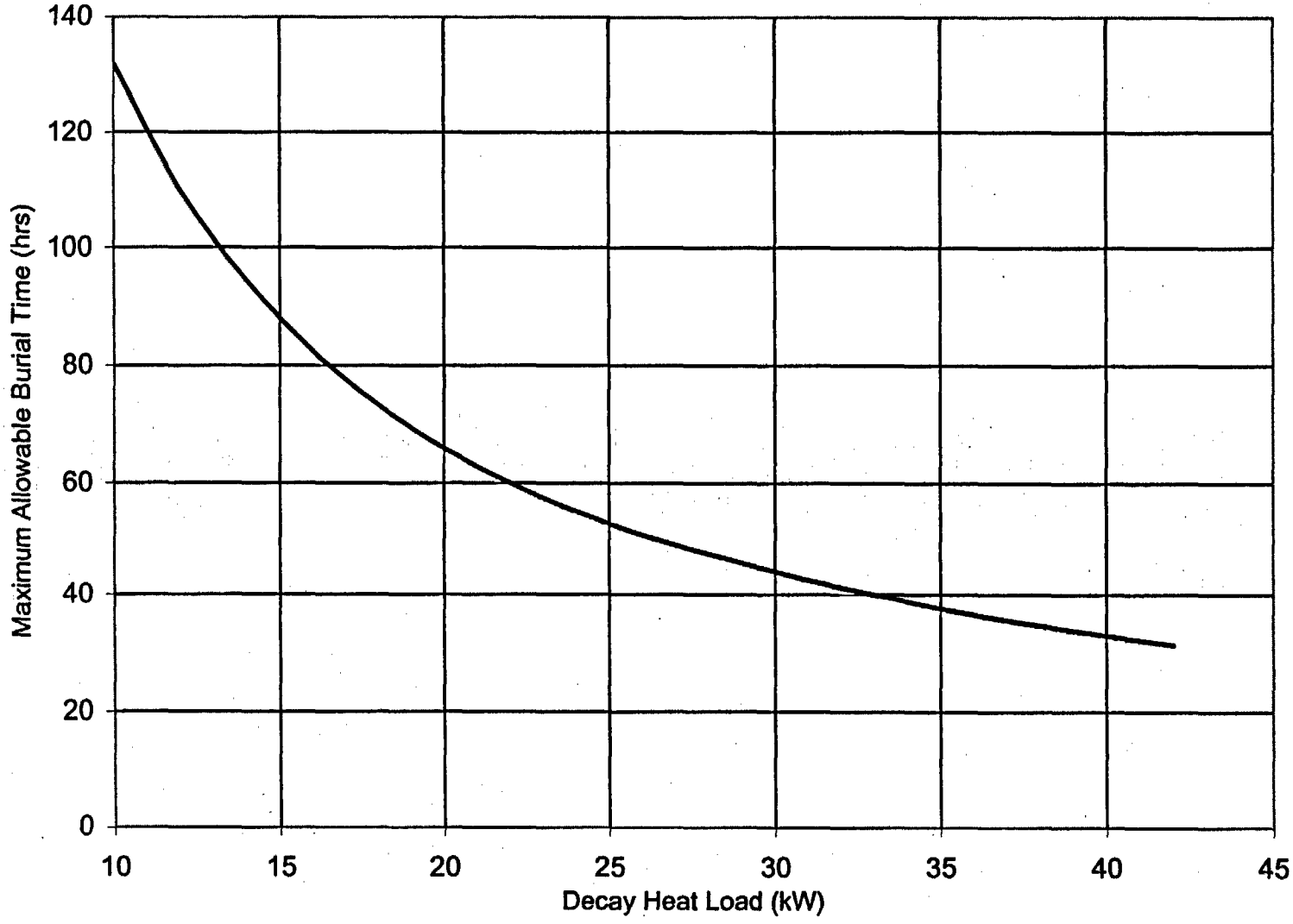


Figure 11.2.6: ALLOWABLE BURIAL UNDER DEBRIS TIME VERSUS DECAY HEAT LOAD.

FIGURE 11.2.7

[INTENTIONALLY DELETED]

12.1 PROPOSED OPERATING CONTROLS AND LIMITS

12.1.1 NUREG-1536 (Standard Review Plan) Acceptance Criteria

12.1.1.1 This portion of the FSAR establishes the commitments regarding the HI-STORM 100 System and its use. Other 10CFR72 [12.1.2] and 10CFR20 [12.1.3] requirements in addition to the Technical Specifications may apply. The conditions for a general license holder found in 10CFR72.212 [12.1.2] shall be met by the licensee prior to loading spent fuel into the HI-STORM 100 System. The general license conditions governed by 10CFR72 [12.1.2] are not repeated with these Technical Specifications. Licensees are required to comply with all commitments and requirements.

12.1.1.2 The Technical Specifications provided in Appendix A to CoC 72-1014 and the authorized contents and design features provided in Appendix B to CoC 72-1014 are primarily established to maintain subcriticality, confinement boundary and intact fuel cladding integrity, shielding and radiological protection, heat removal capability, and structural integrity under normal, off-normal and accident conditions. Table 12.1.1 addresses each of these conditions respectively and identifies the appropriate Technical Specification(s) designed to control the condition. Table 12.1.2 provides the list of Technical Specifications for the HI-STORM 100 System.

Table 12.1.1

HI-STORM 100 SYSTEM CONTROLS

Condition to be Controlled	Applicable Technical Specifications†
Criticality Control	~~Refer to Appendix B to Certificate of Compliance 72-1014 for fuel specifications and design features~~ 3.3.1 Boron Concentration
Confinement Boundary and Intact Fuel Cladding Integrity	3.1.1 Multi-Purpose Canister (MPC) 5.6 ~~Fuel Cladding Oxide Thickness Evaluation Program~~ *3.4.10 TRANSFER CASK Operating Limits ~~(CoC 72-1014, Appendix B – Design Features)~~*
Shielding and Radiological Protection	~~Refer to Appendix B to Certificate of Compliance 72-1014 for fuel specifications and design features~~ 3.1.1 Multi-Purpose Canister (MPC) 3.1.3 Fuel Cool-Down 3.2.1 ~~TRANSFER CASK Average Surface Dose Rates~~ 3.2.2 TRANSFER CASK Surface Contamination ~~3.2.3OVERPACK Average Surface Dose Rates~~ *5.7 Radiation Protection Program*
Heat Removal Capability	~~Refer to Appendix B to Certificate of Compliance 72-1014 for fuel specifications and design features~~ 3.1.1 Multi-Purpose Canister (MPC) 3.1.2 SFSC Heat Removal System
Structural Integrity	3.5 Cask Transfer Facility (CTF) ~~(CoC 72-1014, Appendix B – Design Features)~~ 5.5 Cask Transport Evaluation Program

† Technical Specifications are located in Appendix A to CoC 72-1014. *Authorized contents are specified in FSAR Section 2.1.9*

Table 12.1.2

HI-STORM 100 SYSTEM TECHNICAL SPECIFICATIONS

NUMBER	TECHNICAL SPECIFICATION
1.0	USE AND APPLICATION 1.1 Definitions 1.2 Logical Connectors 1.3 Completion Times 1.4 Frequency
2.0	Not Used. ~~Refer to Appendix B to CoC 72-1014 for fuel specifications.~~
3.0	LIMITING CONDITION FOR OPERATION (LCO) APPLICABILITY SURVEILLANCE REQUIREMENT (SR) APPLICABILITY
3.1.1	Multi-Purpose Canister (MPC)
3.1.2	SFSC Heat Removal System
3.1.3	Fuel Cool-Down
3.2.1	~~TRANSFER CASK Average Surface Dose Rates~~*Deleted*
3.2.2	TRANSFER CASK Surface Contamination
3.2.3	~~OVERPACK Average Surface Dose Rates~~*Deleted*
3.3.1	Boron Concentration
Table 3-1	MPC ~~Model Dependent~~*Cavity Drying* Limits
Table 3-2	*MPC Helium Backfill Limits*
4.0	Not Used. ~~Refer to Appendix B to CoC 72-1014 for design features.~~
5.0	ADMINSTRATIVE CONTROLS AND PROGRAMS
5.1	Deleted
5.2	Deleted
5.3	Deleted
5.4	Radioactive Effluent Control Program
5.5	Cask Transport Evaluation Program
5.6	~~Fuel Cladding Oxide Thickness Evaluation Program~~*Deleted*
5.7	*Radiation Protection Program*
Table 5-1	TRANSFER CASK and OVERPACK Lifting Requirements

12.2 DEVELOPMENT OF OPERATING CONTROLS AND LIMITS

This section provides a discussion of the operating controls and limits, *and training requirements* for the HI-STORM 100 System to assure long-term performance consistent with the conditions analyzed in this FSAR. ~~In addition to the controls and limits provided in the Technical Specifications contained in Appendix A to Certificate of Compliance 72-1014 and the Approved Contents and Design Features in Appendix B to Certificate of Compliance 72-1014, the licensee shall ensure that the following training and dry run activities are performed.~~

12.2.1 Training Modules

Training modules are to be developed under the licensee's training program to require a comprehensive, site-specific training, assessment, and qualification (including periodic re-qualification) program for the operation and maintenance of the HI-STORM 100 Spent Fuel Storage Cask (SFSC) System and the Independent Spent Fuel Storage Installation (ISFSI). The training modules shall include the following elements, at a minimum:

1. HI-STORM 100 System Design (overview);
2. ISFSI Facility Design (overview);
3. Systems, Structures, and Components Important to Safety (overview)
4. HI-STORM 100 System Final Safety Analysis Report (overview);
5. NRC Safety Evaluation Report (overview);
6. Certificate of Compliance conditions;
7. HI-STORM 100 Technical Specifications, Approved Contents, Design Features and other Conditions for Use;
8. HI-STORM 100 Regulatory Requirements (e.g., 10CFR72.48, 10CFR72, Subpart K, 10CFR20, 10CFR73);
9. Required instrumentation and use;
10. Operating Experience Reviews

11. HI-STORM 100 System and ISFSI Procedures, including

- Procedural overview
- Fuel qualification and loading
- MPC /HI-TRAC/overpack rigging and handling, including safe load pathways
- MPC welding operations
- HI-TRAC/overpack closure
- Auxiliary equipment operation and maintenance (e.g., draining, moisture removal, helium backfilling, and cooldown)
- MPC/HI-TRAC/overpack pre-operational and in-service inspections and tests
- Transfer and securing of the loaded HI-TRAC/overpack onto the transport vehicle
- Transfer and offloading of the HI-TRAC/overpack
- Preparation of MPC/HI-TRAC/overpack for fuel unloading
- Unloading fuel from the MPC/HI-TRAC/overpack
- Surveillance
- Radiation protection
- Maintenance
- Security
- Off-normal and accident conditions, responses, and corrective actions

12.2.2 Dry Run Training

A dry run training exercise of the loading, closure, handling, and transfer of the HI-STORM 100 System shall be conducted by the licensee prior to the first use *of* the system to load spent fuel assemblies. The dry run shall include, but is not limited to the following:

1. Receipt inspection of HI-STORM 100 System components.

2. Moving the HI-STORM 100 MPC/HI-TRAC into the spent fuel pool.

3. Preparation of the HI-STORM 100 System for fuel loading.

4. Selection and verification of specific fuel assemblies to ensure type conformance.

5. Locating specific assemblies and placing assemblies into the MPC (using a dummy fuel assembly), including appropriate independent verification.

6. Remote installation of the MPC lid and removal of the MPC/HI-TRAC from the spent fuel pool.

7. Replacing the HI-TRAC pool lid with the transfer lid (HI-TRAC 100 and 125 only).

8. MPC welding, NDE inspections, hydrostatic testing, draining, moisture removal, helium backfilling and leakage testing (for which a mockup may be used).

9. HI-TRAC upending/downending on the horizontal transfer trailer or other transfer device, as applicable to the site's cask handling arrangement.

10. Placement of the HI-STORM 100 System at the ISFSI.

11. HI-STORM 100 System unloading, including cooling fuel assemblies, flooding the MPC cavity, and removing MPC welds (for which a mock-up may be used).

12.2.3 Functional and Operating Limits, Monitoring Instruments, and Limiting Control Settings

The controls and limits apply to operating parameters and conditions which are observable, detectable, and/or measurable. The HI-STORM 100 System is completely passive during storage and requires no monitoring instruments. The user may choose to implement a temperature monitoring system to verify operability of the overpack heat removal system in accordance with Technical Specification Limiting Condition for Operation (LCO) 3.1.2.

12.2.4 Limiting Conditions for Operation

Limiting Conditions for Operation specify the minimum capability or level of performance that is required to assure that the HI-STORM 100 System can fulfill its safety functions.

12.2.5 Equipment

The HI-STORM 100 System and its components have been analyzed for specified normal, off-normal, and accident conditions, including extreme environmental conditions. Analysis has shown in this FSAR that no credible condition or event prevents the HI-STORM 100 System from meeting its safety function. As a result, there is no threat to public health and safety from any postulated accident condition or analyzed event. When all equipment is loaded, tested, and placed into storage in accordance with procedures developed for the ISFSI, no failure of the system to perform its safety function is expected to occur.

12.2.6 Surveillance Requirements

The analyses provided in this FSAR show that the HI-STORM 100 System fulfills its safety functions, provided that the Technical Specifications ~~in Appendix A to CoC 72-1014~~ and the Authorized Contents ~~and Design Features in Appendix B to CoC 72-1014~~*described in Section 2.1.9* are met. Surveillance requirements during loading, unloading, and storage operations are provided in the Technical Specifications.

12.2.7 Design Features

This section describes HI-STORM 100 System design features that are Important to Safety. These features require design controls and fabrication controls. The design features, detailed in this FSAR and in Appendix B to CoC 72-1014, are established in specifications and drawings which are controlled through the quality assurance program. Fabrication controls and

inspections to assure that the HI-STORM 100 System is fabricated in accordance with the design drawings and the requirements of this FSAR are described in Chapter 9.

12.2.8 MPC

a. Basket material composition, properties, dimensions, and tolerances for criticality control.

b. Canister material mechanical properties for structural integrity of the confinement boundary.

c. Canister and basket material thermal properties and dimensions for heat transfer control.

d. Canister and basket material composition and dimensions for dose rate control.

12.2.9 HI-STORM Overpack

a HI-STORM overpack material mechanical properties and dimensions for structural integrity to provide protection of the MPC and shielding of the spent nuclear fuel assemblies during loading, unloading and handling operations.

b. HI-STORM overpack material thermal properties and dimensions for heat transfer control.

c. HI-STORM overpack material composition and dimensions for dose rate control.

12.2.10 Verifying Compliance with Fuel Assembly Decay Heat, Burnup, and Cooling Time Limits

The examples below execute the methodology and equations described in Section 2.1.9.1 for determining allowable fuel assemble decay heat, burnup, and cooling time.

Example 1

In this example a demonstration of the use of burnup versus cooling time tables for regionalized fuel loading is provided. In this example it will be assumed that the MPC-32 is being loaded with array/class 16x16A fuel in a regionalized loading pattern.

Step 1: Pick a value of X between 1 and 6. For this example X will be 2.8.

Step 2: Calculate $q_{Region\ 2}$ using equation 2.1.9.1:

$$q_{Region\ 2} = (2 \times 38)/[(1 + (2.8)^{0.15}) \times ((12 \times 2.8) + 20)] = 0.6543\ kW^{\dagger}$$

Step 3: Calculate $q_{Region\ 1}$ using equation 2.1.9.2:

$$q_{Region\ 1} = X \times q_{Region\ 2} = 2.8 \times 0.6543 = 1.83204\ kW$$

† *This result is arbitrarily rounded to four decimal places.*

Step 4: Develop a burnup versus cooling time table. Since this table is enrichment dependent, it is permitted and advisable to create multiple tables for different enrichments. In this example, two enrichments will be used: 3.1 and 4.185. Tables 12.2.1 and 12.2.2 show the burnup versus cooling time tables calculated for these enrichments for Region 1 and Region 2 using Equation 2.1.9.3.

Table 12.2.3 provides three hypothetical fuel assemblies in the 16x16A array/class that will be evaluated for acceptability for loading in the MPC-32 example above. The decay heat values in Table 12.2.3 would be calculated by the users. Other information would be taken from the fuel assembly and reactor operating records.

Fuel Assembly Number 1 is not acceptable for storage because its enrichment is lower than that used to determine the allowable burnups in Table 12.2.1 and 12.1.2. The solution is to develop another table using an enrichment of 3.0 wt.% ^{235}U or less to determine this fuel assembly's suitability for loading in this MPC-32.

Fuel Assembly Number 2 is not acceptable for loading unless a unique maximum allowable burnup for a cooling time of 3.3 years is calculated by linear interpolation between the values in Table 12.2.1 for 3 years and 4 years of cooling. Linear interpolation yields a maximum burnup of 43,597 MWD/MTU (rounded down from 43,597.9), making Fuel Assembly Number 2 acceptable for loading only in Region 1 due to decay heat limitations.

Fuel Assembly Number 3 is acceptable for loading based on the higher allowable burnups in Table 12.2.2, which were calculated using a higher minimum enrichment that those in Table 12.2.1, which is still below the actual initial enrichment of Fuel Assembly Number 3. Due to its relatively low decay heat, Fuel Assembly Number 3 may be stored in Region 1 or Region 2.

Example 2

In this example, each fuel assembly in Table 12.2.3 will be evaluated to determine whether it may be stored in the same hypothetical MPC-32 in a regionalized storage pattern. Assuming the same value 'X', the same maximum fuel assembly decay heats are calculated. Equation 2.1.9.3 is executed for each fuel assembly using its exact initial enrichment to determine its maximum allowable burnup. Linear interpolation will be used to further refine the maximum allowable burnup value between cooling times, if necessary.

Fuel Assembly Number 1: The calculated allowable burnup for 3.0 wt.% ^{235}U and a decay heat value of 1.83204 kW ($q_{region1}$) is 52,637 MWD/MTU at 4 years minimum cooling. Therefore the assembly is acceptable for storage in Region 1. Its decay heat is too high for loading in Region 2.

Fuel Assembly Number 2: The calculated allowable burnup for 3.2 wt.% ^{235}U and a decay heat value of 1.83204 kW ($q_{region1}$) is 39,844 MWD/MTU for 3 years cooling and 53,195 MWD/MTU for 4 years cooling. Linearly interpolating between these values for a cooling time of 3.3 years yields a maximum allowable burnup of 43,849 MWD/MTU and, therefore, the assembly is

acceptable for storage in Region 1. This fuel assembly's decay heat is also too high for loading in Region 2.

Fuel Assembly Number 3: The calculated allowable maximum burnup for 4.3 wt.% ^{235}U and a decay value of 0.6543 ($q_{region2}$) is 48,891 MWD/MTU for 18 years cooling. Therefore, the assembly is acceptable for storage in Region 2. This fuel assembly would also be acceptable for loading in Region 1 (this conclusion is inferred, but not demonstrated).

Table 12.2.1

EXAMPLE BURNUP VERSUS COOLING TIME LIMITS FOR REGIONALIZED LOADING (MPC-32, Array/Class 16x16A, X = 2.8, and Enrichment = 3.1 wt.% ^{235}U)

MINIMUM COOLING TIME (years)	MAXIMUM ALLOWABLE BURNUP IN REGION 1 (MWD/MTU)	MAXIMUM ALLOWABLE BURNUP IN REGION 2 (MWD/MTU)
≥3	39604	13568
≥4	52917	21040
≥5	61912	27028
≥6	68200	31446
≥7	68200	34520
≥8	68200	36814
≥9	68200	38588
≥10	68200	40059
≥11	68200	41280
≥12	68200	42375
≥13	68200	43349
≥14	68200	44263
≥15	68200	45144
≥16	68200	45991
≥17	68200	46809
≥18	68200	47628
≥19	68200	48433
≥20	68200	49247

Table 12.2.2

EXAMPLE BURNUP VERSUS COOLING TIME LIMITS FOR REGIONALIZED LOADING (MPC-32, Array/Class 16x16A, X = 2.8, and Enrichment =4.185 wt.% ^{235}U)

MINIMUM COOLING TIME (years)	MAXIMUM ALLOWABLE BURNUP IN REGION 1 (MWD/MTU)	MAXIMUM ALLOWABLE BURNUP IN REGION 2 (MWD/MTU)
≥3	42060	14019
≥4	55800	21814
≥5	65028	27883
≥6	68200	32304
≥7	68200	35386
≥8	68200	37695
≥9	68200	39482
≥10	68200	40989
≥11	68200	42247
≥12	68200	43365
≥13	68200	44377
≥14	68200	45326
≥15	68200	46237
≥16	68200	47115
≥17	68200	47947
≥18	68200	48796
≥19	68200	49629
≥20	68200	50466

Table 12.2.3

SAMPLE FUEL ASSEMBLIES TO DETERMINE ACCEPTABILITY FOR STORAGE
(Array/Class 16x16A)

ASSEMBLY NUMBER	ENRICHMENT (wt. % ^{235}U)	BURNUP (MWD/MTU)	COOLING TIME (years)	DECAY HEAT (kW)
1	3.0	37100	4.7	1.01
2	3.2	40250	3.3	1.75
3	4.3	41976	18.2	0.4

BASES TABLE OF CONTENTS

3.0 LIMITING CONDITION FOR OPERATION (LCO) APPLICABILITY ..... B 3.0-1
3.0 SURVEILLANCE REQUIREMENT (SR) APPLICABILITY ..... B 3.0-5

3.1 SFSC INTEGRITY ..... B 3.1.1-1
3.1.1 Multi-Purpose Canister (MPC) .....B 3.1.1-1
3.1.2 SFSC Heat Removal System ..... B 3.1.2-1
3.1.3 Fuel Cool-Down ..... B 3.1.3-1

3.2 SFSC RADIATION PROTECTION ..... B 3.2.1-1
3.2.1 ~~TRANSFER CASK Average Surface Dose Rates~~*Deleted* .B 3.2.1-1
3.2.2 TRANSFER CASK Surface Contamination ..... B 3.2.2-1
3.2.3 ~~OVERPACK Average Surface Dose Rates~~*Deleted* ..... B 3.2.3-1

3.3 SFSC CRITICALITY CONTROL .....B 3.3.1-1
Boron Concentration .....B 3.3.1-1

B 3.1 SFSC Integrity

B 3.1.1 Multi-Purpose Canister (MPC)

BASES

BACKGROUND

A TRANSFER CASK with an empty MPC is placed in the spent fuel pool and loaded with fuel assemblies meeting the requirements of the CoC. A lid is then placed on the MPC. The TRANSFER CASK and MPC are raised to the top of the spent fuel pool surface. The TRANSFER CASK and MPC are then moved into the cask preparation area where ~~dose rates are measured and~~ the MPC lid is welded to the MPC shell and the welds are inspected and tested. The water is drained from the MPC cavity and ~~moisture removal~~*drying* is performed. The MPC cavity is backfilled with helium. ~~Additional dose rates are measured and~~*Then,* the MPC vent and drain *port* cover plates and closure ring are installed and welded. Inspections are performed on the welds. ~~TRANSFER CASK bottom pool lid is replaced with the transfer lid to allow eventual transfer of the MPC into the OVERPACK.~~

MPC cavity moisture removal using vacuum drying or forced helium ~~recirculation~~ *dehydration* is performed to remove residual moisture from the MPC ~~fuel~~ cavity *space* after the MPC has been drained of water. If vacuum drying is used, any water that has not drained from the fuel cavity evaporates from the fuel cavity due to the vacuum. This is aided by the temperature increase due to the decay heat of the fuel and by the heat added to the MPC from the optional warming pad, if used.

If *forced* helium *dehydration*~~recirculation~~ is used, the dry gas introduced to the MPC cavity through the vent or drain port absorbs the residual moisture in the MPC. This humidified gas exits the MPC via the other port and the absorbed water is removed through condensation and/or mechanical drying. The dried helium is then forced back to the MPC until the temperature acceptance limit is met.

(continued)

BASES

BACKGROUND (continued)

After the completion of ~~moisture removal~~*drying*, the MPC cavity is backfilled with helium meeting the requirements of the CoC.

Backfilling of the MPC fuel cavity with helium promotes gaseous heat dissipation and the inert atmosphere protects the fuel cladding. ~~Providing a~~*Backfilling the MPC with* helium ~~pressure~~ in the required ~~range at room temperature (70°F)~~ *quantity*~~,~~ eliminates air inleakage over the life of the MPC because the cavity pressure rises due to heat up of the confined gas by the fuel decay heat during storage. ~~Providing helium in the required density range accomplishes the same function.~~

In-leakage of air could be harmful to the fuel. Prior to moving the SFSC to the storage pad, the MPC helium leak rate is determined to ensure that the fuel is confined.

APPLICABLE SAFETY ANALYSIS

The confinement of radioactivity during the storage of spent fuel in the MPC is ensured by the multiple confinement boundaries and systems. The barriers relied on are the fuel pellet matrix, the metallic fuel cladding tubes in which the fuel pellets are contained, and the MPC in which the fuel assemblies are stored. Long-term integrity of the fuel and cladding depend on storage in an inert atmosphere. This is accomplished by removing water from the MPC and backfilling the cavity with an inert gas. The thermal analyses of the MPC assume that the MPC cavity is filled with dry helium of a minimum quantity to ensure the assumptions used for convection heat transfer are preserved. Keeping the backfill pressure below the maximum value preserves the initial condition assumptions made in the MPC overpressurization evaluation.

(continued)

BASES (continued)

LCO

A dry, helium filled and sealed MPC establishes an inert heat removal environment necessary to ensure the integrity of the multiple confinement boundaries. Moreover, it also ensures that there will be no air in-leakage into the MPC cavity that could damage the fuel cladding over the storage period.

APPLICABILITY

The dry, sealed and inert atmosphere is required to be in place during TRANSPORT OPERATIONS and STORAGE OPERATIONS to ensure both the confinement barriers and heat removal mechanisms are in place during these operating periods. These conditions are not required during LOADING OPERATIONS or UNLOADING OPERATIONS as these conditions are being established or removed, respectively during these periods in support of other activities being performed with the stored fuel.

ACTIONS

A note has been added to the ACTIONS which states that, for this LCO, separate Condition entry is allowed for each MPC. This is acceptable since the Required Actions for each Condition provide appropriate compensatory measures for each MPC not meeting the LCO. Subsequent MPCs that do not meet the LCO are governed by subsequent Condition entry and application of associated Required Actions.

A.1

If the cavity vacuum drying pressure or demoisturizer exit gas temperature limit has been determined not to be met during TRANSPORT OPERATIONS or STORAGE OPERATIONS, an engineering evaluation is necessary to determine the potential quantity of moisture left within the MPC cavity. Since moisture remaining in the cavity during these modes of operation may represent a long-term degradation concern, immediate action is not necessary. The Completion Time is sufficient to complete the engineering evaluation commensurate with the safety significance of the CONDITION.

(continued)

BASES

ACTIONS
(continued)

A.2

Once the quantity of moisture potentially left in the MPC cavity is determined, a corrective action plan shall be developed and actions initiated to the extent necessary to return the MPC to an analyzed condition. Since the quantity of moisture estimated under Required Action A.1 can range over a broad scale, different recovery strategies may be necessary. Since moisture remaining in the cavity during these modes of operation may represent a long-term degradation concern, immediate action is not necessary. The Completion Time is sufficient to develop and initiate the corrective actions commensurate with the safety significance of the CONDITION.

B.1

If the helium backfill *quantity*~~density or pressure~~ limit has been determined not to be met during TRANSPORT OPERATIONS or STORAGE OPERATIONS, an engineering evaluation is necessary to determine the quantity of helium within the MPC cavity. Since too much or too little helium in the MPC during these modes represents a potential overpressure or heat removal degradation concern, an engineering evaluation shall be performed in a timely manner. The Completion Time is sufficient to complete the engineering evaluation commensurate with the safety significance of the CONDITION.

(continued)

BASES

ACTIONS
(continued)

B.2

Once the quantity of helium in the MPC cavity is determined, a corrective action plan shall be developed and initiated to the extent necessary to return the MPC to an analyzed condition. Since the quantity of helium estimated under Required Action B.1 can range over a broad scale, different recovery strategies may be necessary. Since elevated or reduced helium quantities existing in the MPC cavity represent a potential overpressure or heat removal degradation concern, corrective actions should be developed and implemented in a timely manner. The Completion Time is sufficient to develop and initiate the corrective actions commensurate with the safety significance of the CONDITION.

~~C.1~~

~~If the helium leak rate limit has been determined not to be met during TRANSPORT OPERATIONS or STORAGE OPERATIONS, an engineering evaluation is necessary to determine the impact of increased helium leak rate on heat removal and off-site dose. Since the HI-STORM OVERPACK is a ventilated system, any leakage from the MPC is transported directly to the environment. Since an increased helium leak rate represents a potential challenge to MPC heat removal and the off-site doses calculated in the FSAR confinement analyses, reasonably rapid action is warranted. The Completion Time is sufficient to complete the engineering evaluation commensurate with the safety significance of the CONDITION.~~

(continued)

BASES

ACTIONS (continued)

~~C.2~~

~~Once the cause and consequences of the elevated leak rate from the MPC are determined, a corrective action plan shall be developed and initiated to the extent necessary to return the MPC to an analyzed condition. Since the recovery mechanisms can range over a broad scale based on the evaluation performed under Required Action C.1, different recovery strategies may be necessary. Since an elevated helium leak rate represents a challenge to heat removal rates and off-site doses, reasonably rapid action is required. The Completion Time is sufficient to develop and initiate the corrective actions commensurate with the safety significance of the CONDITION.~~

~~D~~C.1

If the MPC fuel cavity cannot be successfully returned to a safe, analyzed condition, the fuel must be placed in a safe condition in the spent fuel pool. The Completion Time is reasonable based on the time required to replace the transfer lid with the pool lid *(if required)*, perform fuel cooldown operations *(if required)*, re-flood the MPC, cut the MPC lid welds, move the TRANSFER CASK into the spent fuel pool, remove the MPC lid, and remove the spent fuel assemblies in an orderly manner and without challenging personnel.

SURVEILLANCE REQUIREMENTS

SR 3.1.1.1~~,~~*and* SR 3.1.1.2~~, and SR 3.1.1.3~~

~~SR 3.1.1.1 is modified by a note that states, in addition to the requirements of SR 3.1.1.1 for high burnup fuel, MPCs with heat loads in excess of a certain value shall be dried using the helium recirculation method. The basis for this note is that, if vacuum drying were used for higher heat load MPCs, it would need to be completed in a relatively short period of time to avoid exceeding the short term peak fuel cladding temperature limit. Applying a time limit that is too restrictive could inhibit the ability to dry the MPC in a normal time frame. The helium recirculation method of moisture removal continuously cools the fuel while removing moisture, thereby eliminating the need to establish a time limit, allowing completion of the moisture removal process in a deliberate, controlled manner.~~

BASES

SURVEILLANCE REQUIREMENTS

SR 3.1.1.1, *and* SR 3.1.1.2~~, and SR 3.1.1.3~~ (continued)

The long-term integrity of the stored fuel is dependent on storage in a dry, inert environment. For moderate burnup fuel cavity dryness may be demonstrated either by evacuating the cavity to a very low absolute pressure and verifying that the pressure is held over a specified period of time or by recirculating dry helium through the MPC cavity to absorb moisture until the *gas temperature or dew point at the specified location* ~~demoisturizer exit temperature~~ reaches and remains below the acceptance limit for the specified time period. A low vacuum pressure or a demoisturizer exit temperature meeting the acceptance limit is an indication that the cavity is dry. For high burnup fuel *and high decay heat load MPCs*, the forced helium ~~recirculation~~ *dehydration* method of moisture removal must be used to provide necessary cooling of the fuel during drying operations. Cooling provided by normal operation of the forced helium dehydration system ensures that the fuel cladding temperature remains below the applicable limits since forced recirculation of helium provides more effective heat transfer than that which occurs during normal storage operations.

Table 3-1 of Appendix A to the CoC provides the appropriate requirements for drying the MPC cavity based on the burnup class of the fuel (moderate or high), the decay heat load of the MPC, and the applicable short-term temperature limit. The temperature limits and associated cladding hoop stress calculation requirements are consistent with the guidance in NRC Interim Staff Guidance (ISG) Document 11.

Having the proper *quantity of* helium *in the MPC*~~backfill density or pressure~~ ensures adequate heat transfer from the fuel to the fuel basket and surrounding structure of the MPC *and precludes any overpressure event from challenging the normal, off-normal, or accident design pressure of the MPC.*

~~Meeting the helium leak rate limit ensures there is adequate helium in the MPC for long term storage and the leak rate assumed in the confinement analyses remains bounding for off-site dose.~~

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.1.1.1~~,~~ *and* SR 3.1.1.2~~, and SR 3.1.1.3~~ (continued)

~~The leakage rate acceptance limit is specified in units of atm-cc/sec. This is a mass-like leakage rate as specified in ANSI N14.5 (1997). This is defined as the rate of change of the pressure-volume product of the leaking fluid at test conditions. This allows the leakage rate as measured by a mass spectrometer leak detector (MSLD) to be compared directly to the acceptance limit without the need for unit conversion from test conditions to standard, or reference conditions.~~

~~All three~~*Both* of these surveillances must be successfully performed once, prior to TRANSPORT OPERATIONS to ensure that the conditions are established for SFSC storage which preserve the analysis basis supporting the cask design.

REFERENCES

1. FSAR Sections 1.2, 4.4, 4.5, 7.2, 7.3 and 8.1
2. *Interim Staff Guidance Document 11*
3. *Interim Staff Guidance Document 18*

B 3.1 SFSC Integrity

B 3.1.2 SFSC Heat Removal System

BASES

BACKGROUND

The SFSC Heat Removal System is a passive, air-cooled, convective heat transfer system ~~which~~ *that* ensures heat from the MPC canister is transferred to the environs by the chimney effect. Relatively cool air is drawn into the annulus between the OVERPACK and the MPC through the ~~four~~ inlet air ducts at the bottom of the OVERPACK. The MPC transfers its heat from the canister surface to the air via natural convection. The buoyancy created by the heating of the air creates a chimney effect and the air is forced back into the environs through the ~~four~~ outlet air ducts at the top of the OVERPACK.

APPLICABLE SAFETY ANALYSIS

The thermal analyses of the SFSC take credit for the decay heat from the spent fuel assemblies being ultimately transferred to the ambient environment surrounding the OVERPACK. Transfer of heat away from the fuel assemblies ensures that the fuel cladding and other SFSC component temperatures do not exceed applicable limits. Under normal storage conditions, the ~~four~~ inlet and ~~four~~ outlet air ducts are unobstructed and full air flow (i.e., maximum heat transfer for the given ambient temperature) occurs.

Analyses have been performed for the complete obstruction of ~~two~~*half*, ~~three,~~ and ~~four~~*all* inlet air ducts. Blockage of ~~two~~ *half of the* inlet air ducts reduces air flow through the OVERPACK annulus and decreases heat transfer from the MPC. Under this off-normal condition, *performed for design-basis heat load,* no SFSC components exceed the short term temperature limits.

~~Blockage of three inlet air ducts further reduces air flow through the OVERPACK annulus and decreases heat transfer from the MPC. Under this accident condition, no SFSC components exceed the short term temperature limits.~~

(continued)

BASES

APPLICABLE SAFETY ANALYSIS (continued)

The complete blockage of all ~~four~~ inlet air ducts stops normal air cooling of the MPC. The MPC will continue to radiate heat to the relatively cooler inner shell of the OVERPACK. With the loss of normal air cooling, the SFSC component temperatures will increase toward their respective short-term temperature limits. *The results of this event are dependent upon the decay heat load of the MPC. Therefore, two analyses were performed. The first analysis was performed assuming a decay heat load of 28.74 kW and* ~~N~~*n*one of the components reach their temperature limits over the 72-hour duration of the analyzed event. *The second analysis was performed for the MPC-68 at its design decay heat load of* ~~41.22~~*35.5 kW and all component temperatures remain below their respective short term temperature limits up to 24 hours after event initiation. The MPC-68 analysis provides a bounding case for all MPC models since it yields the highest component temperatures.* ~~Therefore, the limiting component is assumed to be the fuel cladding.~~

LCO

The SFSC Heat Removal System must be verified to be operable to preserve the assumptions of the thermal analyses. Operability of the heat removal system ensures that the decay heat generated by the stored fuel assemblies is transferred to the environs at a sufficient rate to maintain fuel cladding and other SFSC component temperatures within design limits.

The intent of this LCO is to address those occurrences of air duct blockage that can be reasonably anticipated to occur from time to time at the ISFSI (i.e., Design Event I and II class events per ANSI/ANS-57.9). These events are of the type where corrective actions can usually be accomplished within one 8-hour operating shift to restore the heat removal system to operable status (e.g., removal of loose debris).

(continued)

BASES

LCO (continued)

This LCO is not intended to address low frequency, unexpected Design Event III and IV class events such as design basis accidents and extreme environmental phenomena that could potentially block one or more of the air ducts for an extended period of time (i.e., longer than the total Completion Time of the LCO). This class of events is addressed site-specifically. ~~as required by Section 3.4.9 of Appendix B to the CoC.~~

APPLICABILITY

The LCO is applicable during STORAGE OPERATIONS. Once an OVERPACK containing an MPC loaded with spent fuel has been placed in storage, the heat removal system must be operable to ensure adequate ~~heat transfer~~*dissipation* of the decay heat ~~away~~ from the fuel assemblies.

ACTIONS

A note has been added to the ACTIONS which states that, for this LCO, separate Condition entry is allowed for each SFSC. This is acceptable since the Required Actions for each Condition provide appropriate compensatory measures for each SFSC not meeting the LCO. Subsequent SFSCs that don't meet the LCO are governed by subsequent Condition entry and application of associated Required Actions.

A.1

If the heat removal system has been determined to be inoperable, it must be restored to operable status within eight hours. Eight hours is a reasonable period of time (typically, one operating shift) to take action to remove the obstructions in the air flow path.

(continued)

BASES

ACTIONS (continued)

B.1

If the heat removal system cannot be restored to operable status within eight hours, the innermost portion of the OVERPACK concrete may experience elevated temperatures. Therefore, ~~Surveillance Requirement (SR) 3.2.3.1~~*dose rates* ~~is~~ *are* required to be ~~performed~~ *measured* to ~~determine~~ *verify* the effectiveness of the radiation shielding provided by the concrete. This ~~SR~~ *Action* must be performed immediately and repeated every twelve hours thereafter to provide timely and continued evaluation of ~~whether~~ the *effectiveness of the* concrete ~~is providing adequate~~ shielding. As necessary, the cask user shall provide additional radiation protection measures such as temporary shielding. The Completion Time is reasonable considering the expected slow rate of deterioration, if any, of the concrete under elevated temperatures.

B.2.1

In addition to Required Action B.1, efforts must continue to restore cooling to the SFSC. Efforts must continue to restore the heat removal system to operable status by removing the air flow obstruction(s) unless optional Required Action B.2.2 is being implemented.

This Required Action must be complete in ~~48~~ *64* hours *if the decay heat load of the MPC is less than or equal to 28.74 kW or within 16 hours if the decay heat load of the MPC is greater than 28.74 kW. These Completion Times are consistent with the two thermal analyses of this event, which show that all component temperatures remain below their short-term temperature limits up to 72 or 24 hours after event initiation, for MPC heat loads of 28.74 and the highest authorized heat load of* ~~41.22~~ *38 kW, respectively. The 35.5 kW case analyzed for MPC-68 bounds the 38 kW case for the PWR MPCs.*

(continued)

BASES

ACTIONS

B.2.1 (continued)

The *two* Completion Times reflects *the 8 hours to complete Required Action A.1 and the appropriate balance of time consistent with the applicable analysis results. In each case, the event is assumed to begin at the time the SFSC heat removal system is declared inoperable. This is reasonable considering the low probability of all inlet or outlet ducts becoming simultaneously blocked by trash or debris.* a ~~conservative total time period without any cooling of 80 hours, assuming all of the inlet air ducts become blocked immediately after the last previous successful Surveillance. The results of the thermal analysis of this accident show that the fuel cladding temperature does not reach its short term temperature limit for more than 72 hours. It is also unlikely that an unforseen event could cause complete blockage of all four air inlet ducts immediately after the last successful Surveillance.~~

B.2.2

In lieu of implementing Required Action B.2.1, transfer of the MPC into a TRANSFER CASK will place the MPC in an analyzed condition and ensure adequate fuel cooling until actions to correct the heat removal system inoperability can be completed. Transfer of the MPC into a TRANSFER CASK removes the SFSC from the LCO Applicability since STORAGE OPERATIONS does not include times when the MPC resides in the TRANSFER CASK. *In this case, the requirements of CoC Appendix B, Section 3.4.10 apply.*

An engineering evaluation must be performed to determine if any concrete deterioration has occurred which prevents it from performing its design function. If the evaluation is successful and the air flow obstructions have been cleared, the OVERPACK heat removal system may be considered operable and the MPC transferred back into the OVERPACK. Compliance with LCO 3.1.2 is then restored. If the evaluation is unsuccessful, the user must transfer the MPC into a different, fully qualified OVERPACK to resume STORAGE OPERATIONS and restore compliance with LCO 3.1.2

(continued)

BASES

ACTIONS

B.2.2 (continued)

In lieu of performing the engineering evaluation, the user may opt to proceed directly to transferring the MPC into a different, fully qualified OVERPACK or place the TRANSFER CASK in the spent fuel pool and unload the MPC.

The Completion Times of ~~48~~ *64 and 16* hours reflects *the Completion Times from Required Action B.2.1 to ensure component temperatures remain below their short-term temperature limits for the respective decay heat loads.* a ~~conservative total time period without any cooling of 80 hours, assuming all of the inlet air ducts become blocked immediately after the last previous successful Surveillance. The results of the thermal analysis of this accident show that the fuel cladding temperature does not reach its short-term temperature limit for more than 72 hours. It is also unlikely that an unforeseen event could cause complete blockage of all four air inlet ducts immediately after the last successful Surveillance.~~

SURVEILLANCE REQUIREMENTS

SR 3.1.2.1

The long-term integrity of the stored fuel is dependent on the ability of the SFSC to reject heat from the MPC to the environment. There are two options for implementing SR 3.1.2.1, either of which is acceptable for demonstrating that the heat removal system is OPERABLE.

Visual observation that all four inlet and outlet air ducts are unobstructed ensures that air flow past the MPC is occurring and heat transfer is taking place. Complete blockage of any one or more inlet or outlet air ducts renders the heat removal system inoperable and this LCO not met. Partial blockage of one or more inlet or outlet air ducts does not constitute inoperability of the heat removal system. However, corrective actions should be taken promptly to remove the obstruction and restore full flow through the affected duct(s).

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.1.2.1 (continued)

As an alternative, for OVERPACKs with air temperature monitoring instrumentation installed in the outlet air ducts, the temperature rise between ambient and the OVERPACK air outlet may be monitored to verify operability of the heat removal system. Blocked inlet or outlet air ducts will reduce air flow and increase the temperature rise experienced by the air as it removes heat from the MPC. Based on the analyses, provided the air temperature rise is less than the limits stated in the SR, adequate air flow and, therefore, adequate heat transfer is occurring to provide assurance of long term fuel cladding integrity. The reference ambient temperature used to perform this Surveillance shall be measured at the ISFSI facility.

The Frequency of 24 hours is reasonable based on the time necessary for SFSC components to heat up to unacceptable temperatures assuming design basis heat loads, and allowing for corrective actions to take place upon discovery of blockage of air ducts.

REFERENCES

1. FSAR Chapter 4
2. FSAR Sections 11.2.13 and 11.2.14
3. ANSI/ANS 57.9-1992

B 3.1 SFSC INTEGRITY

B 3.1.3 Fuel Cool-Down

BASES

BACKGROUND

In the event that an MPC must be unloaded, the TRANSFER CASK with its enclosed MPC is returned to the cask preparation area to begin the process of fuel unloading. The MPC closure ring, and vent and drain port cover plates are removed. The MPC gas is sampled to determine the integrity of the spent fuel cladding. The *bulk helium temperature in the* MPC *cavity* is *ensured to be less than or equal to 200°F. This is accomplished via direct measurement of the MPC gas exit temperature or any other appropriate means based on a thermal evaluation of the particular MPC to be unloaded, considering its contents and the duration of time the MPC has been loaded. It is possible that the thermal evaluation may determine that the bulk gas temperature is already within the LCO limit due to low decay contents and/or an extended time since loading, in which case, no additional action is required.* ~~attached to the Cool-Down System. The Cool-Down System is a closed-loop forced ventilation gas cooling system that cools the fuel assemblies by cooling the surrounding helium gas.~~

After ensuring the MPC cavity bulk helium temperature meets the LCO limit, ~~Following fuel cool-down,~~ the MPC is then re-flooded with water and the MPC lid weld is removed leaving the MPC lid in place. The transfer cask and MPC are placed in the spent fuel pool and the MPC lid is removed. The fuel assemblies are removed from the MPC and the MPC and transfer cask are removed from the spent fuel pool and decontaminated.

Ensuring that ~~Reducing~~ the *bulk helium temperature is less than the LCO limit* ~~fuel cladding temperatures~~ significantly reduces the temperature gradients across the *fuel* cladding thus minimizing thermally-induced stresses on the cladding during MPC re-flooding. Reducing the MPC internal temperatures eliminates the risk of high MPC pressure due to sudden generation of *large* steam *quantities* during re-flooding. *The LCO limit of 200°F for bulk helium temperature eliminates the potential for gross steam generation during re-flooding.*

(continued)

BASES

APPLICABLE SAFETY ANALYSIS

The confinement of radioactivity during the storage of spent fuel in the MPC is ensured by the multiple confinement boundaries and systems. The barriers relied on are the fuel pellet matrix, the metallic fuel cladding tubes in which the fuel pellets are contained, and the MPC in which the fuel assemblies are stored. Long-term integrity of the fuel and cladding depend on minimizing thermally-induced stresses to the cladding.

This is accomplished during the unloading operations by lowering the MPC ~~internal~~ *cavity bulk helium* temperatures prior to MPC re-flooding. The ~~I~~integrity of the MPC depends on maintaining the internal cavity pressures within design limits. This is accomplished by reducing the MPC internal temperatures such that there is no sudden formation of *large quantities of* steam during MPC re-flooding. (Ref. 1).

LCO

~~Monitoring~~ *Determining* the ~~circulating~~ MPC ~~gas exit~~*cavity bulk helium* temperature *prior to re-flooding* ensures that there will be no large thermal gradient across the fuel assembly cladding during re-flooding which could be potentially harmful to the cladding. The temperature limit specified in the LCO was selected to ensure that the MPC *cavity bulk helium temperature is sufficiently low to preclude high thermal stresses in the fuel cladding during* ~~gas exit temperature will closely match the desired fuel cladding temperature prior to~~ re-flooding *of* the MPC. The temperature was selected to be lower than the boiling temperature of water with an additional margin.

For the purposes of this LCO, "bulk helium temperature" is defined as the spatial average of the helium temperature in the MPC cavity. The bulk helium temperature will be between the highest and lowest fuel cladding temperature present in the basket.

(continued)

BASES

APPLICABILITY

The MPC *cavity bulk* helium ~~gas exit~~ temperature is measured during UNLOADING OPERATIONS after the transfer cask and integral MPC are back in the FUEL BUILDING and are no longer suspended from, or secured in, the transporter. Therefore, the Fuel Cool-Down LCO does not apply during TRANSPORT OPERATIONS and STORAGE OPERATIONS.

A note has been added to the APPLICABILITY for LCO 3.1.3 which states that the Applicability is only applicable during wet UNLOADING OPERATIONS. This is acceptable since the intent of the LCO is to avoid uncontrolled MPC pressurization due to water flashing during re-flooding operations. This is not a concerning for dry UNLOADING OPERATIONS.

ACTIONS

A note has been added to the ACTIONS which states that, for this LCO, separate Condition entry is allowed for each MPC. This is acceptable since the Required Actions for each Condition provide appropriate compensatory measures for each MPC not meeting the LCO. Subsequent MPCs that do not meet the LCO are governed by subsequent Condition entry and application of associated Required Actions.

A.1

If the MPC *cavity bulk* helium ~~gas exit~~ temperature limit is not met, actions must be taken to restore the parameters to within the limits before re-flooding the MPC. Failure to successfully complete fuel cool-down could have several causes, such as failure of the cool down system, inadequate cool down, or clogging of the piping lines. The Completion Time is sufficient to determine and correct most failure mechanisms and proceeding with activities to flood the MPC cavity with water are prohibited.

(continued)

BASES

ACTIONS (continued)

A.2

If the LCO is not met, in addition to performing Required Action A.1 to restore the gas temperature to within the limit, the user must ensure that the proper conditions exist for the transfer of heat from the MPC to the surrounding environs to ensure the fuel cladding remains below the short term temperature limit. ~~If the TRANSFER CASK is located in a relatively open area such as a typical refuel floor,~~ *In this case the limits specified in Section 3.4.10 of Appendix B to the HI-STORM CoC apply.* ~~no additional actions are necessary.~~

~~However, if the TRANSFER CASK is located in a structure such as a decontamination pit or fuel vault, additional actions may be necessary depending on the heat load of the stored fuel.~~

~~Three acceptable options for ensuring adequate heat transfer for a TRANSFER CASK located in a pit or vault are provided below., based on an MPC loaded with fuel assemblies with design basis heat load in every storage location. Users may develop other alternatives. on a site-specific basis, considering actual fuel loading and decay heat generation.~~

(continued)

BASES

ACTIONS

A.2 (continued)

1. ~~Ensure the annulus between the MPC and the TRANSFER CASK is filled with water. This places the system in a heat removal configuration which is bounded by the FSAR thermal evaluation of the system considering a vacuum in the MPC. The system is open to the ambient environment which limits the temperature of the ultimate heat sink (the water in the annulus) and, therefore, the MPC shell to 212° F.~~

2. ~~Remove the TRANSFER CASK from the pit or vault and place it in an open area such as the refuel floor with a reasonable amount of clearance around the cask and not near a significant source of heat.~~

Immediately is an appropriate Completion Time because it requires action to be initiated promptly and completed without delay, but does not establish any particular fixed time limit for completing the action. This offers the flexibility necessary for users to plan and implement any necessary work activities commensurate with the safety significance of the condition, which is governed by the MPC heat load.

(continued)

BASES

ACTIONS

A.2 (continued)

~~Twenty-two (22) hours is an acceptable time frame to allow for completion of Required Action A.2 based on a thermal evaluation of a TRANSFER CASK located in a pit or vault. In such a configuration, passive cooling mechanisms will be largely diminished. Eliminating 90% of the passive cooling mechanisms with the cask emplaced in the vault, the thermal inertia of the cask (approximately 20,000 Btu/°F) will limit the rate of temperature rise with design basis maximum heat load to approximately 4.5 degrees F per hour. Thus, the fuel cladding temperature rise in 22 hours will be less than 100°F. Large short term temperature margins exist to preclude any cladding integrity concerns under this temperature rise.~~

SURVEILLANCE REQUIREMENTS

SR 3.1.3.1

The long-term integrity of the stored fuel is dependent on the material condition of the fuel assembly cladding. By minimizing thermally-induced stresses across the cladding the integrity of the fuel assembly cladding is maintained. The integrity of the MPC is dependent on controlling the internal MPC pressure. By controlling the MPC internal temperature prior to re-flooding the MPC there is minimal formation of steam during MPC re-flooding.

The MPC *cavity bulk* helium ~~exit gas~~ temperature limit ensures that there will be no large thermal gradients across the fuel assembly cladding during MPC re-flooding and ~~no~~*minimal* formation of steam which could potentially overpressurize the MPC.

(continued)

BASES

SURVEILLANCE REQUIREMENTS

SR 3.1.3.1 (continued)

The SR is met in one of two ways. The temperature of the gas exiting the MPC may be measured directly. Alternatively, a thermal evaluation may be performed, consistent with the methodology in the HI-STORM FSAR, to determine the MPC bulk helium temperature in the canister designated for unloading. This evaluation may consider the particular characteristics of the MPC, such as fuel cooling time, presence of NON-FUEL HARDWARE, and ambient conditions in determining the bulk helium temperature. If the MPC cavity bulk helium temperature LCO is shown to be met by this evaluation, no further actions are required and MPC unloading may proceed. If the LCO is shown not to be met by the thermal evaluation, appropriate means shall be used to cool the MPC cavity until the LCO is met (via direct measurement of the helium gas exit temperature or by an evaluation that includes the cooling process). When the LCO is met, unloading may proceed.

The LCO must be met ~~Fuel cool-down must be performed successfully~~ on each SFSC before the initiation of MPC re-flooding operations to ensure the design and analysis basis are preserved.

REFERENCES

1. FSAR, Sections ~~4.4.1, 4.5.1.1.4,~~*4.4*, *4.5* and 8.3.2.

B 3.2 ~~SFSC Radiation Protection~~Deleted

B 3.2.1 ~~TRANSFER CASK Average Surface Dose Rates~~Deleted

~~BASES~~

~~BACKGROUND~~	~~The regulations governing the operation of an ISFSI set limits on the control of occupational radiation exposure and radiation doses to the general public (Ref. 1). Occupational radiation exposure should be kept as low as reasonably achievable (ALARA) and within the limits of 10CFR Part 20. Radiation doses to the public are limited for both normal and accident conditions.~~
~~APPLICABLE SAFETY ANALYSIS~~	~~The TRANSFER CASK average surface dose rates are not an assumption in any accident analysis, but are used to ensure compliance with regulatory limits on occupational dose and dose to the public.~~
~~LCO~~	~~The limits on TRANSFER CASK average surface dose rates are based on the shielding analysis of the HI-STORM 100 System (Ref. 2). The limits were selected to minimize radiation exposure to the general public and maintain occupational dose ALARA to personnel working in the vicinity of the TRANSFER CASKs. The LCO requires specific locations for taking dose rate measurements to ensure the dose rates measured are indicative of the neutron shielding material=s effectiveness and not the steel channel members.~~
~~APPLICABILITY~~	~~The average TRANSFER CASK surface dose rates apply during TRANSPORT OPERATIONS. These limits ensure that the transfer cask average surface dose rates during TRANSPORT OPERATIONS, AND UNLOADING OPERATIONS are within the estimates contained in the HI-STORM 100 Topical Safety Analysis Report. Radiation doses during STORAGE OPERATIONS are verified for the OVERPACK under LCO 3.2.3 and monitored thereafter by the SFSC user in accordance with the plant-specific radiation protection program required by 10CFR72.212(b)(6).~~

~~(continued)~~

~~BASES (continued)~~

~~ACTIONS~~ ~~A note has been added to the ACTIONS which states that, for this LCO, separate Condition entry is allowed for each TRANSFER CASK. This is acceptable since the Required Actions for each Condition provide appropriate compensatory measures for each TRANSFER CASK not meeting the LCO. Subsequent TRANSFER CASKs that do not meet the LCO are governed by subsequent Condition entry and application of associated Required Actions.~~

~~A.1~~

~~If the TRANSFER CASK average surface dose rates are not within limits, it could be an indication that a fuel assembly was inadvertently loaded into the MPC that did not meet the Functional and Operating Limits in Section 2.0. Administrative verification of the MPC fuel loading, by means such as review of video recordings and records of the loaded fuel assembly serial numbers, can establish whether a mis-loaded fuel assembly is the cause of the out of limit condition. The Completion Time is based on the time required to perform such a verification.~~

~~A.2~~

~~If the TRANSFER CASK average surface dose rates are not within limits, and it is determined that the MPC was loaded with the correct fuel assemblies, an analysis may be performed. This analysis will determine if the OVERPACK, once located at the ISFSI, would result in the ISFSI offsite or occupational doses exceeding regulatory limits in 10 CFR Part 20 or 10 CFR Part 72. If it is determined that the out of limit average surface dose rates do not result in the regulatory limits being exceeded, TRANSPORT OPERATIONS may proceed.~~

~~B.1~~

~~If it is verified that unauthorized fuel was loaded or that the ISFSI offsite radiation protection requirements of 10 CFR Part 20 or 10 CFR Part 72 will not be met with the transfer cask average surface dose rates above the LCO limit, the fuel~~

~~(continued)~~

~~BASES~~

~~ACTIONS~~ ~~B.1 (continued)~~

~~assemblies must be placed in a safe condition in the spent fuel pool. The Completion Time is reasonable based on the time required to replace the transfer lid with the pool lid, perform fuel cooldown operations, re-flood the MPC, cut the MPC lid welds, move the TRANSFER CASK into the spent fuel pool, remove the MPC lid, and remove the spent fuel assemblies in an orderly manner and without challenging personnel.~~

~~SURVEILLANCE REQUIREMENTS~~ ~~SR 3.2.1.1~~

~~This SR ensures that the TRANSFER CASK average surface dose rates are within the LCO limits prior to TRANSPORT OPERATIONS. The surface dose rates are measured on the sides and the top of the TRANSFER CASK at locations described in the SR following standard industry practices for determining average dose rates for large containers. The SR requires specific locations for taking dose rate measurements to ensure the dose rates measured are indicative of the average value around the cask.~~

~~REFERENCES~~

~~1. 10 CFR Parts 20 and 72.~~
~~2. FSAR Sections 5.1 and 8.1.6.~~

B 3.2 ~~SFSC Radiation Protection~~Deleted

B 3.2.3 ~~OVERPACK Average Surface Dose Rates~~Deleted

~~BASES~~

~~BACKGROUND~~	~~The regulations governing the operation of an ISFSI set limits on the control of occupational radiation exposure and radiation doses to the general public (Ref. 1). Occupational radiation exposure should be kept as low as reasonably achievable (ALARA) and within the limits of 10CFR Part 20. Radiation doses to the public are limited for both normal and accident conditions.~~
~~APPLICABLE SAFETY ANALYSIS~~	~~The OVERPACK average surface dose rates are not an assumption in any accident analysis, but are used to ensure compliance with regulatory limits on occupational dose and dose to the public.~~
~~LCO~~	~~The limits on OVERPACK average surface dose rates are based on the shielding analysis of the HI-STORM 100 System (Ref. 2). The limits were selected to minimize radiation exposure to the general public and maintain occupational dose ALARA to personnel working in the vicinity of the SFSCs.~~
~~APPLICABILITY~~	~~The average OVERPACK surface dose rates apply during TRANSPORT OPERATIONS and STORAGE OPERATIONS. These limits ensure that the OVERPACK average surface dose rates are within the estimates contained in the HI-STORM 100 Topical Safety Analysis Report. Radiation doses during STORAGE OPERATIONS are monitored for the OVERPACK by the SFSC user in accordance with the plant-specific radiation protection program required by 10CFR72.212(b)(6).~~

~~(continued)~~

~~BASES (continued)~~

~~ACTIONS~~ ~~A note has been added to the ACTIONS which states that, for this LCO, separate Condition entry is allowed for each SFSC. This is acceptable since the Required Actions for each Condition provide appropriate compensatory measures for each SFSC not meeting the LCO. Subsequent SFSCs that don't meet the LCO are governed by subsequent Condition entry and application of associated Required Actions.~~

~~A.1~~

~~If the OVERPACK average surface dose rates are not within limits, it could be an indication that a fuel assembly was inadvertently loaded into the MPC that did not meet the Functional and Operating Limits in Section 2.0. Administrative verification of the MPC fuel loading, by means such as review of video recordings and records of the loaded fuel assembly serial numbers, can establish whether a mis-loaded fuel assembly is the cause of the out of limit condition. The Completion Time is based on the time required to perform such a verification.~~

~~A.2~~

~~If the OVERPACK average surface dose rates are not within limits, and it is determined that the MPC was loaded with the correct fuel assemblies, an analysis may be performed. This analysis will determine if the OVERPACK, once located at the ISFSI, would result in the ISFSI offsite or occupational doses exceeding regulatory limits in 10 CFR Part 20 or 10 CFR Part 72. If it is determined that the out of limit average surface dose rates do not result in the regulatory limits being exceeded, STORAGE OPERATIONS may proceed.~~

~~B.1~~

~~If it is verified that the correct fuel was not loaded or that the ISFSI offsite radiation protection requirements of 10 CFR Part 20 or 10 CFR Part 72 will not be met with the OVERPACK average surface dose rates above the LCO limit, the fuel~~

~~(continued)~~

~~BASES~~

~~ACTIONS (continued)~~

~~assemblies must be placed in a safe condition in the spent fuel pool. The Completion Time is reasonable based on the time required to transfer the MPC back into the TRANSFER CASK, replace the transfer lid with the pool lid, perform fuel cooldown operations, re-flood the MPC, cut the MPC lid welds, move the SFSC into the spent fuel pool, remove the MPC lid, and remove the spent fuel assemblies in an orderly manner and without challenging personnel.~~

~~SURVEILLANCE REQUIREMENTS~~

~~SR 3.2.3.1~~

~~This SR ensures that the OVERPACK average surface dose rates are within the LCO limits within 24 hours of placing the OVERPACK in its designated storage location on the ISFSI. Surface dose rates are measured at the locations described in the SR following standard industry practices for determining average dose rates for large containers.~~

~~REFERENCES~~

~~1. 10 CFR Parts 20 and 72.~~
~~2. FSAR Sections 5.1 and 8.1.6.~~

B 3.3 SFSC Criticality Control

B 3.3.1 Boron Concentration

BASES

BACKGROUND

A TRANSFER CASK with an empty MPC is placed in the spent fuel pool and loaded with fuel assemblies meeting the requirements of the Certificate of Compliance. A lid is then placed on the MPC. The TRANSFER CASK and MPC are raised to the top of the spent fuel pool surface. The TRANSFER CASK and MPC are then moved into the cask preparation area where ~~dose rates are measured and~~ the MPC lid is welded to the MPC shell and the welds are inspected and tested. The water is drained from the MPC cavity and ~~vacuum~~ drying is performed. The MPC cavity is backfilled with helium. *Then,* ~~Additional dose rates are measured and~~ the MPC vent and drain cover plates and closure ring are installed and welded. Inspections are performed on the welds. The TRANSFER CASK bottom pool lid is replaced with the transfer lid to allow eventual transfer of the MPC into the OVERPACK.

For those MPCs containing PWR fuel assemblies of relatively high initial enrichment, credit is taken in the criticality analyses for boron in the water within the MPC. To preserve the analysis basis, users must verify that the boron concentration of the water in the MPC meets specified limits when there is fuel and water in the MPC. This may occur during LOADING OPERATIONS and UNLOADING OPERATIONS.

APPLICABLE SAFETY ANALYSIS

The spent nuclear fuel stored in the SFSC is required to remain subcritical ($k_{eff} < 0.95$) under all conditions of storage. The HI-STORM 100 SFSC is analyzed to stored a wide variety of spent nuclear fuel assembly types with differing initial enrichments. For all PWR fuel loaded in the MPC-32 *and MPC-32F*, and for relatively high enrichment PWR fuel loaded in the MPC-24, -24E, and -24EF, credit was taken in the criticality analyses for neutron poison in the form of soluble boron in the water within the MPC. Compliance with this LCO preserves the assumptions made in the criticality analyses regarding credit for soluble boron.

(continued)

BASES (continued)

LCO

Compliance with this LCO ensures that the stored fuel will remain subcritical with a $k_{eff} \leq 0.95$ while water is in the MPC. LCOs 3.3.1.a and 3.3.1.b provide the minimum concentration of soluble boron required in the MPC water for the MPC-24, and MPC-24E/24EF, respectively, *for MPCs containing all INTACT FUEL ASSEMBLIES.* The limits are applicable to the respective MPCs if one or more fuel assemblies to be loaded in the MPC had an initial enrichment of U-235 greater than the value in Table 2.1-2 of Appendix B to the CoC for loading with no soluble boron credit.

LCO 3.3.1.e provides the minimum concentration of soluble boron required in the MPC water for the MPC-24E and MPC-24EF containing at least one DAMAGED FUEL ASSEMBLY or one fuel assembly classified as FUEL DEBRIS.

LCO 3.3.1.~~c~~*f* provides the minimum ~~boron~~ concentration *of soluble boron* required in the MPC water for the MPC-32 *and MPC-32F based on the fuel assembly array/class and the classification of the fuel as a DAMAGED FUEL ASSEMBLY or FUEL DEBRIS.* ~~if one or more to fuel assemblies to be loaded had an initial enrichment less than or equal to 4.1 wt.% U-235. LCO 3.3.1.d provides the minimum boron concentration required in the MPC water for the MPC-32 if one or more to fuel assemblies to be loaded had an initial enrichment greater than 4.1 wt.% U-235.~~

All fuel assemblies loaded into the MPC-24, MPC-24E, MPC-24EF, ~~and~~ MPC-32, *and MPC-32F* are limited by analysis to maximum enrichments of 5.0 wt.% U-235.

The LCO also requires that the minimum soluble boron concentration for the most limiting fuel assembly array/class and classification to be stored in the same MPC be used. This means that the highest minimum soluble boron concentration limit for all fuel assemblies in the MPC applies in cases where fuel assembly array/classes and fuel classifications (intact vs. damaged) are mixed in the same MPC. The ensures the assumptions pertaining to soluble bom used in the criticality analyses are preserved.

(continued)

BASES

APPLICABILITY

The boron concentration LCO is applicable whenever an MPC-24, -24E, -24EF, ~~or~~ -32, *or -32F* has at least one PWR fuel assembly in a storage location and water in the MPC, For the MPC-24 and MPC-24E/24EF, when all fuel assemblies to be loaded have initial enrichments less than the limit for no soluble boron credit as provided in CoC Appendix B, Table 2.1-2, the boron concentration requirement is implicitly understood to be zero.

During LOADING OPERATIONS, the LCO is applicable immediately upon the loading of the first fuel assembly in the MPC. It remains applicable until the MPC is drained of water

During UNLOADING OPERATIONS, the LCO is applicable when the MPC is re-flooded with water after helium cooldown operations. Note that compliance with SR 3.0.4 assures that the water to be used to flood the MPC is of the correct boron concentration to ensure the LCO is upon entering the Applicability.

ACTIONS

A note has been added to the ACTIONS which states that, for this LCO, separate Condition entry is allowed for each MPC. This is acceptable since the Required Actions for each Condition provide appropriate compensatory measures for each MPC not meeting the LCO. Subsequent MPCs that do not meet the LCO are governed by subsequent Condition entry and application of associated Required Actions.

A.1 and A.2

Continuation of LOADING OPERATIONS, UNLOADING OPERATIONS or positive reactivity additions (including actions to reduce boron concentration) is contingent upon maintaining the SFSC in compliance with the LCO. If the boron concentration of water in the MPC is less than its limit, all activities LOADING OPERATIONS, UNLOADING OPERATIONS or positive reactivity additions must be suspended immediately.

(continued)

BASES

ACTIONS (continued)

A.3

In addition to immediately suspending LOADING OPERATIONS, UNLOADING OPERATIONS and positive reactivity additions, action to restore the concentration to within the limit specified in the LCO must be initiated immediately.

One means of complying with this action is to initiate boration of the affected MPC. In determining the required combination of boration flow rate and concentration, there is no unique design basis event that must be satisfied; only that boration be initiated without delay. In order to raise the boron concentration as quickly as possible, the operator should begin boration with the best source available for existing plant conditions.

Once boration is initiated, it must be continued until the boron concentration is restored. The restoration time depends on the amount of boron that must be injected to reach the required concentration.

SURVEILLANCE REQUIREMENTS (continued)

SR 3.3.1.1

The boron concentration in the MPC water must be verified to be within the applicable limit within four hours ~~of~~ *prior to* entering the Applicability of the LCO. For LOADING OPERATIONS, this means within four hours of loading the first fuel assembly into the cask.

For UNLOADING OPERATIONS, this means verifying the source of borated water to be used to re-flood the MPC within four hours of commencing re-flooding operations. This ensures that when the LCO is applicable (upon introducing water into the MPC), the LCO will be met.

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

Surveillance Requirement 3.3.1.1 is modified by a note which states that SR 3.3.1.1 is only required to be performed if the MPC is submerged in water or if water is to be added to, or recirculated through the MPC. This reflects the underlying premise of this SR which is to ensure, once the correct boron concentration is established, it need only be verified thereafter if the MPC is in a state where the concentration could be changed.

There is no need to re-verify the boron concentration of the water in the MPC after it is removed from the spent fuel pool unless water is to be added to, or recirculated through the MPC, because these are the only credible activities that could potentially change the boron concentration during this time. This note also prevents the interference of unnecessary sampling activities while lid closure welding and other MPC storage preparation activities are taking place in an elevated radiation area atop the MPC. Plant procedures should ensure that any water to be added to, or recirculated through the MPC is at a boron concentration greater than or equal to the minimum boron concentration specified in the LCO

REFERENCES

1. FSAR Chapter 6.

CHAPTER 13[†]: QUALITY ASSURANCE

13.0 ~~INTRODUCTION~~*QUALITY ASSURANCE PROGRAM*

13.0.1 Overview

This chapter provides a summary of the quality assurance program implemented for activities related to the design, qualification analyses, material procurement, fabrication, assembly, testing and use of structures, systems, and components of the HI-STORM 100 System and HI-TRAC transfer cask designated as important to safety.

~~Table 2.2.6 identifies the structures, systems and components (SSCs) of the HI-STORM 100 System and HI-TRAC transfer cask that are considered important to safety. Table 8.1.6 identifies the ancillary equipment needed for handling and loading operations that has been designated as important to safety.~~

Important-to-safety activities related to construction and deployment of the HI-STORM 100 System are controlled under the NRC-approved Holtec Quality Assurance Program. Revision 13 of the Holtec QA program manual (Reference [13.0.2]) was submitted to the NRC for review and approval on August 17, 2001 and approved by the NRC on September 25, 2001 (Reference [13.0.4]) under Docket 71-0784. The Holtec QA program satisfies the requirements of 10 CFR 72, Subpart G and 10 CFR 71, Subpart H. In accordance with 10 CFR 72.140(d), this previously-approved 10 CFR 71 QA program will be applied to spent fuel storage cask activities under 10 CFR 72. The additional recordkeeping requirements of 10 CFR 72.174 are addressed in the Holtec QA program manual and must also be complied with.

The Holtec QA program is implemented through a hierarchy of procedures and documentation, listed below.

1. *Holtec Quality Assurance Program Manual*
2. *Holtec Quality Assurance Procedures*
3. a. *Holtec Standard Procedures*
   b. *Holtec Project Procedures*

Quality activities performed by others on behalf of Holtec are governed by the supplier's quality assurance program or Holtec's QA program extended to the supplier. The type

[†] This chapter has been prepared in the format and section organization set forth in Regulatory Guide 3.61. However, the material content of this chapter also fulfills the ~~requirements~~ *intent* of NUREG-1536. Pagination and numbering of sections, figures, and tables are consistent with the convention set down in Chapter 1, Section 1.0, herein. Finally, all terms-of-art used in this chapter are consistent with the terminology of the glossary (Table 1.0.1) and component nomenclature of the Bill-of-Materials (Section 1.5).

and extent of Holtec QA control and oversight is specified in the procurement documents for the specific item or service being procured. The fundamental goal of the supplier oversight portion of Holtec's QA program is to provide assurance that activities performed in support of the supply of safety-significant items and services are performed correctly and in compliance with the procurement documents.

13.0.2 Graded Approach to Quality Assurance

For the HI-STORM 100 System, a graded approach to quality assurance is used by Holtec. This graded approach is controlled by Holtec Quality Assurance (QA) program documents as described in Section 13.0.1.

NUREG/CR-6407 [13.0.1] provides descriptions of quality categories A, B and C. Using the guidance in NUREG/CR-6407, Holtec International assigns a quality category to each individual, important-to-safety component of the HI-STORM 100 System and HI-TRAC transfer cask. The categories assigned to the cask components are identified in Table 2.2.6. Quality categories for ancillary equipment are provided in Table 8.1.6 on a generic basis. Quality categories for other equipment needed to deploy the HI-STORM 100 System at a licensee's ISFSI are defined on a case-specific basis considering the component's design function.

Activities affecting quality are defined by the purchaser's procurement contract for use of the HI-STORM 100 System at an independent spent fuel storage installation (ISFSI) under the general license provisions of 10CFR72, Subpart K. They may include any or all of the following: design, procurement, fabrication, handling, shipping, storing, cleaning, assembly, inspection, testing, operation, maintenance, repair and monitoring of HI-STORM 100 structures, systems, and components that are important to safety.

The quality assurance program described in the QA Program Manual fully complies with the requirements of 10CFR72 Subpart G and the intent of NUREG-1536 [13.0.3]. However, NUREG-1536 does not explicitly address incorporation of a QA program manual by reference. Therefore, invoking the NRC-approved QA program in this FSAR constitutes a literal deviation from NUREG-1536 and has accordingly been added to the list of deviations in Table 1.0.3. This deviation is acceptable since important-to-safety activities are implemented in accordance with the latest revision of the Holtec QA program manual and implementing procedures. Further, incorporating the QA Program Manual by reference in this FSAR avoids duplication of information between the implementing documents and the FSAR and any discrepancies that may arise from simultaneous maintenance to the two program descriptions governing the same activities.

13.1 ~~GRADED APPROACH TO QUALITY ASSURANCE~~

This section intentionally deleted.

~~For the HI-STORM 100 System and HI-TRAC transfer cask, a graded approach to quality is used by Holtec. This graded approach is controlled by Holtec Quality Assurance (QA) program documents.~~

~~NUREG/CR-6407 [13.1.1] provides descriptions of quality categories A, B and C. These descriptions are provided below.~~

~~Category A:~~ ~~Category A items include structures, systems, and components whose failure could directly result in a condition adversely affecting public health and safety. The failure of a single item could cause loss of primary containment leading to release of radioactive material, loss of shielding, or unsafe geometry compromising criticality control.~~

~~Category B:~~ ~~Category B items include structures, systems, and components whose failure or malfunction could indirectly result in a condition adversely affecting public health and safety. The failure of a Category B item, in conjunction with the failure of an additional item, could result in an unsafe condition.~~

~~Category C:~~ ~~Category C items include structures, systems, and components whose failure or malfunction would not significantly reduce the packaging effectiveness and would not be likely to create a situation adversely affecting public health and safety.~~

~~Using these descriptions along with the quality category assignments from NUREG/CR-6407 [13.1.1], Holtec International has assigned a quality category to each individual component of the HI-STORM 100 System and HI-TRAC transfer cask. The categories are identified in Table 2.2.6.~~

~~Activities affecting quality are defined by the purchaser's procurement contract for use of the HI-STORM 100 System on a site-specific independent spent fuel storage installation (ISFSI) under the general license provisions of 10CFR72, Subpart K. They may include any or all of the following: design, procurement, fabrication, handling, shipping, storing, cleaning, assembly, inspection, testing, operation, maintenance, repair and monitoring of HI-STORM 100 structures, systems, and components which are important to safety. Regardless of the provisions of the procurement contract, the quality requirements set forth in this document constitute the minimum set of acceptable bases. Activities performed in the course of the previous and ongoing work effort on HI-STORM 100 comply with Holtec International's quality assurance program. Holtec International's QA program was developed to meet Nuclear Regulatory Commission (NRC) requirements delineated in 10CFR50, Appendix B, and has been expanded to include provisions of 10CFR71, Subpart H and 10CFR72, Subpart G, for structures, systems, and components designated as~~

~~important to safety. A topical report [13.1.2] on the Holtec International QA program has been previously submitted to the NRC. Quality Assurance Program Approval for Radioactive Packages No. 0784 was issued by the NRC. This quality assurance program also applies to the design, material procurement, fabrication, inspection, testing, handling, and repair of the HI-STORM 100 System, including the HI-TRAC transfer cask.~~

~~The quality assurance program described in this chapter fully complies with the requirements of 10CFR72 Subpart G, and NUREG-1536 [13.1.3].~~

13.2 ~~PROJECT ORGANIZATION~~

This section is intentionally deleted.

~~The HI-STORM 100 System project has been established under Holtec International's project identification number 5014. This project has been designated as important to safety (ITS), which automatically mandates a rigorously formulated and carefully articulated project management system in accordance with the Holtec Quality Assurance Manual (HQAM). The first requirement of the HQAM is to identify a project team, and to prepare and approve a Project Plan. The HQAM mandates that all activities of an important to safety project be carried out in accordance with the Project Plan. Section 13.3 herein presents the essential elements of the HI-STORM 100 project programmatic quality requirements.~~

~~The HI-STORM 100 project team consists of a project manager, the licensing manager, the QA manager, and a team of technical specialists. A description of Holtec's organizational structure, functions, lines of responsibility, and levels of authority can be found in Holtec Quality Assurance documents.~~

13.3 QUALITY ASSURANCE PROGRAM

This section intentionally deleted.

~~13.3.1 Overview~~

~~Important to safety (ITS) work on the HI-STORM 100 project is performed by Holtec International in accordance with Holtec International's quality assurance program which is designed to satisfy the requirements imposed within 10CFR50 Appendix B, 10CFR71, Subpart H, and 10CFR72, Subpart G. The following provides a summary of Holtec International's quality assurance program implementation to comply with the applicable regulatory requirements.~~

~~13.3.2 Quality Assurance Program Documents~~

~~Holtec International's quality assurance program has three levels of controlling documents. The highest level, and overall controlling document, is the Holtec International Quality Assurance Manual (HQAM) which provides the requirements and commitments that Holtec International must follow during the course of any nuclear safety-related or important to safety project. The manual is organized into 18 sections that correspond to the eighteen QA program criteria cited in the above-referenced regulations.~~

~~The second level of quality assurance program controlling documents is the Holtec International Quality Procedures (HQPs). These procedures provide specific details on how Holtec International implements the requirements and commitments in the quality assurance manual..~~

~~Standard and project specific procedures comprise the third level of quality assurance program controlling documents. These procedures are used to control specific project activities and requirements which are not addressed within the Holtec International quality procedures. Examples of this would be a visual weld examination procedure, liquid penetrant examination procedure, or an in-process inspection procedure. These procedures are considered quality assurance records and are controlled in accordance with Holtec International's quality assurance program.~~

~~13.3.3 Quality Assurance Program Content~~

~~The requirements and commitments of Holtec International's quality assurance program as specified in the Holtec International quality assurance manual and corresponding quality procedures and project specific procedures (hereafter called quality assurance program documents) are summarized below. Each criterion is summarized separately.~~

~~1. Organization~~

~~Holtec International's quality assurance program documents define the quality assurance program related responsibilities of Holtec International personnel, as well as the breakdown of the organizational responsibilities within Holtec International. The Holtec International organization is detailed in the HQAM and HQP 1.0.~~

~~Holtec International's quality assurance program requires that the President of Holtec International review the status of the quality program on an annual basis. Furthermore, as part of Holtec International upper management's commitment to Holtec International's quality assurance program, a statement of policy authored by the President of Holtec International is contained in the quality assurance manual. This policy defines Holtec International's commitment to meeting the requirements of 10CFR50, Appendix B; 10CFR71, Subpart H; and 10CFR72, Subpart G, as applicable, on safety-related and important to safety projects and also delegates overall responsibility of quality program maintenance to the Quality Assurance Manager. The listing of Structures, Systems, and Components (SSC), defined as important to safety for the HI-STORM 100 System, is provided in Table 2.2.6 of this FSAR.~~

~~The Quality Assurance Manager is the person responsible for establishing and maintaining the QA Program. He reports to the Executive Vice President of Holtec International on all quality matters and has the authority and organizational freedom to enforce QA requirements, identify problem areas, recommend or provide solutions to QA problems, and verify the effectiveness of those solutions. As necessary, the Quality Assurance Manager can communicate directly to the President of Holtec International on quality-related issues. The minimum qualification requirements for the position of Quality Assurance Manager are contained in the Holtec QA program procedures. Regardless of the education and experience requirements, the QA manager shall be knowledgeable of the applicable codes and standards.~~

~~The Quality Assurance Manager has the following typical responsibilities:~~

~~a. Monitor quality issues and keep Management informed of significant conditions adverse to quality.~~

~~b. Initiate, recommend, or provide solutions and verify implementation of corrective actions to nonconforming conditions.~~

~~c. Control or stop further processing, delivery, or installation of a nonconforming item, deficiency, or unsatisfactory condition until proper dispositioning has occurred.~~

~~d. Maintain and control the HQAM, HQPs, and standard and project procedures.~~

~~e. Review contractual documents to assure inclusion of applicable quality assurance requirements.~~

~~f. Interface with clients and regulators during audits.~~

~~g. Schedule, perform, and/or oversee audits/surveillances of suppliers of quality related items and services to verify proper implementation of the quality assurance program.~~

~~h. Schedule, perform, and/or oversee audits of internal activities to verify compliance with the HQAM.~~

~~i. Approve Quality Procedures and Project Plans.~~

~~j. Perform periodic reviews of nonconformance reports to identify adverse quality trends for management review and assessment.~~

~~k. Coordinate activities to assess the adequacy and effectiveness of the QA program.~~

~~l. Schedule and conduct training and indoctrination of personnel performing activities affecting quality.~~

~~m. Maintain current qualifications/certifications for personnel performing quality related activities, as appropriate.~~

~~n. Maintain a current Approved Vendors List for vendors approved to provide quality related items/services.~~

~~o. Maintain a current list of approved computer programs.~~

~~Some of the above listed activities may be performed by personnel designated by the Quality Assurance Manager, although the Quality Assurance Manager retains overall responsibility for assuring proper implementation of the Quality Assurance Program.~~

~~Holtec International may contract with another organization to perform work on important to safety activities. The other organization could be a design agent, manufacturer, supplier, or subcontractor. Any organization performing functions affecting quality of important to safety work must have a QA position with the required authority and organizational freedom, as well as, direct access to upper levels of management. Holtec International shall retain overall responsibility for the QA Program.~~

~~2. Quality Assurance Program~~

~~The Holtec International quality assurance program requires that activities important to safety involving design, procurement, fabrication, inspection and testing are performed in accordance with written procedures. Additional project specific procedures are written as needed when specific project requirements are not covered by quality procedures. These additional project specific quality procedures are considered quality assurance records which are controlled in accordance with Holtec International's quality assurance program. QA manuals and procedures, as well as project specific procedures, are controlled and distributed in accordance with the quality assurance program.~~

~~Holtec International personnel performing important to safety activities must be indoctrinated in the Holtec International quality assurance program prior to performing important to safety work in order assure requirements of the QA program are understood. Additionally, a training session is held each year for Holtec International personnel in order to review specific quality assurance requirements. The effectiveness of the quality program is assessed by upper management through annual audits, in-process assessments, and other means.~~

~~Holtec International personnel performing inspection, testing or auditing activities are qualified in accordance with written procedures using guidelines established by the American Society for Nondestructive Testing, American Society of Mechanical Engineers, American National Standards Institute, or other recognized authority, as applicable. These procedures define education, training, experience, and examination requirements for qualifying personnel to perform inspection, testing or auditing. Qualification records are maintained by the quality assurance manager, or designee, and include certification records, bases for qualification, qualification time period, experience and training records, and examination scores, as applicable. Proficiency of qualified personnel shall be maintained as required through retraining, re-examination, and/or re-certification.~~

~~Contractors used by Holtec International to perform important to safety work may have their own quality assurance program which meets or exceeds Holtec International's, or shall perform the work under Holtec International's quality assurance program.~~

~~QA programs of contractors performing important to safety work are reviewed by Holtec's quality assurance organization through audits, assessments, and surveillances to assure applicable QA criteria will be met.~~

~~A project plan is generated for each important to safety project. The project plan contains the necessary information to enable the project team to execute the project in a well-coordinated manner.~~

~~Disputes involving quality which arise from the difference of opinion between personnel from other departments shall be resolved by the QA Manager.~~

~~3. Design Control~~

~~Holtec International's quality assurance program documents establish measures necessary to assure the control of the design process, from input through verification. A design basis is defined in a design specification so that appropriate codes, standards and other relevant documents are used during the course of the design process. Design parameters, as well as miscellaneous design requirements, such as maintenance, repair and storage, are also defined within the Holtec design specification.~~

~~Drawings, procedures and design reports are the three main documents produced by Holtec International through its design process. Holtec International quality program requirements for procedures and drawings are defined in criterion 5 of the HQAM. Measures are established to assure applicable requirements from design bases documents are translated into drawings, procedures, and reports.~~

~~Quality assurance program documents are established to identify and control the authority and responsibilities of all individuals or groups responsible for design reviews and verification activities.~~

~~Holtec International's quality assurance program documents require that all design reports include, as applicable, a defined purpose, assumptions, references, inputs, outputs and results. Design reports are signed by the author and are reviewed by the Project Manager. Additionally, the design report is verified by an individual or group of individuals other than the author of the report. Verification may be made either by qualification testing, design review or alternate calculations. A design verification checklist is used as part of the review process. When qualification testing is used, the prototype shall be subjected to the most adverse design conditions. Surveillances are performed by members of Holtec's Quality Assurance Department to verify that design reports comply with the requirements of Holtec's QA program.~~

~~Measures are established to assure that design verification shall be performed by qualified personnel who did not perform the design analysis. The verifier shall not have influenced inputs or approaches utilized in the analysis. The analyst's supervisor may perform the verification pursuant to the requirements of NQA-1 [13.3.1].~~

~~Holtec International quality assurance program documents require that design verification, if other than by prototype or lead production quality testing, must be satisfactorily completed prior to release for fabrication unless the timing cannot be met. In this case, written justification must be provided to the Quality Assurance Manager or designee and unverified portions of the design must be identified and controlled.~~

~~Changes to a Holtec International design report and specification are subject to the same design controls and must be reviewed and approved in a similar manner to the original.~~

~~Errors in design shall be addressed in accordance with Criteria 15 and 16.~~

~~When applicable, use of commercial items in an important to safety system, structure, or component shall be reviewed for suitability to their intended function.~~

~~Measures are established for the review and disposition of vendor documents including procedures and drawings.~~

~~Measures are established in the QA program to assure valid industry standards and specifications are used in the selection of design inputs (including suitable materials and processes).~~

~~4. Procurement Document Control~~

~~Holtec International's quality assurance program establishes measures to control the preparation, review, approval and issuance of all important to safety purchase orders. Only suppliers approved in accordance with Criterion 7 shall be qualified to supply important to safety items.~~

~~Measures are established within Holtec International's quality assurance program to ensure that purchase orders contain the following information, codes, standards, and specifications, as applicable:~~

~~a. a statement of the scope of work to be performed by the vendor;~~

~~b. the design basis technical requirements including codes, standards, specifications, etc., to which the item must be designed or manufactured;~~

~~c. quality assurance requirements including as applicable, but not limited to, compliance by the vendor with the requirements of 10CFR21 [13.3.2], 10CFR50, Appendix B, 10CFR71, Subpart H, or 10CFR72, Subpart G; and direct reference to the vendor's quality assurance program.~~

~~d. permission to gain access to the supplier's or subtier supplier's plant facilities and records;~~

~~e. identification of documentation required to be supplied by the vendor for approval by Holtec;~~

~~f. requirements for reporting and approving disposition of nonconformances;~~

~~g. required procedures, tests, and inspections; and~~

~~h. record retainage and control requirements.~~

~~All safety significant purchase orders shall be subject to at least one independent review and concurrence. The QA Department shall conduct required surveillances to ensure that safety significant purchase orders are being issued in accordance with the QA program.~~

~~Changes and revisions to purchase orders shall be subjected to the same or equivalent review and approval requirements as the original document.~~

~~5. Instructions, Procedures and Drawings~~

~~Holtec International quality assurance program documents require that activities that are important to safety must be prescribed and accomplished in accordance with written instructions, procedures or drawings. Methods for complying with the 18 criteria set forth within 10CFR50 Appendix B, 10CFR71, Subpart H, and 10CFR72, Subpart G, are also required to be described within defined procedures.~~

~~Instructions, procedures and drawings are required by the Holtec International quality assurance program to include qualitative and quantitative acceptance criteria in order to verify that activities important to safety have been satisfactorily accomplished.~~

~~Measures are established through the Holtec International quality assurance program to prepare, review, approve, and control these instructions, procedures and drawings. The review of these documents is required to be performed by a cognizant verifier other than the author. Revisions to instructions, procedures and drawings are required to be reviewed and approved in a similar manner to the original revision.~~

~~6. Document Control~~

~~Holtec International's quality assurance program documents establish methods to control the review, approval, and issuance of documents and changes thereto, before release, to ensure that the documents are adequate and applicable quality requirements have been incorporated. Documents that must be controlled shall include, but not be limited to: design specifications; design reports; design and fabrication drawings; procurement documents; QA manuals; design criteria documents; and procedures and instructions (i.e., fabrication, inspection, and testing).~~

~~Measures are established in quality assurance program documents to define individuals or organizations responsible for the review, approval, and control of the documents identified above. Document revisions are required to be reviewed, approved, and controlled in a similar manner to the original document. Review of documents is required to be performed by qualified personnel.~~

~~Quality assurance program documents require that documents required to perform a specific activity shall~~

~~be available at the location where the activity is being performed. Quality assurance program documents also require that obsolete or superseded documents are controlled in order to prevent their inadvertent use.~~

~~An index of project documents is maintained in order to allow identification of the latest revision of applicable documents. This list includes, but is not limited to, design reports, specifications, procedures, and drawings.~~

~~7. Control of Purchased Material, Equipment and Services~~

~~Holtec International quality assurance program documents define measures to ensure that important to safety materials, equipment and services conform to procurement documents. Procedures are established to define requirements for procurement document control, supplier evaluation and selection, vendor surveillance, and receipt inspection in order to assure purchased items are properly controlled from the procurement phase through item receipt.~~

~~Holtec International quality assurance program documents require that Holtec International qualified personnel evaluate Holtec International subcontractors supplying important to safety items and services prior to contract award. A vendor shall be evaluated to determine its technical capability as well as its production capability. Those vendors found to have satisfactory technical and production capabilities are submitted to the quality assurance department for a quality assurance evaluation. The quality assurance evaluation, which shall be documented, shall assess past performance and also determine the capabilities of the vendor to comply with required codes and QA criteria through audit, surveillance, or other source evaluation, as applicable. Unacceptable conditions discovered by Holtec International quality assurance are addressed through nonconformances and audit findings, as applicable. Holtec International shall impose its own quality assurance program on vendors which are determined not to have an adequate quality assurance program; or shall require changes in the supplier's quality assurance program to make it acceptable to Holtec International; or shall perform dedication of the items through surveillance, inspections, and tests in accordance with Holtec International's QA program, as applicable. Qualified suppliers of important to safety items, equipment, and services must be placed on Holtec International's Approved Vendors List. Specific requirements for placing vendors on the Approved Vendor List are defined within Holtec International quality assurance program documents. As applicable, this includes an audit, surveillance, or other source evaluation of the vendor to verify QA program conformance to applicable codes and implementation of the QA program. Measures for performing audits, surveillances, and other source evaluations are defined in quality assurance program documents. As applicable, the QA program requires triennial audits, surveillances, or other source evaluations in order to verify continued implementation of their QA program and maintenance on the Approved Vendors List.~~

~~Measures for performing supplier surveillances are defined within Holtec International quality assurance program documents. Source surveillance is used to determine that in-process work is being~~

~~performed by the supplier in accordance with purchase order requirements. The Project Manager, in conjunction with the Quality Assurance Manager, must determine the extent of source surveillance required for a particular job or supplier based on the important to safety classification, complexity of the item, and quantity. Holtec International quality assurance program documents define types of surveillance activities that may be performed including hold point verification. Project specific procedures and procurement documents define, when applicable, necessary inspection points to be performed by Holtec, and inspection and test acceptance criteria.~~

~~Measures for performing receipt inspection activities are defined within Holtec International quality assurance program documents. Receipt inspection is performed in order to verify received items meet the requirements of the purchase order. The extent of receipt inspection to be performed on vendor-furnished items in order to assure items are properly identified and conform to purchase order requirements is established through Holtec International quality and project procedures, or vendor procedures approved by Holtec. Inspection records, material test reports, and/or certificates of conformance attesting to the acceptance of the item are reviewed, as applicable, for acceptability as part of the receipt inspection process. When item acceptance is contingent on post-installation testing or inspection, the acceptance criteria shall be defined with vendors through procurement documents prior to item use. Items and materials that have completed receipt inspection and are released for fabrication or further use are controlled in accordance with quality assurance program documents.~~

~~Measures have been established through Holtec International quality assurance program documents to control items discovered during receipt inspection to have a nonconforming condition. These measures include segregation and identification of items, evaluation of the nonconforming items, and disposition with justification, as required.~~

~~Holtec International quality assurance program documents establish measures to assure that a supplier provides the documentation for a received part as required by the purchase order. These documents include, but are not limited to, material test reports, inspection and test reports, certificates of conformance and nonconformance reports, as applicable. Review of these documents for conformance to procurement documents is required.~~

~~8. Identification and Control of Materials, Parts and Components~~

~~Holtec International quality assurance program documents establish measures to ensure that materials, parts and components, including partially fabricated assemblies, are adequately identified and controlled in order to preclude the use of incorrect or nonconforming items. Measures are established by Holtec International through its quality documents to ensure that limited life items are controlled in order to preclude their use once the shelf life of these items has expired.~~

~~Measures are established by Holtec International through quality assurance program documents in order to provide the means for material, part or component identification so that items maintain traceability to~~

~~appropriate documentation such as drawings and test reports throughout fabrication, installation and use, and to preclude use of incorrect or defective items. Markings are required to be made such that they are not detrimental to the item. Any specific identification or marking requirements are identified through drawings, procedures, or specifications.~~

~~9. Control of Special Processes~~

~~Holtec International quality assurance program documents establish measures to ensure that special processes such as welding, lead pouring, neutron shield material installation, and NDE examinations are controlled. Specific special processes are typically identified in fabrication specifications. Procedures, equipment, and personnel used to perform special processes are required to be qualified in accordance with applicable codes, standards and specifications. Special process operations shall be performed by appropriately qualified personnel using written and approved procedures, as applicable. Special process operations are required to be documented and verified. Special process records including procedure, equipment and personnel qualifications, as well as special process operation results are required to be maintained as quality records.~~

~~10. Licensee Inspection~~

~~Inspections are required to be performed in accordance with written procedures in order to verify conformance of quality affecting activities. Drawings and specifications are used in conjunction with the procedures to define specific acceptance criteria. Inspection procedures include, as applicable, identification of characteristics and activities to be inspected, acceptance and/or rejection criteria, methods of inspection, identification of the individuals or groups responsible for performing the inspection operation, recording of inspection results, identification of hold and witness points, approval requirements for inspection data and inspection prerequisites such as personnel qualifications. Inspection results are documented and signed by the applicable inspector. Inspections through sampling shall use known standards as applicable for the basis of acceptance.~~

~~Measures are established within Holtec International quality assurance program documents to ensure that structures, systems, and components important to safety are, upon receipt, inspected to verify that the item meets purchase order requirements. Control of materials, both before and after receipt inspection, are defined for both accepted and nonconforming material within Holtec International quality assurance program documents.~~

~~Measures for in-process control are established through project-specific procedures for situations when direct inspection would be impractical. In-process controls when required, may include, but are not limited to, monitoring of processing methods, equipment and personnel, as well as review of in-process documentation.~~

~~Measures are established within the quality assurance program documents to assure that reworked or repaired items are inspected to the original requirements, or approved deviation and new~~

~~requirements.~~

~~Holtec International quality assurance program documents establish measures to ensure that nonconformances identified during the course of fabrication are resolved prior to, or during final inspection; that items which are inspected must be identifiable and traceable to specific records; and that inspection records must be reviewed by the Holtec International QA Manager, or designee, to verify the inspection requirements have been satisfied.~~

~~Holtec International quality assurance program documents require that inspectors shall be qualified in accordance with applicable codes and standards and shall be properly trained. Inspector qualification records are maintained within the quality assurance files and are required to be kept current. Measures are defined within Holtec International quality assurance program documents to ensure that inspection personnel are independent from personnel performing the activity being inspected.~~

~~11. Test Control~~

~~Holtec International quality assurance program documents establish measures to ensure that applicable test programs (i.e., load tests, leak tests, hydrostatic tests, production tests, etc.) are performed in accordance with written procedures, as applicable. Test procedures include, as applicable: test equipment and calibration requirements; material requirements; personnel qualifications; prerequisites (including environmental conditions); detailed performance instructions; hold points; acceptance and rejection criteria; instructions for documenting and evaluating results; and documentation approval requirements.~~

~~The acceptance test program is defined in Chapter 9 of the FSAR for the HI-STORM 100 System and shall be implemented for each system to verify that SSCs conform to the specified requirements and will perform satisfactorily in service.~~

~~Only qualified personnel shall evaluate test results for acceptability.~~

~~12. Control of Measuring and Test Equipment~~

~~Holtec International quality assurance program documents establish measures to ensure that measurement and test equipment shall be calibrated, adjusted and maintained at prescribed intervals or prior to use. Calibrations are required to be performed in accordance with written procedures or standards. Measuring and test equipment is required to be controlled such that the next calibration date and traceability back to calibration records is maintained.~~

~~Measures are established within Holtec International quality assurance program documents to ensure that calibrations of measuring and test equipment are performed using calibration standards that are both traceable and have known valid relationships to nationally recognized standards. When no known~~

~~recognized standard exists, the basis for the calibration is required to be defined and documented.~~

~~Measures are established within Holtec International quality assurance program documents to control measuring and test equipment which is found to be out of calibration. These controls include validation of all previous inspection and test results from the time the item was found to be out of calibration back to the time of the previous acceptable calibration of the same item. Measuring or test equipment found to be out of calibration is required by Holtec International quality assurance program documents to be repaired and recalibrated prior to next use, or replaced.~~

~~A master list of calibrated tools and equipment is required to be kept in order to maintain a complete calibration status of each item.~~

~~13. Handling, Storage and Shipping~~

~~Holtec International quality assurance program documents establish measures to ensure that cleaning, handling, storage and shipping of items are accomplished in accordance with design requirements to preclude damage, loss, or deterioration by environmental conditions. These activities are performed in accordance with written instructions or procedures as necessary. Measures for establishing provisions for the use of special handling, lifting or storage equipment in order to adequately identify and preserve items, components or assemblies are provided within Holtec International quality assurance program documents.~~

~~Measures are established within Holtec International quality assurance program documents to ensure that a review of packaging be performed prior to item shipment in order to assure packaging meets approved drawings, specifications and codes. Additionally, verification of completion of documentation, including procedures, manuals and inspection and test results is required to be performed prior to shipment. Physical identification of the item shall be verified prior to shipment.~~

~~14. Inspection, Test and Operating Status~~

~~Holtec International quality assurance program documents establish measures to ensure the inspection, test and operating status of items is known by organizations responsible for quality activities.~~

~~Measures are established by Holtec International through its quality assurance program documents to control the application and removal of status indicators such as markers and tags. Additionally, Holtec International quality assurance program documents establish measures to ensure that if required operations such as tests or inspections are bypassed, such action is taken through controlled procedures and under cognizance of the quality assurance department.~~

~~Controls on nonconforming items are summarized in Criterion 15.~~

~~15. Nonconforming Materials, Parts or Components~~

~~Holtec International quality assurance program documents establish measures to ensure control of nonconforming important to safety items, services, and activities. This includes provisions for the identification, documentation, tracking, segregation, review, disposition of nonconforming items, and notification of the affected organizations, as appropriate.~~

~~Holtec International quality assurance program documents establish measures to ensure that nonconforming items, services or activities shall be reviewed and dispositioned. Provisions are included to ensure that nonconforming services or activities, including those of suppliers, for which the recommended disposition is "accept as is" or "repair", shall be submitted to the client for approval, if required.~~

~~Measures are established within Holtec International quality assurance program documents to require nonconformances to be identified through deviation reports and corresponding corrective actions (which may include repair, rework, and inspection requirements). Individuals responsible for review and disposition of nonconforming items are identified within Holtec International quality assurance program documents.~~

~~Measures are established within Holtec International quality assurance program documents to control further processing, delivering, or installation of nonconforming or defective items pending a decision on its disposition. Measures are established through Holtec International quality assurance program documents to ensure that nonconforming items are segregated and controlled until proper disposition is completed.~~

~~Holtec International quality assurance program documents establish measures to ensure that the acceptability of nonconforming items is verified by inspecting or testing the nonconforming item against original requirements after designated repair or rework. Final disposition of nonconforming items shall be defined and documented.~~

~~Measures are established within Holtec International quality assurance program documents to permit anyone who discovers a nonconformance to report it in accordance with quality assurance program documents. Provisions are established to ensure that nonconformances are evaluated for the purpose of determining if reporting pursuant to 10CFR21 [13.3.2] is required.~~

~~Holtec International quality assurance program documents require that nonconformances be assessed by the Quality Assurance Manager on a defined basis to determine any quality trends. Any trends or significant results shall be evaluated by appropriate management personnel for development of correction actions.~~

~~Nonconformance reports are considered part of the quality records package. As-built conditions are required to be documented as applicable.~~

~~16. Corrective Action~~

~~Holtec International quality assurance program documents establish measures to ensure that causes of conditions adverse to quality are promptly identified and reported to upper management through deviation reports and corrective action reports. Measures are also established to ensure that corrective actions are performed on identified nonconforming conditions or items, and that follow-ups are performed and documented as applicable to verify implementation and effectiveness of the corrective action.~~

~~Measures are established within Holtec International quality assurance program documents to ensure that follow-up activities are performed to verify that corrective actions have been correctly implemented so as to minimize the possibility of recurrence of the nonconforming condition. Individuals responsible for verifying and documenting corrective action are identified within Holtec International quality assurance program documents.~~

~~Measures are established within Holtec International quality assurance program documents to document and evaluate significant conditions adverse to quality through root cause evaluations. These evaluations are performed by qualified individuals and reviewed by cognizant levels of management.~~

~~17. Quality Assurance Records~~

~~Holtec International quality assurance program documents require that evidence of activities affecting quality shall be documented and shall provide sufficient information to permit identification of the record with the items or activities to which it applies. Quality assurance records include, but are not limited to, design, procurement, manufacturing and installation records; audits (internal and external); nonconformance reports; inspection and test results; drawings (including as-built) and specifications; analysis reports (i.e., failure, seismic, etc.); personnel qualifications and training (including retraining) records; procedures (i.e., inspection, testing, calibration, etc.); calibration records; equipment qualification; corrective action reports; operating logs and completed travelers; material test reports; and design review documents.~~

~~Holtec International quality assurance program documents require that inspection and test records shall, as applicable, contain observations, evidence of inspection or test performance, results of inspections or tests, names of inspectors, date of tests, test personnel and data recorders, equipment identification, and evidence of acceptability. Any nonconforming conditions shall be addressed in accordance with Criterion 15.~~

~~Holtec International quality assurance program documents establish measures to ensure that documents defined as quality assurance records are legible and that they reflect the total of work performed.~~

~~Quality assurance records are defined as either "lifetime" or "nonpermanent", as appropriate. Holtec International quality assurance program documents define which quality assurance records are "lifetime" and which are "nonpermanent". "Lifetime" records are those records that pertain to the design, fabrication and installation of a particular item such that the records can demonstrate the capability of the item and provide evidence of activities supporting the acceptability of the item. These records demonstrate the capability for safe operation; provide evidence of repair, rework, replacement or modification; aid in determining the cause for an accident or malfunction of an item; or provide a baseline for inservice inspection. Examples of "lifetime" records include design reports, drawings, procedures and inspection reports. "Nonpermanent" records are those records that show evidence of an activity being performed but do not meet the criteria for "lifetime" records. Examples of "nonpermanent" records include document transmittal forms and surveillance reports. "Nonpermanent" record retention times are defined within Holtec International quality assurance program documents.~~

~~Holtec International quality assurance program documents establish measures to ensure quality assurance records are properly controlled from receipt through long term storage. Responsibilities for receipt, storage, retrieval and disposal of quality assurance records are provided within Holtec International quality assurance program documents. Records are required to be indexed so that they are readily retrievable.~~

~~Holtec International quality assurance program documents define storage requirements in order to assure quality assurance records are not damaged or destroyed. Quality assurance records are required to be stored in boxes, cabinets or shelves, or on the electronic network, and shall be protected from such conditions as water, fire, etc. Measures are established through Holtec International quality assurance documents to ensure records requiring special storage requirements are stored properly. Quality assurance record storage areas are required by Holtec International quality assurance program documents to have controlled access. In the case where a quality assurance record is damaged or lost, it is required to be replaced immediately in a controlled manner by responsible personnel.~~

~~18. Audits~~

~~Holtec International quality assurance program documents define a comprehensive audit program including independence of the auditors from the area being audited, audit schedule requirements, identification of auditors and their required qualifications, access provisions for audit personnel, documentation requirements, methods for reporting audit findings, and methods for corrective actions and follow-ups.~~

~~Holtec International quality assurance program documents require that schedules be defined for internal and external audits. Audit plans are required to be written for each audit and shall define the key activities or areas to be audited.~~

~~Audits are performed in accordance with written procedures and/or checklists. Audits are performed in order to provide a comprehensive independent verification and evaluation of procedures and activities affecting quality, and to verify and evaluate a supplier's QA program, procedures, and activities. As appropriate, audit teams may contain members who are technical experts in the areas being audited. Holtec International internal audits are required to be performed annually and shall review all aspects of Holtec International's quality assurance program in order to determine the effectiveness of the program. External audits are performed per Criterion 7, as necessary, and shall evaluate all applicable and Holtec International relevant portions of the vendor's quality assurance program.~~

~~Holtec International quality assurance program documents establish qualification requirements for auditors including lead auditors. Additionally, responsibilities of audit personnel regarding the performance of the audit as well as the follow-up documentation (i.e., audit report, findings etc.) are defined within the same documents.~~

~~The Holtec International quality assurance program documents establish requirements for the performance of pre- and post- audit conferences. The pre-audit conference is used to define the scope of the audit as well as the specific areas to be audited, and define a schedule and agenda for the audit. The post-audit conference is used to discuss the results of the audit with the audited party.~~

~~Holtec International quality assurance program documents establish measures for writing of audit reports and provide instructions for the processing of findings and their corresponding corrective actions. Corrective action responses are required to clearly state the corrective action taken to correct the nonconforming condition and date of implementation. Audit reports shall be transmitted to responsible personnel at the audited organization for review~~
~~and implementation of corrective actions, when required. Reports of internal audits shall be transmitted to the president of Holtec International.~~

~~Holtec International quality assurance program documents require that the audit team verify that corrective action responses are made in a timely manner, that the corrective action responses are adequate, and that corrective actions have been properly implemented.~~

13.5 ~~REGULATORY COMPLIANCE~~

This section intentionally deleted.

~~The structure of the Holtec International organization and the assignment of responsibilities for each activity ensures that the designated responsible parties will perform the necessary work to achieve and maintain the quality requirements specified in the HQAM. Conformance to established requirements will be verified by individuals and groups not directly responsible for the performance of the work. The QA Manager, who directly reports to the Executive Vice President of Holtec International, has been designated as the party responsible for verifying quality, and he has the required authority and organizational freedom, including independence from influence of cost and schedule, to effectively complete his responsibilities. The QA Manager can also communicate directly to the President of Holtec International regarding quality assurance activities.~~

~~The Holtec International Quality Assurance Program is documented in the HQAM, HQPs and project specific procedures, and provides adequate control over activities affecting quality, as well as structures, systems, and components that are important to safety, to the extent consistent with their relative importance to safety. The QA program describes a management system and controls, that when properly implemented, will comply with the requirements of Subpart G to 10CFR Part 72 and 10CFR Part 21 [13.3.2].~~

~~Design analyses and engineering documentation for the thermal, structural, confinement, criticality, shielding, and operational capabilities of the HI-STORM 100 System for normal, off-normal and postulated accident conditions are carried out in accordance with the 18 criteria in the HQAM. In addition, those activities and items designated as important to safety and related to the material specification and procurement for the HI-STORM overpack and MPC canister, as well as the HI-STORM 100 lifting equipment, are subject to Holtec QA program procedures. Governing procedures include those for procurement document control, control of purchased items and services, material handling, and instructions and drawings which control material requirements.~~

~~Further, the fabrication, testing and inspection of the HI-STORM 100 System by Holtec International and its subcontractors will be conducted in accordance with all QA program requirements, including those activities and project procedures addressed by the 18 criteria, especially those covering design control, identification, and control of materials, parts and components, test control, inspection procedures, control of special processes, control of measuring and test equipment, and inspection and test status documentation.~~

~~The operation, maintenance, repair and modification of the HI-STORM 100 System will be governed by the licensee's (e.g., utility) QA program with support and record maintenance as required by Holtec's QA program and regulatory requirements. These activities will be verified and audited on a periodic basis with respect to control of nonconforming materials, parts or components, corrective action, quality assurance~~

~~records, audits, and reviews of ongoing inspections, surveillances, and operating status.~~

~~In conclusion, the Holtec International QA Program complies with the applicable NRC regulations and industry standards, and will be implemented for the HI-STORM 100 dry cask storage system.~~

13.6 REFERENCES

[13.~~1~~*0*.1] NUREG/CR-6407, "Classification of Transportation Packaging and Dry Spent Fuel Storage System Components According to Importance to Safety," February 1996.

[13.~~1~~*0*.2] Holtec International Quality Assurance Program, *Revision 13.* ~~Topical Report for 10CFR71, Subpart H and 10CFR72, Subpart G, Holtec International Report HI-941152, Rev. 2 (8/4/94)~~.

[13.~~1~~*0*.3] NUREG-1536, "Standard Review Plan for Dry Cask Storage Systems," January 1997.

[13.0.4] NRC QA Program Approval for Radioactive Material Packages No. 0784, Revision 3.

~~[13.3.1] NQA-1, "Quality Assurance Program Requirements for Nuclear Facilities"~~

~~[13.3.2] U.S. Code of Federal Regulations, Title 10, "Energy", Part 21, "Reporting of Defects and Noncompliance."~~

HI-STORM FSAR

APPENDIX 13.A

~~DESIGN VERIFICATION CHECKLIST~~

INTENTIONALLY DELETED

~~APPENDIX 13.A CONTAINS A TOTAL OF 10 PAGES, INCLUDING THIS PAGE~~

APPENDIX 13.B

INTENTIONALLY DELETED

~~HI-STORM FSAR – APPENDIX 13.B~~

~~HOLTEC QA PROCEDURES~~

~~PROCEDURE NUMBER~~	~~TITLE OF PROCEDURE~~	~~10CFR72 SUBPART G QA CRITERIA~~
~~1.0~~	~~Organization and Responsibilities~~	~~1~~
~~2.0~~	~~Quality Assurance Program~~	~~2~~
~~2.1~~	~~Quality Assurance Manual and Procedure Control~~	
~~2.2~~	~~Execution of HQAM and Extension to a Fabricator's Facility~~	
~~2.3~~	~~Quality Forms for Quality Assurance Program Implementation~~	
~~2.4~~	~~Quality Assurance Requirements for 10CFR71 and 10CFR72~~	
~~2.5~~	~~Quality Assurance Requirements for Supply of ASME Section III Materials, Components, and Equipment~~	
~~2.6~~	~~Execution of Quality Requirements and Extension to a Fabricator's Facility for Important to Safety Categories B and C Items.~~	
~~3.0~~	~~Contract Administration and Design Control~~	~~3~~
~~3.1~~	~~Design Input Requirements~~	
~~3.2~~	~~Design Analysis~~	
~~3.3~~	~~Design Verification~~	
~~3.4~~	~~Design Specifications and Design Criteria Documents~~	
~~4.1~~	~~Purchase Orders~~	~~4~~
~~4.2~~	~~Material Purchase Specifications~~	
~~5.1~~	~~Engineering Drawings~~	~~5~~
~~5.3~~	~~Standard and Project Procedures~~	
~~5.4~~	~~Fast Fax Analysis~~	

~~HI-STORM FSAR - APPENDIX 13.B~~

~~HOLTEC QA PROCEDURES~~

~~PROCEDURE NUMBER~~	~~TITLE OF PROCEDURE~~	~~10CFR72 SUBPART G QA CRITERIA~~
~~6.0~~	~~Document Control~~	~~6~~
~~6.1~~	~~Project Document Transmittal and Control~~	
~~6.2~~	~~Document Classification~~	
~~7.0~~	~~Receipt Inspection~~	~~7~~
~~7.1~~	~~Supplier Selection~~	
~~7.2~~	~~Supplier Surveillance~~	
~~7.3~~	~~Material Dedication – Steel and Weld Wire (Excluding Section III Material)~~	
~~7.4~~	~~Approved Vendor List~~	
~~7.5~~	~~Material Dedication Procedure (For Items Not Covered by HQP 7.3)~~	
~~7.6~~	~~Sampling Plan~~	
~~8.0~~	~~Material and Item Identification and Control~~	~~8~~
~~9.0~~	~~Qualification of Personnel Performing Holtec Special Processes~~	~~9~~
~~9.1~~	~~Written Practice for Qualification of NDE Personnel~~	
~~9.2~~	~~Welder Qualification Requirements~~	
~~9.3~~	~~Inspector Qualification for Non-NDE Activities~~	
~~11.0~~	~~Computer Programs (Formerly HQP 5.2)~~	~~11~~
~~12.0~~	~~Equipment Calibration and Control of Measuring and Test Equipment~~	~~12~~
~~14.0~~	~~Inspection and Test Status~~	~~10,11,14~~
~~15.1~~	~~Reporting of Defects and Noncompliances per 10CFR21~~	~~15~~
~~15.2~~	~~Nonconformances~~	

~~HI-STORM FSAR – APPENDIX 13.B~~

~~HOLTEC QA PROCEDURES~~

~~PROCEDURE NUMBER~~	~~TITLE OF PROCEDURE~~	~~10CFR72 SUBPART G QA CRITERIA~~
~~16.0~~	~~Corrective Action (formerly Non-Conformance and Corrective Action)~~	~~16~~
~~16.1~~	~~Root Cause Evaluations~~	
~~17.0~~	~~Quality Assurance Records~~	~~17~~
~~18.1~~	~~Certification of Audit Personnel~~	~~18~~
~~18.2~~	~~Audits~~	
~~19.1~~	~~Personnel Reliability Program~~	~~2~~
~~19.2~~	~~Field Services~~	
~~19.3~~	~~Qualification Requirements and Duties of Registered Professional Engineers for Section III, Division 1 Certifying Activities~~	

~~Notes: 1. Handling, Storage, and Shipping Requirements are specified in the QA Manual (Section XIII). These activities are performed by Holtec subcontractors in accordance with project specific procedures.~~