HOLTEC INTERNATIONAL

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HI-STORM 100 CERTIFICATE OF COMPLIANCE 72-1014

CERTIFICATE AMENDMENT REQUEST 1014-1

REVISION 1

AUGUST, 2000

NON-PROPRIETARY VERSION

SUMMARY OF PROPOSED HI-STORM 100 CHANGES[1]

SECTION I – PROPOSED CHANGES TO CERTIFICATE OF COMPLIANCE 1014

Proposed Change No. 1

Certificate of Compliance, Appendix A, LCO 3.1.1, SR 3.1.1.2, and Table 3-1:

The MPC helium backfill *density* limit is revised to be a maximum helium backfill *pressure* range as shown in the attached marked-up LCO and table.

Reason for Proposed Change

The existing units of g-mole/liter for this TS limit was found, in practice, to be cumbersome to implement. Therefore, a change in favor of a simpler requirement is warranted. Pressure is a readily measurable parameter in the field.

Justification for Proposed Change

The proposed change to the MPC helium backfill TS requires the users to backfill the MPC within a range of helium pressures. This ensures the presence of helium in the MPC free space. Helium backfill pressure in the range specified by the CoC is consistent with the governing thermal analyses. Helium backfill pressure within this pressure range ensures the proper density of helium to support MPC internal convection heat dissipation.

Proposed Change No. 2

Certificate of Compliance, Appendix A, SR 3.1.2.1 and LCO 3.1.3:

a. Revise the Surveillance Requirement acceptance criterion to 126 degrees F.

b. Revise the Completion Time for LCO 3.1.3, Required Action A.2 from 24 hours to 22 hours

Reason and Justification for Proposed Changes

The revised delta T limit and Completion Time are necessary due to the higher heat duty for the cask system as discussed elsewhere in this section (see Proposed

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[1] Proposed changes marked with a "*" have previously been submitted under License Amendment Request (LAR) 1008-1 for HI-STAR 100 (Docket 1008, 11/24/99). These changes have been reviewed by the NRC (SFPO) and forwarded to NMNS for rulemaking as of the date of this LAR.

Change No. 28). The higher heat duty is based on credit being taken for internal convection heat dissipation inside the MPC. These changes ensure fuel cladding temperatures are maintained below established limits for all heat loads, up to and including the design basis maximum.

Proposed Change No. 3

Certificate of Compliance, Appendix A, LCO 3.2.1:

Revise the HI-TRAC dose rate acceptance criteria as shown on the attached mark-ups of the LCO.

Reason for Proposed Change

The addition of the MPC-32 basket, higher fuel burnups and non-fuel hardware have increased the dose rates for the loaded HI-TRAC 100 and HI-TRAC 125 transfer casks.

Justification for Proposed Change

The HI-TRAC dose rates are based on conservative, design basis source terms, using relatively low cooling times and high burnups. Users, simply through the nature of core operating cycles, will likely not have any one MPC loaded with design basis fuel. Users will determine the actual (lower) expected dose rates based on their particular fuel characteristics prior to fuel loading. The purpose of this LCO is simply to provide a limit above which users should suspect that a fuel assembly (or multiple fuel assemblies) not meeting the CoC has been loaded into the MPC, and they must take the action required by the Technical Specifications. Users' radiation protection/ALARA programs and operating procedures will control the use of temporary shielding and specific operating activities, as appropriate to ensure doses are ALARA. Note that the TSAR currently recommends that users choose the 125-ton HI-TRAC transfer cask because it provides better shielding. However, users with lower capacity cranes will need to perform an ALARA evaluation to either upgrade their crane capacity or implement temporary shielding to ensure occupational exposures are ALARA.

Proposed Change No. 4

Certificate of Compliance, Appendix A, LCO 3.2.3:

a. The LCO acceptance criteria for the side of the overpack and the inlet and outlet vents are increased to 50 and 40 mrem/hr, respectively.

b. The LCO Applicability is revised to delete "TRANSPORT OPERATIONS."

c. Required Action A.2 is revised to substitute a written evaluation in lieu of an analysis.

Reason for Proposed Changes

a. Both dose rate limits are increased due to the addition of the MPC-32 basket, higher burnup fuel, and non-fuel hardware. The inlet and outlet vent duct dose rate limit is also slightly increased due to the new HI-STORM 100S overpack design and high burnup fuel.

b. The dose rate acceptance criteria are not required to be met until the overpack is in its final storage configuration and in its designated storage location at the ISFSI. Therefore, having this LCO applicable during TRANSPORT OPERATIONS is not appropriate.

c. This change is proposed to provide appropriate flexibility for user in evaluating the nonconforming condition.

Justification for Proposed Change

a. In both cases, the higher dose rate acceptance criteria are a result of increasing the number of PWR fuel assemblies in the MPC with the addition of MPC-32, adding high burnup fuel, as well as adding non-fuel hardware to the contents of the PWR MPCs. The duct dose rates are also affected by the design changes made to create the HI-STORM 100S, which include shortening the overall length of the HI-STORM overpack (see Proposed Change No. 33). This involved changes to the lid design, which incorporates the outlet ducts directly into the lid, and shortening the pedestal upon which the MPC rests. These changes moved the MPC closer to the top of the inlet ducts and closer to the bottom of the outlet ducts.

   While these changes increase dose rates somewhat, they remain low. Further, use of the 32-assembly MPC will reduce the total number of MPCs to be loaded by a given PWR user, thereby reducing the total occupational dose over an entire loading campaign. Increasing the dose rate limits will not jeopardize the ability of the system to meet the 10CFR72.104 requirements for off-site dose. In addition, each site will perform an evaluation considering their specific fuel to demonstrate compliance with 10CFR72.104 prior to utilizing the HI-STORM 100 system.

b. In its final storage configuration, the overpack has its gamma shield cross plates installed in the inlet and outlet ducts. If the overpack is transported while supported from the bottom (e.g., with air pads) these shielding devices cannot be installed until the overpack is at its final storage location. This change is also consistent with the current Surveillance Requirement Frequency, which does not require measuring dose rates until within the first 24 hours after the beginning of STORAGE OPERATIONS. By definition, STORAGE OPERATIONS begin when the overpack is at the ISFSI.

c. A written evaluation may include an analysis but does not necessarily need to. Depending upon the circumstances and magnitude of the high dose rates, an evaluation may include something less than an analysis and the user should have the option of performing the appropriate type of evaluation for the situation. This proposed change makes HI-STORM consistent with the dose rate LCO for HI-STAR (LCO 2.2.1).

Proposed Change No. 5

Certificate of Compliance, Appendix A, LCO 3.3.1:

This new LCO is added to provide limits for the minimum soluble boron concentration during wet loading and unloading operations for the MPC-32 with relatively higher enriched fuel in the MPC-24, MPC-24E, and MPC-24EF.

Reason for Proposed Change

Many PWR users need to load fuel up to 5% initial enrichment. In order to authorize storage of any reasonably enriched PWR fuel in the MPC-32 and relatively higher enriched PWR fuel in the MPC-24, MPC-24E, and MPC-24EF (discussed later in Section I), credit for soluble boron in the MPC water during wet loading and unloading operations was taken in the criticality analyses. Since this is a licensee-controlled operational activity related to reactivity, a new technical specification LCO is being created to establish appropriate limits, actions, and surveillance requirements for boron concentration during these operations.

Justification for Proposed Change[2]

Criticality calculations have been performed demonstrating that for the listed conditions (maximum enrichment and minimum soluble boron concentration) for each MPC, the cask system is in compliance with the regulatory requirement of k_{eff} <0.95 for all PWR fuel array/classes. The maximum k_{eff} calculated for the HI-TRAC is 0.9447 for the MPC-24, 0.9399 for the MPC-24E and MPC-24EF, and 0.9470 for the MPC-32. In the HI-STORM storage configuration, where no water is present inside the MPC, the maximum k_{eff} is below 0.52 for all PWR fuel array/classes and MPC models. Additional results, including results from the HI-STAR TSAR, which are directly applicable to the HI-TRAC, can be found in Tables 6.1.2 and 6.1.4 through 6.1.6 in Section 6.1 of the Proposed Rev. 11 of the TSAR (see Attachment 5).

Proposed Change No. 6

Certificate of Compliance, Appendix A, Sections 5.1 and 5.2:

Delete the training program and pre-operational testing and training exercise requirements entirely.

Reason and Justification for Proposed Changes

Part 72 training requirements are governed directly by the regulations at 10 CFR 72.144(d), 72.190, and 72.192, and through licensees' Quality Assurance programs. Both the regulations and the QA program require licensees to have trained and qualified personnel performing activities important to safety. Therefore, it is unnecessary to duplicate training requirements in the CoC. Further, while the Systematic Approach to Training (SAT) is a commonly used training program development technique, it is inappropriate to impose SAT on licensees via the CoC. All topical areas to be included in the licensees' dry spent fuel storage training program, including the pre-operational testing and training exercises currently in the CoC, are already part of the HI-STORM 100 FSAR, Chapter 12 and, as such, are required to be implemented by licensees. This change is consistent with those being proposed generically by the industry through the NEI technical specification improvement effort.

2 This justification is focused on the criticality aspects of soluble boron. Refer to the new Bases for LCO 3.3.1 proposed to be added to TSAR Chapter 12, Appendix 12.A (Proposed Change No.41) for discussion of the Required Actions and Surveillance Requirements, Frequencies, etc.

Proposed Change No. 7

Certificate of Compliance, Appendix A, Section 5.3:

Move "Special Requirements for First Systems in Place" from the TS to the CoC proper as new Item 9. Re-number existing Item 9 as new Item 10.

Reason and Justification for Proposed Change

This is an administrative change. One-time requirements are more appropriately located as conditions to the CoC (similar to Part 50 license conditions) rather than technical specifications. This change is consistent with those being proposed generically by the industry through the NEI technical specification improvement effort.

Proposed Change No. 8

Certificate of Compliance, Appendix A, Section 5.5 and Table 5-1:

a. The Cask Transport Evaluation Program description has been re-formatted and revised as shown in the attached CoC mark-up pages to modify Table 5-1 and add Subsection 5.5.b to distinguish between the transport of free-standing overpacks and overpacks to be deployed in high-seismic regions (HI-STORM 100A).

b. The Cask Transport Evaluation Program description, at Subsections 5.5.a.1 and 5.5.a.2, has been revised as a conforming change to support the change to Design Features Section 3.4.6 to eliminate the specific ISFSI pad design criteria (see Proposed Change No. 32).

c. New specification item 5.5.a.3 is added to address the transport of the loaded TRANSFER CASK or free-standing OVERPACK from the FUEL BUILDING to the ISFSI. The new section allows lifting of the loaded TRANSFER CASK or OVERPACK to any height necessary provided the lift device is designed in accordance with ANSI N14.6 and includes redundant drop protection features.

Reason for Proposed Changes

a. This change is necessary because there is no specific (generic) drop height or reference ISFSI pad established for the HI-STORM 100A overpack design. Each user must determine a lift height on a site-specific basis, except as provided for in Subsection 5.5.b.2. Subsection 5.5.b.2 allows for no lift height to be established if the cask is lifted with appropriately designed lift devices.

b. This is a conforming change. References to ISFSI pad design criteria are no longer meaningful, as these criteria (in Design Features Section 3.4.6) are being deleted from the CoC as part of this LAR (see Proposed Change No. 32).

c. This change is proposed based on user feedback which indicated there were no requirements established for onsite transport of the TRANSFER CASK or OVERPACK that address lifting the TRANSFER CASK or OVERPACK above the lift height limits outside the scope of the Cask Transfer Facility (CTF). This flexibility may be required at some sites based on the transport path between the FUEL BUILDING and the ISFSI.

Justification for Proposed Change

a. The HI-STORM 100A overpack design includes unique design features that make the existing, free-standing drop and tipover analyses (and the lift heights in Table 5-1) not applicable. Each ISFSI pad on which a HI-STORM 100A is deployed will be designed site-specifically accounting for the unique seismic spectra for the site. Therefore, the lift heights for the HI-STORM 100A overpack design will also be determined site-specifically, if required, based on the type of handling device contemplated for use (per Specification 5.5.b.1). If lift devices designed in accordance with ANSI N14.6 and having redundant drop protection features are used, drop events are not credible and, therefore, no lift height limit need be established.

b. Conforming change in support of the removal of ISFSI pad design criteria from the CoC.

c. A lift device designed in accordance with ANSI N14.6 and having redundant drop protection features ensures that a drop of the TRANSFER CASK or OVERPACK is not a credible event. This change provides necessary flexibility for users with non-compliant transport path conditions (e.g., a portion of the path that is harder than the "pre-approved" pad design parameters described in TSAR Table 2.2.9). This change is consistent with HI-STAR 100 LCO 2.1.3.b.

<u>Proposed Change No. 9</u>

<u>Certificate of Compliance, Appendix B, Section 1.0, and Tables 2.1-2 and 2.1-3:</u>

The definitions of DAMAGED FUEL ASSEMBLY and INTACT FUEL ASSEMBLY are revised as shown in the attached marked-up CoC changes. The terms "No. of Fuel Rods", "Clad OD", "Clad ID", and "Pellet Diameter" are all revised for clarity.

Reason for Proposed Change

The revised definitions and terms more accurately reflect the criticality analyses and eliminate potential unintended CoC compliance problems for licensees.

Justification for Proposed Changes

The criticality analyses were performed for a large variety of fuel assembly arrays and classes. Where appropriate to the fuel assembly array/class, fuel rods were modeled in all fuel rod locations. However, situations may arise for licensees where a particular fuel assembly may not, and may never have had, fuel rods in all fuel rods locations. In such cases, it is important to ensure the fuel rod locations are filled with dummy fuel rods that occupy space (in lieu of moderator) at least as large as the fuel rod modeled there. Further, fuel assemblies with missing fuel rods *not* replaced with dummy rods are to be classified as DAMAGED FUEL ASSEMBLIES. DAMAGED FUEL ASSEMBLIES must meet the fuel specifications of Tables 2.1-2 and 2.1-3. The current "No. of Fuel Rods" requirement in these tables clearly cannot be met by this type of DAMAGED FUEL ASSEMBLY. The wording change for this term eliminates this potential compliance problem.

<u>Proposed Change No. 10</u>

<u>Certificate of Compliance, Appendix B, Section 1.0:</u>

The definition of DAMAGED FUEL CONTAINER (DFC)[3] in Appendix B is revised to include three additional DFCs in addition to the previously approved Holtec DFC designed exclusively for Dresden Unit 1 and Humboldt Bay fuel. The new DFC designs are: 1) a Transnuclear (TN) DFC currently containing Dresden Unit 1 (D-1) fuel*, 2) a Holtec generic PWR DFC, and 3) a Holtec generic BWR DFC. Detailed drawings for the TN/D-1 DFC are contained in Holtec LAR 1008-1 for HI-STAR 100 submitted to the NRC on November 24, 1999. Sketches of the TN/D-1 DFC and the two new Holtec-designed DFCs are included as proposed new TSAR Figures 2.1.2, 2.1.2B and 2.1.2C (see Attachment 5). In all cases, only outline sketches showing key DFC dimensions

[3] The terms Damaged Fuel Container and Damaged Fuel Canister are used interchangeably throughout this document and "DFC" is applicable to both.

and general fabrication details are included in proposed TSAR Revision 11. Detailed design drawings of the Holtec DFC are being removed from the TSAR with this amendment request. This change is consistent with previously approved changes for the HI-STAR 100 System under LAR 1008-1.

Reason for Proposed Changes

TN/D-1 DFC

There are a significant number of Dresden Unit 1 fuel assemblies meeting the HI-STORM fuel specifications which are currently stored in TN DFCs. Authorizing this fuel for storage in the HI-STORM 100 system without having to remove it from the TN/D-1 DFCs and load it into the Holtec DFCs will avoid imposing undue burden on the general licensee with no additional safety benefit. Implementation of this change will allow Dresden Unit 1 to complete decommissioning of the plant in a timely manner. Further, the fuel in the TN/D-1 DFCs is currently located in the Dresden Unit 2/3 spent fuel pool. Removal of this fuel is necessary to maintain full core offload capability and allow D-2/3 to continue operation.

Holtec Generic PWR and BWR DFCs

The current HI-STORM CoC authorizes only damaged fuel and fuel debris from the Dresden Unit 1 and Humboldt Bay plants for storage in HI-STORM 100. Many other customers have informed Holtec that some of their fuel would be classified as damaged fuel or fuel debris. These new generic DFC designs allow for storage of a much broader scope of damaged fuel and fuel debris for both PWR and BWR fuel.

Justification for Proposed Changes

TN/D-1 DFC

The justification for this proposed change is provided below, arranged by technical discipline, as applicable. Supporting changes to the TSAR are summarized in Section II of this attachment and included in Attachment 5.

Structural Evaluation

The TN/D-1 DFC was previously approved for use in the TN-9 transportation package. In addition, the TN/D-1 DFC has been structurally evaluated by Holtec International and found to meet all design requirements for storage in the HI-STORM 100 system. The details of this evaluation are contained in proposed new TSAR Appendix 3.AR, included in Attachment 5. All required safety margins are greater than zero or, in other words, the factors of safety are greater than 1.0.

The TSAR Chapter 3 NUREG-1536 compliance matrix has been revised to address the new DFCs and the supporting appendix. Since all required text changes are confined to the new appendix, no new chapter text is required.

Thermal Evaluation

Storage of D-1 damaged fuel and fuel debris meeting the specifications of the CoC is permitted in the HI-STORM MPC-68, MPC-68F, and MPC–68FF when encased in a DFC. The thermal characteristics of the TN/D-1 DFC and the Holtec DFC were compared in support of this amendment request. The TN/D-1 DFC is a square shaped canister box fabricated from 12 gage stainless steel plates. A bounding thermal calculation has been prepared in support of this amendment to determine the most heat resistive fuel from the Low Heat Emitting (LHE) group of assemblies encased in a DFC. It is noted that in this configuration, interruption of radiation heat exchange between the fuel assembly and the fuel basket by the DFC boundary renders the DFC configuration as the bounding case when compared with the absence of a DFC. Both canister designs were evaluated and the one exhibiting lower heat dissipation characteristics was adopted for analysis.

For the LHE group of assemblies, the low decay heat load of D-1 fuel (approximately 8 kW) guarantees large thermal margins to permit safe storage of D-1 fuel in the TN/D-1 DFC. The HI-STORM temperature field for this case was calculated and is reported in proposed revisions to HI-STORM TSAR Chapter 4 at Subsection 4.4.1.1.13 (see Attachment 5). Substantial cladding thermal margins are demonstrated by the analysis.

Shielding Evaluation

Storage of D-1 damaged fuel and fuel debris meeting the specifications of the CoC is permitted in the HI-STORM MPC-68, MPC-68F, and MPC-68FF when encased in a DFC. Sections 5.4.2 and 5.4.5 of the HI-STORM TSAR, Revision 10 discuss the post-accident shielding evaluation for D-1 and Humboldt Bay damaged fuel. These sections assume that the damaged fuel assemblies and fuel debris collapse to a height of 80 inches. This dimension was calculated based on the inside dimension of the DFC and the dimensions of the fuel assemblies. Since the TN/D-1 DFC has a smaller inside dimension than the Holtec DFC, the analysis in Sections 5.4.2 and 5.4.5 of the HI-STORM TSAR is applicable and conservative. In addition, the shielding analysis does not take credit for the DFC container in determining the acceptability of storing the approved damaged fuel and fuel debris. Therefore, the use of the TN/D-1 DFC does not affect the shielding analysis and no changes to the Chapter 5 of the TSAR are necessary as a result of this proposed change.

Criticality Evaluation

The TN/D-1 DFC was analyzed with the same set of contents used for the analysis of the Holtec DFC documented in Rev. 10 of the HI-STORM 100 TSAR. This set includes 6x6 and 7x7 fuel assemblies with various numbers of rods missing, a collapsed assembly and dispersed fuel powder. The maximum k_{eff} values for both DFCs are listed in proposed Revision 11 TSAR Table 6.4.5 (Attachment 5). There is no significant difference in reactivity between the two DFCs. For only one case (collapsed assembly), the reactivity for the TN/D-1 DFC is increased marginally ($\Delta k = 0.0012$) compared to the Holtec DFC. In all other cases, the reactivity for the TN/D-1 DFC is below the reactivity of the Holtec DFC with the same contents. Therefore, with the TN/D-1 DFC used instead of the Holtec DFC, the cask system is still in compliance with the regulatory requirement of $k_{eff} < 0.95$ for all authorized contents.

HOLTEC GENERIC PWR DFC

Structural Evaluation

The proposed Holtec generic PWR DFC design (see new TSAR Figure 2.1.2B) is a square shaped tube fabricated from 0.075-inch stainless steel. An appropriate cover is included that permits lifting of the unit. The structural evaluation of the generic DFC design for PWR fuel is based on the same design criteria used for the approved Holtec DFC for Dresden/Humboldt Bay fuel. Structural analyses have been performed for the lifting condition (where NUREG-0612 stress limits are applicable) and for a handling accident leading to an end impact (ASME Code Level D limits are applicable). Positive safety margins are achieved. The results are presented in Appendix 3.AS (see Attachment 5).

Thermal Evaluation

The proposed PWR DFC design (see proposed TSAR Rev.11 Figure 2.1.2B in Attachment 5) is a square shaped tube fabricated from 0.075-inch stainless steel. Bounding thermal calculations have been prepared in support of this amendment to determine the most heat resistive Zircaloy and stainless steel clad fuels encased in DFCs. In this configuration, interruption of thermal radiation heat exchange between the fuel assembly and the fuel basket by the DFC renders the DFC configuration as bounding when compared with non-canistered assemblies. Storage of damaged PWR fuel assemblies in generic DFCs is evaluated in proposed TSAR Revision 11, Subsection 4.4.1.1.4 (see Attachment 5). The MPC-24E/24EF is designed with four enlarged fuel storage cells to accommodate the DFC. The CoC requires damaged fuel to be stored only in these particular fuel storage locations to preserve the assumptions of the analysis. At least 20 of the 24 fuel storage locations will be occupied by intact fuel assemblies. Therefore, the overall effect of DFC storage on the basket heat dissipation rate is quite small. Conservatively, a 5% reduction MPC heat rating is specified for accommodating damaged, Zircaloy clad fuel. Stainless steel clad fuel storage is evaluated in TSAR Subsection 4.3.2 for a bounding storage configuration (within a DFC).

Shielding Evaluation

The Holtec generic PWR DFC is designed to accommodate any PWR 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 cannot 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-TRAC or the HI-STORM overpacks, both of which provide more shielding than the 100-ton HI-TRAC.

Fuel debris or a damaged fuel assembly which has collapsed can have an average fuel density that 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 four peripheral damaged fuel locations in the MPC-24E (or MPC-24EF) and the 16 peripheral locations in the MPC-68 (including the MPC-68FF) with a fuel density that was twice the normal fuel density and correspondingly increasing the source term 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.

The results of this analysis indicate that the dose rates in the top and bottom portion of the 100-ton HI-TRAC increase slightly while the dose rate in the center of the HI-TRAC actually decreases a little bit. The increase in the top and bottom is due to the assumed flat power distribution. 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 amendment request. Section 5.4.2 of proposed TSAR Revision 11 (see Attachment 5) provides the discussion and a presentation of the results of the damaged fuel analysis.

Criticality Evaluation

Criticality calculations have been performed for the MPC-24E and MPC-24EF loaded with intact fuel, damaged fuel, and fuel debris (up to 4 DFCs per basket) with a maximum enrichment of 4.0 wt% ^{235}U. The calculations use a bounding approach to account for the possible wide variation of fuel distribution inside the DFC, based on the analysis of arrays of bare fuel rods. Additionally, typical damaged fuel conditions such as missing rods or collapsed assemblies are analyzed for selected array/classes. The analyses are presented in Section 6.4.4.2 of the Proposed Rev. 11 of the TSAR (see Attachment 5). The maximum calculated k_{eff} for the HI-TRAC is 0.9486, which demonstrates that the cask system is in compliance with the regulatory requirement of k_{eff} <0.95 for all PWR fuel array/classes.

HOLTEC GENERIC BWR DFC

Structural Evaluation

The proposed Holtec generic BWR DFC design (see new TSAR Figure 2.1.2C) is a square shaped tube fabricated from 0.035-inch stainless steel. An evaluation of structural integrity under lifting and handling accident conditions has been performed, similar to that performed for the generic PWR DFC. Positive safety margins are achieved. Structural integrity results are reported in Appendix 3.AS (see Attachment 5).

Thermal Evaluation

Bounding thermal calculations have been prepared for the Holtec generic BER DFC design to determine the most heat resistive Zircaloy and stainless steel clad fuels encased in DFCs. In this configuration, interruption of thermal radiation heat exchange between the fuel assembly and the fuel basket by the DFC renders the DFC configuration as bounding when compared with non-canistered assemblies. Storage of damaged BWR fuel assemblies in generic DFCs is evaluated in proposed TSAR Revision 11, Subsection 4.4.1.1.4 (see Attachment 5). The MPC-68 and MPC-68FF are analyzed assuming damaged fuel is stored in up to 16 peripheral fuel storage cells in DFCs. The CoC requires damaged fuel to be stored only in these particular fuel storage locations to preserve the assumptions of the analysis. At least 52 of the 68 fuel storage locations will be occupied by intact fuel assemblies. Therefore, the overall effect of DFC storage on the basket heat dissipation rate is quite small. Conservatively, a 5% reduction MPC heat rating is specified for accommodating damaged, Zircaloy clad fuel. Stainless steel clad fuel storage is evaluated in TSAR Subsection 4.3.2 for a bounding storage configuration (within a DFC).

Shielding Evaluation

See justification for Holtec Generic PWR DFC.

Criticality Evaluation

Criticality calculations have been performed for an MPC-68 loaded with intact fuel, damaged fuel, and fuel debris (up to 16 DFCs) Maximum enrichments of up to 4.0 wt% ^{235}U for the damaged fuel/fuel debris and up to 3.7 wt% ^{235}U for the intact fuel were analyzed. The calculations use a bounding approach to account for the possible wide variation of fuel distribution inside the DFC, based on the analysis of arrays of bare fuel rods. Also, typical damaged fuel conditions such as missing rods or collapsed assemblies are analyzed for selected array/classes. The analyses are presented in Section 6.4.4.2 of the Proposed Rev. 11 of the TSAR. The maximum calculated k_{eff} is 0.9328, which demonstrates that the cask system is in compliance with the regulatory requirement of k_{eff} <0.95 for all PWR fuel array/classes.

Proposed Change No. 11

Certificate of Compliance, Appendix B, Subsection 2.1.1 and Table 2.1-1:

a. The wording of Item 2.1.1.a is revised to add the words "and NON-FUEL HARDWARE" and "and other referenced tables."

b. Item 2.1.1.c is revised to add a clarification that this requirement applies only to uniform loading.

c. New Item 2.1.1.e is added; the note in Table 2.1-1, Item II.C is revised; and the word "Zircaloy" is removed from Table 2.1-1, Items II.A.1 through 4 to reflect the authorization for loading of LaCrosse BWR fuel assemblies in stainless steel channels (array/class 10x10D and 10x10E) in the MPC-68. Similar provisions are made for storage of stainless steel channels in the MPC-68FF (see Proposed Change No. 21).

Reason for Proposed Changes

a. This change is provided to clarify that PWR fuel may be stored with non-fuel hardware as discussed in Proposed Change Number 14, and to clarify that Table 2.1-1 incorporates other tables by reference.

b. Without this clarification, regionalized fuel loading would not be possible with damaged fuel assemblies and fuel debris due to this limitation on decay heat.

c. LaCrosse plant has stainless steel channels and is a Private Fuel Storage, LLC (PFS) member. HI-STORM 100 is one of the storage cask designs referenced in the PFS Part 72 license application.

Justification for Proposed Changes

a. Clarification to recognize that non-fuel hardware (as defined in Table 2.1-1) is authorized for loading with PWR fuel. The second change is editorial.

b. For the regionalized fuel storage configuration described in proposed TSAR subsection 4.4.1.1.9, low heat emitting fuel is arrayed away from the central region occupied by hotter fuel. The note is added so that the regionalized loading strategy is not unduly restricted by a stipulation designed for uniform loading.

c. The justification for this change is presented by technical discipline below.

<u>Structural Evaluation</u>

As the CoC does not permit the total weight of the fuel assembly plus the non-fuel hardware to exceed the design basis weights (BWR -700 lb., PWR –1680 lb.), there are no new structural evaluations nor changes to existing evaluations required.

Thermal Evaluation

Zircaloy and stainless steel have comparable thermal conductivities, the latter being approximately 10% greater than the former. The thermal analysis presented in Revision 10 of the TSAR and proposed Revision 11 utilize the thermal properties of Zircaloy. Even though the thermal conductivity of the stainless steel channels is greater than that of a Zircaloy channel, the aggregate impact of the thermal properties of the fuel channel on the overall basket conductivity is quite modest. As a result, small differences in the thermal properties (e.g., conductivity, emissivity, etc.) of stainless steel and Zircaloy channels produce a second order effect on the thermal performance of the storage system. Therefore, the analyses using Zircaloy channel properties are also considered to be applicable to stainless steel channels.

Shielding Evaluation

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 the shielding evaluation. 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 or MPC-68FF is being limited to sixteen and Appendix B to the CoC 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 shut down 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 of Revision 10 of the HI-STORM TSAR). 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 or MPC-68FF, a comparison of source terms is performed. Table 5.2.8 of Revision 10 of the HI-STORM TSAR 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 used in the shielding analysis. Therefore, it is concluded that the storage of 16 stainless steel channels in an MPC-68 is acceptable.

This discussion is provided in Section 5.2.8 of proposed TSAR Revision 11 provided in Attachment 5.

Criticality Evaluation

The criticality calculations presented in Chapter 6 of the HI-STORM TSAR for BWR fuel array/classes 10x10D and 10x10E have been performed using Zircaloy as the material for the flow channels. Stainless steel, which is used for some of the these assemblies, has a higher neutron absorption than Zircaloy, which would lead to a slight reduction in reactivity. The calculations using Zircaloy are therefore bounding for assemblies with stainless steel channels and no further calculations are required.

Proposed Change No. 12

Certificate of Compliance, Appendix B, Sections 2.1.2 and 2.1.3; and Tables 2.1-6 and 2.1-7:

a. Subsection 2.1.2 is revised to state that preferential loading is applicable during uniform loading (which is also defined) and to state that regionalized loading meets the intent of preferential loading.

b. New Subsection 2.1.3 and Figures 2.1-1 through 2.1-4 are added to introduce regionalized fuel loading as an option. Specific cooling time, burnup, and decay heat limits for regionalized fuel loading are specified in Tables 2.1-6 and 2.1-7 in the Approved Contents section of Appendix B to the CoC.

Reason for Proposed Change

a. Clarification to distinguish between uniform fuel loading and regionalized fuel loading and to clarify that regionalized loading meets the intent of preferential fuel loading.

b. Regionalized fuel loading, in accordance with Figures 2.1-1 through 2.1-4 and Tables 2.1-6 and 2.1-7, as applicable, allows users to load relatively higher heat emitting fuel assemblies than would otherwise be allowed using uniform fuel loading.

Justification for Proposed Change

a. Clarification

b. This change is proposed to allow users a method to store fuel assemblies with higher heat emission rates with those having lower heat emission rates, while remaining within the total heat dissipation capabilities of the storage cask design. The specific technical justification is arranged by affected technical discipline below.

Thermal Evaluation

In the regionalized fuel loading scenario, a two-region fuel configuration is analyzed. The two regions are defined as an inner region (Region 1) for storing relatively hot fuel, and an outer region (Region 2) physically enveloping the inner region and storing relatively cooler fuel. These regions are specifically defined by fuel storage cell number in Appendix B to the CoC. To permit hot fuel storage in the inner region, a low decay heat rate is specified for fuel in the outer region. The maximum allowable heat load for the inner region fuel is then a function of fuel age-dependent permissible cladding temperatures. The regionalized fuel loading thermal modeling is discussed in detail in proposed TSAR Subsection 4.4.1.1.9 and the results of the analysis are provided in proposed TSAR Subsection 4.4.2 (see Attachment 5).

Shielding Evaluation

Regionalized loading in the HI-STORM cask system is used to place fuel with higher heat emission rates (higher burnups and shorter cooling times) in the center of an MPC surrounded by fuel with lower heat emission rates (lower burnup and longer cooling time). From a shielding perspective, the older fuel on the outside of the MPC is serving as shielding for the fuel on the center of the MPC for the dose rates on the side of the casks. The dose rates on the ends of the casks, however, increase as a result of putting hotter fuel on the inside of the MPC. However, this is a localized effect.

Proposed TSAR Revision 11, Section 5.4 in Attachment 5 provides a discussion of regionalized fuel loading and its effect on dose rates. Generally, the radial dose rates for uniform loading bound the dose rates for regionalized loading.

Confinement Evaluation

Regionalized loading allows higher heat emitting fuel (higher burnup fuel at shorter decay times) to be loaded into the HI-STORM cask. From a confinement perspective the newer, high burnup fuel in the center of the cask has an increased radionuclide inventory due to increased fission products. The radionuclide inventories for each of the MPC designs that allow regionalized loading was revised to ensure that bounding source terms are maintained. The resultant doses are presented in Table 7.3.2 through Table 7.3.4 in proposed Revision 11 of the TSAR (see Attachment 5). Additionally, Table 7.3.8 of proposed Revision 11 of the TSAR presents bounding doses for casks containing PWR and BWR fuel and compares them directly to the limits of 10CFR72.

Proposed Change No. 13

Certificate of Compliance, Appendix B, Table 2.1-1 (throughout):

a. Cooling time, burnup, and decay heat limits are presented by array/class designation instead of by cladding material.

b. The wording in the right side of the table for cooling time, burnup, and decay heat is made consistent.

c. Fuel assembly weights are clarified to include non-fuel hardware (PWR), channels (BWR), and damaged fuel canisters, as applicable.

d. The maximum allowed length for standard BWR fuel is increased from 176.2 inches (nominal) to 176.5 inches (nominal).

Reason and Justification for Proposed Change

a. With the addition of more fuel types and unique limits for certain Zircaloy clad fuel assemblies, the presentation format became too complex for users to follow. This change simplifies the presentation.

b. Editorial clarification.

c. The MPC has been analyzed with a maximum bounding weight assumed and divided among the total number of fuel storage cells. The user must ensure that all components loaded into a storage location, in total, do not exceed that limit. There is no need to distinguish among the components.

d. Customer feedback indicates some of their BWR fuel assemblies are longer than the current nominal limit of 176.2 inches. The new nominal length limit of 176.5 inches bounds these fuel assemblies and has been evaluated against the MPC height tolerance as well as growth of the limiting length assembly due to irradiation and thermal expansion and found to be acceptable. There is no impact on the structural, thermal, shielding, criticality, or confinement evaluations due to this change.

Proposed Change No. 14

Certificate of Compliance, Appendix B, Table 2.1-1, and new Table 2.1-8:

MPC-24, Items I.A and C, are revised; new Note 1 is added to Item I, and new Table 2.1-8 is added as shown in the attached marked-up CoC pages to allow storage of non-fuel hardware, including Burnable Poison Rod Assemblies (BPRAs)*, Thimble Plug Devices (TPDs)*, Control Rod Assemblies (CRAs), Axial Power Shaping Rods (APSRs) and similarly designed devices with different names. Non-fuel hardware is also proposed to be authorized for loading into MPC-24E, MPC-24EF, and MPC-32 and the same limits are specified for those MPC models later in Table 2.1-1.

Reason for Proposed Change

A large number of PWR plant fuel assemblies are currently stored in spent fuel pools with either BPRAs or TPDs as integral hardware to the assemblies. A smaller number of PWR assemblies are stored with CRAs or APSRs. This irradiated hardware must be authorized for dry storage with the assemblies to accommodate user needs (particularly for plants who wish to decommission their spent fuel pools) and is therefore proposed to be added to the authorized contents.

Justification for Proposed Change

Structural Evaluation

There is no effect on the structural evaluation because these changes do not change the fuel assembly geometry or weight used in the structural analyses. The limits on these parameters as stated elsewhere in the CoC fuel tables remain the same and fuel assemblies containing these components must meet these limits.

Thermal Evaluation

The non-fuel bearing hardware (i.e. BPRAs TPDs, CRAs, and APSRs) becomes activated as a result of in-core irradiation. In the dry cask storage scenario, this hardware represents a Low Heat Emitting (LHE) source distributed over the length of the fuel assembly. The non-fuel hardware contribution to the total decay heat load burden of a cask is quite small.

The BPRAs, CRAs, and APSRs, when inserted in the fuel assemblies, displace the gas in the guide tubes and replace them with solid materials (neutron absorbers and metals) which conduct heat much more readily. As a result, dissipation of heat by the fuel assemblies is enhanced by the presence of these components. In the thermal evaluation supporting this amendment request, no credit was taken for this enhanced decay heat dissipation. Thus, the design basis heat load of the HI-STORM cask is conservatively unaltered by this proposed change. To conservatively compute a lower bound value for the permissible burnup and cooling time limits for storage in the HI-STORM cask, the limiting fuel type for the class of PWR fuel (i.e., the one with the highest uranium mass) is utilized. In the CoC, a requirement is specified to comply with these burnup and cooling time limits. In addition, each assembly proposed for storage must be confirmed to have a total heat emission rate less than the design maximum, including the fuel and any non-fuel hardware, as applicable.

The addition of this non-fuel hardware has two effects on the MPC cavity pressures. As discussed in the last paragraph, non-fuel hardware enhances heat dissipation, thus lowering fuel and MPC cavity fill gas temperatures. The gas volume displaced by the mass of the non-fuel hardware lowers the cavity free volume. These two effects, namely, temperature lowering and free volume reduction, have opposing influences on the MPC cavity pressure. The first effect lowers the gas pressure while the second effect raises it. In the HI-STORM thermal analysis, the computed temperature field (with non-fuel hardware *excluded*) provides a conservatively bounding thermal response of the HI-STORM cask. The MPC cavity free space was computed based on displacement by the heaviest fuel (bounding weight) with non-fuel hardware *included*. Thus, the previously computed MPC cavity pressure results remain conservative with respect to gas temperature and free space as affected by the changes proposed in this amendment.

PWR fuel assemblies with BPRAs containing helium gas have been evaluated under the hypothetical accident condition where 100% of the BPRAs rupture, releasing all of the contained helium into the MPC cavity. The maximum helium backfill pressure TS limit for the PWR MPCs is adjusted appropriately so that the resultant post-accident MPC cavity pressure, including BPRA gas release, is limited to an acceptable value, within the design pressure of the MPC. Appropriate discussion has been added to proposed Revision 11 TSAR Chapters 4 and 11 (see Attachment 5).

Shielding Evaluation –BPRAs and TPDs

Burnable Poison Rod Assemblies (including Wet Annular Burnable Absorbers and other similarly designed devices with different names) and Thimble Plug Devices (including orifice rod assemblies, guide tube plugs, and other similarly designed devices with different names) are an integral, yet removable, part of a large portion of PWR fuel. The TPDs are not used in all assemblies in a reactor core, but are re-used from cycle to cycle. Therefore, these devices can achieve very high burnups. In contrast, BPRAs are burned with a fuel assembly in core and are not reused. In fact, many BPRAs are removed after one or two cycles before the fuel assembly is discharged. Therefore, the achieved burnup for BPRAs is not significantly different than fuel assemblies.

TPDs are made of stainless steel and contain a small amount of Inconel. These devices extend down into the plenum region of the fuel assembly but do not extend into the active fuel region with the exception of the Westinghouse 14x14 water displacement guide tube plugs. Since these devices are made of stainless steel, there is a significant amount of Co-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 two to 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. 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 of the HI-STORM TSAR, Rev. 10 in order to account for the reduced flux levels above the fuel assembly. The total curies of cobalt and the decay heat load 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 zero burnup 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 fuel containing BPRAs and TPDs (Westinghouse and B&W 14x14 through 17x17) found in TSAR references [5.2.5] and [5.2.7] listed in Section 5.6 of the TSAR to determine the TPD and BPRA which produced the highest Co-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 single hypothetical BPRA. The masses of this TPD and BPRA are listed in Table 5.2.30 of the proposed Revision 11 of the HI-STORM TSAR (see Attachment 5). As mentioned above, TSAR reference [5.2.5] describes the Westinghouse 14x14 water displacement guide tube plug as having a steel portion that 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 Co-60 source from the BPRA and TPD were specified: 50 Curies for each TPD, and 831 Curies for each BPRA. Table 5.2.31 of the proposed Revision 11 of the HI-STORM TSAR 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 the BPRAs and TPDs themselves. These burnup and cooling times assure that the Co-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 Co-60 source for a given cooling time actually decreases as the burnup continues to increase. This is due to a decrease in the Co-60 production rate as the initial Co-59 impurity is depleted. Conservatively, a constant cooling time has been specified for burnups from 180,000 to 630,000 MWD/MTU for the TPDs.

Shielding Evaluation – CRAs and APSRs

Control Rod Assemblies (CRAs) and Axial Power Shaping Rods (APSRs) are an integral portion of many PWR fuel assemblies going into dry storage. 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 varies 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. Silver-Indium-Cadmium (Ag-In-Cd) is possibly the most common absorber, although B_4C in aluminum is used, and hafnium has also been used. Ag-In-Cd 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 Ag-In-Cd CRAs are the bounding CRAs.

APSRs are used to flatten the axial 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 TSAR reference [5.2.5], the black APSRs have 36 inches of Ag-In-Cd 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 specified in Appendix B to the CoC.

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 Ag-In-Cd 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 Ag-In-Cd 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, of the proposed Revision 11 of the TSAR, for the CRAs and Table 5.2.33, of the proposed Revision 11 of the TSAR, for the APSR. The masses of the materials listed in these tables were determined from a review of TSAR reference [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 TSAR reference [5.2.5] because the values in the reference were adjusted to the lengths shown in the tables.

Configuration 1: CRA and APSR

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 of proposed Revision 11 of the TSAR present the source terms 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 Ag-In-Cd is in the range of 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. The very large 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 was similar to the assumed value for steel, 0.8 gm/kg, this source would decrease by approximately a factor of 5.8.

Shielding Summary

Section 5.4.6 of proposed Revision 11 of the HI-STORM TSAR provides the dose rate increase due to the inclusion of BPRAs, TPDs, CRAs, and APSRs. The data in this section indicate that BPRAs result in the highest dose rate increase on the radial surfaces of the cask while the APSRs result in the largest dose rate increase in the bottom of the cask. The increase in the dose rates at the bottom of the cask will not significantly affect occupational exposure. Therefore, the additional dose rate from the BPRAs was included in the design basis analysis presented in Section 5.1 and in the dose rates calculated in Section 5.4 of the proposed Revision 11 of the HI-STORM TSAR found in Attachment 5. The occupational exposure estimates provided in Chapter 10 of the TSAR were also revised to include the dose rate contribution from BPRAs. These new values can be found in proposed Revision 11 of Chapter 10 in Attachment 5. The controlled area boundary dose rate analysis provided in Chapter 5 of Revision 10 of the TSAR was not revised to include the effect of BPRAs because this analysis had been performed with a bounding burnup and cooling time of 52.5 GWD/MTU and 5 year cooling.

In conclusion, the shielding analysis has been revised to include the additional dose rate from non-fuel hardware. While the dose rates around the HI-TRAC have increased as a result of including this non-fuel hardware, the safety of the system has not been compromised.

Criticality Evaluation

For MPCs filled with pure water, the reactivity of any PWR assembly with non-fuel hardware inserted into the guide tubes is bounded by (i.e. lower than) the reactivity of the same assembly without the inserts. This is due to the fact that the inserts reduce the amount of moderator, while the amount of fissile material remains unchanged. In the presence of soluble boron in the water, especially for higher soluble boron concentrations, it is possible that the non-fuel hardware in the PWR assembly results in an increase of reactivity. This is due to the fact that the insert not only replaces water, but also the neutron absorber in the water. To account for this effect, analyses with and without non-fuel hardware in the assemblies have been performed for higher soluble boron concentrations (see Tables 6.4.6 and 6.4.10 of Proposed Rev. 11 of the TSAR). The highest reactivities for either case are used as the basis of the criticality evaluation.

Proposed Change No. 15

Certificate of Compliance, Appendix B, Table 2.1-1:

Item II.A.2 is revised to authorize a broader range of BWR damaged fuel, beyond the currently authorized Dresden Unit 1 and Humboldt Bay damaged fuel. The additional damaged fuel must be loaded into the new generic BWR DFC for loading into the MPC-68. Further, the damaged fuel is only authorized for loading into the 16 peripheral fuel storage locations, called out numerically in revised Item II.B.3. Damaged fuel assemblies meeting the same specifications are also proposed to be authorized for loading into the MPC-68FF as discussed later in this section.

Reason for Proposed Change

Most users have at least some fuel assemblies destined for dry storage that would be classified as damaged fuel assemblies in accordance with the CoC. The current CoC only authorizes damaged fuel from Dresden Unit 1 and Humboldt Bay for storage. The CoC needs to be expanded to accommodate customer needs.

Justification for Proposed Change

Structural Evaluation

The only structural requirements on the contents of a BWR or PWR fuel basket are that the total weight per cell does not exceed the design basis weight (700 lbs for BWR assemblies and 1,680 lbs for PWR assemblies), and that loading of these assemblies does not alter the temperature limits used in the design basis analyses. The damaged fuel assemblies covered by this change, together with the appropriate DFC, satisfy these restrictions, so that no additional structural evaluation is necessary.

Thermal Evaluation

The thermal performance characteristics of the most heat resistive Zircaloy and stainless steel clad BWR fuel assemblies, encased in the proposed DFC design, have been evaluated in support of this amendment. The interruption of thermal radiation heat exchange between the fuel assembly and the fuel basket by the DFC renders the DFC configuration more restrictive than the non-DFC configuration. The thermal performance characteristics of MPC-68s loaded entirely with fuel assemblies in BWR DFCs were evaluated, using the same methods employed to evaluate the previously approved MPC-68 with Dresden Unit 1 and Humboldt Bay damaged fuel, and appropriate decay heat loads determined. It is noted that this amendment only requests loading of 16 BWR DFCs, so the thermal evaluations of MPCs completely loaded with fuel in DFCs is highly conservative.

Shielding Evaluation

See the shielding evaluation for Proposed Change Number 10.

Criticality Evaluation

Criticality calculations have been performed for an MPC-68 loaded with intact and damaged fuel/fuel debris (up to 16 damaged fuel assemblies placed in DFCs) and maximum enrichments of up to 4.0 wt% ^{235}U for the damaged fuel/fuel debris and up to 3.7 wt% ^{235}U for the intact fuel. The calculations use a bounding approach to account for the possible wide variation of fuel distribution inside the DFC, based on the analysis of arrays of bare fuel rods. Also, typical damaged fuel conditions such as missing rods or collapsed assemblies are analyzed for selected array/classes. The analyses are presented in Section 6.4.4.2 of proposed Revision 11 of the TSAR (see Attachment 5). The maximum calculated k_{eff} is 0.9328, which demonstrates that the cask system is in compliance with the regulatory requirement of $k_{eff} < 0.95$ for all BWR fuel array/classes.

Proposed Change No. 16*

Certificate of Compliance, Appendix B, Table 2.1-1:

New Items II.A.5 and III.A.7 are added to Table 2.1-1 for MPC-68 and MPC–68F as shown in the attached marked-up pages of the CoC table to allow storage of one Dresden Unit 1 (D-1) Thoria Rod Canister in these MPC models. Drawings of the D-1 Thoria Rod Canister were provided in LAR 1008-1 submitted to the NRC in November, 1999 for the HI-STAR 100 System (Docket 72-1008). Figure 2.1.2A is added to the TSAR showing key dimensions and major fabrication details for the Thoria Rod Canister (see Attachment 5). Conforming revisions are also made to Appendix B, Items II.B and III.B.

Reason for Proposed Change

Dresden Unit 1 needs to place one Thoria Rod Canister into dry storage to support plant decommissioning.

Justification for Proposed Change

Structural Evaluation

The Dresden Unit 1 Thoria Rod Canister has been structurally evaluated by Holtec International and found to meet all required design requirements for storage in the HI-STORM 100 system. The details of this evaluation are contained in proposed Revision 11 TSAR Appendix 3.AR, included in Attachment 5 to this letter. All required safety margins are greater than zero or, in other words, the factors of safety are greater than 1.0.

Thermal Evaluation

The Thoria Rod Canister is designed to hold a maximum of 20 fuel rods arrayed in a 5x4 configuration. Eighteen rods are actually in the canister. The fuel rods contain a mixture of enriched UO_2 and thorium oxide in the fuel pellets. The fuel rods were originally constituted as part of an 8x8 fuel assembly and used in the second and third cycle of Dresden-1 operation. The maximum fuel burnup of these rods is quite low (< 16,000 MWD/MTIHM). The Thoria Rod Canister internal design is a honeycomb structure formed from 12 gage stainless steel plates. The rods are loaded in individual square cells and thus are isolated from each other by the cell walls. The few number of rods (18 per assembly) and very low burnup of fuel stored in these Dresden-1 canisters render them as miniscule sources of decay heat. The canister all-metal internal honeycomb construction serves as an additional means of heat dissipation in the fuel cell space. In accordance with preferential fuel loading requirements imposed in the Approved Contents section of Appendix B to the CoC, low burnup fuel is required to be loaded toward the basket periphery (i.e., away from the hot central core of the fuel basket). All these considerations provide ample assurance that these fuel rods will be stored in a benign thermal environment and therefore remain protected during long-term storage.

Shielding Evaluation

The Dresden Unit 1 Thoria Rod Canister contains 18 thoria rods that have obtained a relatively low burnup, 16,000 MWD/MTIHM. These rods were removed from two 8x8 fuel assemblies that contained 9 rods each. The irradiation of thorium produces an isotope that is not commonly found in depleted uranium fuel. Th-232, when irradiated, produces U-233. The U-233 can undergo an (n,2n) reaction that produces U-232. The U-232 decays to produce Tl-208 that produces a 2.6 MeV gamma during beta decay. This results in a significant source in the 2.5-3.0 MeV range that 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/MTIHM and a cooling time of 18 years. Table 5.2.36 of proposed Revision 11 of the HI-STORM TSAR (Attachment 5) describes the 8x8 fuel assembly that contains the thoria rods. Table 5.2.37 and 5.2.38 of proposed Revision 11 of the HI-STORM TSAR shows 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 TSAR Tables 5.2.7 and 5.2.18 clearly indicates that the design basis source terms bound the thoria rod 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.

It is obvious that the neutron spectrum from the 6x6 fuel assembly bounds the thoria rod neutron spectra 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 transfer cask and the HI-STORM overpack was estimated conservatively assuming an MPC-68 filled with 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 17% 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 only one thoria rod canister is proposed to be authorized for storage in the HI-STORM 100 System clearly demonstrates that the Thoria Rod Canister is acceptable for storage in the MPC-68 or the MPC-68F.

Criticality Evaluation

The Thoria Rod Canister is similar to a DFC with an internal separator assembly containing 18 fuel rods. The configuration is illustrated in proposed Revision 11 TSAR Figure 6.4.19 (see Attachment 5). The k_{eff} value for an MPC-68/68F filled with Thoria Rod Canisters is calculated to be 0.18. 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 outside diameter is 0.412"), and the absorption in the separator assembly. Together with the maximum k_{eff} values listed in TSAR 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 $k_{eff} < 0.95$.

Confinement Evaluation

The HI-STORM confinement analyses have been revised to account for several new isotopes associated with the Thoria Rod Canister. These isotopes (Bi-212, Pb-212, Po-216, Ra-224, Rn-220, Th-228 and U-232) had a negligible effect on the resulting doses because only one Thoria Rod Canister is authorized for loading in an MPC-68 or –68F with 67 other design basis BWR assemblies. Therefore, the Thoria Rod isotopes are not included in the presentation of the confinement analysis inputs or results in the TSAR.

Proposed Change No. 17*

Certificate of Compliance, Appendix B, Table 2.1-1

New Items II.D and III.D are added as shown in the attached marked-up CoC pages to authorize Dresden Unit 1 fuel assemblies containing up to one antimony-beryllium neutron source in the assembly lattice for storage.

Reason for Proposed Change

Dresden Unit 1 needs to place fuel assemblies containing antimony-beryllium neutron sources into dry storage to support plant decommissioning.

Justification for Proposed Change

Structural Evaluation

The structural evaluation is not affected because the fuel assembly parameters used in the design basis structural evaluations are not affected by this change. The neutron sources have no impact on component temperatures or fuel assembly size and weight.

Thermal Evaluation

The substitution of antimony-beryllium sources in a fuel assembly in lieu of heat emitting fuel rods is bounded by the existing thermal analyses, which assume decay heat production from the replaced fuel rods.

Shielding Evaluation

Dresden Unit 1 has antimony-beryllium neutron sources that are placed in the water rod location of their fuel assemblies. These sources are steel rods that contain a cylindrical antimony-beryllium source that 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, that decays by beta decay with a half-life of 60.2 days, produces a gamma of energy 1.69 MeV that 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 required 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 that 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 that is produced in the MPC from neutron activation by 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 of Revision 10 of the HI-STORM TSAR. A single fuel rod was removed and replaced by a guide tube. In order to determine the amount of Sb-124 that is 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 likely encased in another material that 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 that 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 that 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 17.1 below, 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 minimum burnup assuming the Sb-Be source was in the reactor for the entire 18-year life of Dresden Unit 1. The cooling time was assumed to be 18 years that is the minimum cooling time for Dresden Unit 1 fuel. The source from the steel is 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.

Table 17.1
Comparison of Neutron Source per Inch per Second for Design Basis 7x7 Fuel and Design Basis Dresden Unit 1 Fuel

Assembly	Active fuel length (inches)	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 Rev. 11 HI-STORM TSAR 40 GWD/MTU and 5 year cooling
6x6 design basis	110	2.0E+5	2.6E+5	Table 5.2.18 Rev. 10 HI-STORM TSAR
6x6 design basis MOX	110	3.06E+5	3.66E+5	Table 5.2.23 Rev. 10 HI-STORM TSAR

Criticality Evaluation

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 k_{eff} 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 also not lead to an increase of reactivity.

Proposed Change No. 18

Certificate of Compliance, Appendix B, Table 2.1-1

Items III.A.1.f, g, and h are revised as shown in the attached CoC markups to correct these dimensional limits to match the dimensions for Zircaloy fuel assembly array/classes 6x6A, 6x6C, 7x7A, and 8x8A (Dresden Unit 1 and Humboldt Bay). Only these array/classes (and 6x6B MOX fuel) are authorized for loading into the MPC-68F. This is simply an editorial change because fuel assemblies exceeding the correct dimensional limits would not be able to be inadvertently loaded as they would not fall into the above-mentioned array/classes.

Reason and Justification for Proposed Change

Editorial corrections.

Proposed Change No. 19

Certificate of Compliance, Appendix B, Table 2.1-1

New Item IV is added to the table for MPC-24E. See also, proposed Change Number 29.

Reason for Proposed Change

The MPC-24E provides for storage of higher enriched fuel than the MPC-24 through the optimization of the storage cell layout. In addition, storage of damaged PWR fuel assemblies in generic PWR DFC is authorized. This change is required to meet customers' needs for storage of higher enriched fuel and damaged fuel. The MPC-24E has been analyzed for storage of two ranges of enrichment for PWR fuel. The lower of the two ranges has been analyzed with unborated water in the MPC during wet loading and unloading operations and the higher range has been analyzed with credit taken for soluble boron in the MPC water (see associated changes to Table 2.1-2 and Proposed Change Number 3).

Justification for Proposed Change

The MPC-24E is a very close variant of the previously approved MPC-24. Holtec's engineers and analysts have taken advantage of optimizing the fuel storage cell configuration, flux trap sizes, and ^{10}B loading in the Boral, while still meeting subcriticality requirements. The basic honeycomb basket structure remains unchanged. The structural and thermal characteristics of the basket are virtually the same as the MPC-24. There is an effect on the confinement analysis due to the addition of damaged fuel. A detailed discussion of this change is provided below, arranged by technical discipline.

Structural Evaluation

A finite element model of the MPC-24E fuel basket was prepared in the same manner that was used for the previously approved MPC-24 and MPC-68 fuel baskets. The analyses of the MPC-24E fuel basket under applied inertia loads, simulating a handling accident, have been carried out to obtain primary stresses in the fuel basket structure and in the MPC shell. The safety factors, after applying the appropriate dynamic amplifier, exceed 1.0 and are reported in the proposed TSAR revision in appropriate tables in Chapter 3, Subsection 3.4. Text in Chapter 3 of the TSAR has been appropriately modified to reflect the addition of this new fuel basket. All other structural analyses currently approved have been reviewed to ensure that the bounding loads used as input for the specific structural analyses remained bounding. The bounding weights used as input for the TSAR analyses were not changed by the addition of this new basket; therefore, previously reported safety factors in the TSAR are not altered by this new fuel basket. See Attachment 5 for proposed TSAR changes.

Thermal Evaluation

With respect to thermal performance, the MPC-24E configuration is slightly different (symmetric basket layout) from the previously approved MPC-24, but employs the same general construction (integral honeycomb basket) and the same heat rejection mechanisms. The thermal performance of the MPC-24E design has been evaluated, in support of this amendment request, using the analysis methods employed to determine the performance of the previously approved MPC-24 and MPC-68. The substantial conservative assumptions embedded in the evaluations of the MPC-24 and MPC-68 designs have also been incorporated in the evaluations of the MPC-24E. Allowable decay heat loads have been determined for design-basis (DB) intact Zircaloy clad, damaged Zircaloy clad, and stainless steel clad fuel that ensure safe long-term storage of SNF in the MPC-24E. The HI-STORM temperature field for the MPC-24E loaded with design-basis heat emitting fuel was calculated and is reported in proposed revisions to HI-STORM TSAR Chapter 4 (see Attachment 5).

Shielding Evaluation

From a shielding perspective, the new MPC-24E is identical to the MPC-24 and therefore was not explicitly analyzed. The different fuel cell pitch in the MPC-24E, compared to the MPC-24, will have little impact on the dose rates outside the overpack. In addition, all of the steel fuel cell walls in the MPC-24E are 5/16 inch thickness and provide somewhat more shielding compared to the MPC-24 (which utilizes both 9/32 and 5/16 inch walls). The analysis of the MPC-24 in Chapter 5 of the proposed Revision 11 of the HI-STORM TSAR conservatively bounds the allowable contents for both the MPC-24 and the MPC-24E.

Criticality Evaluation

In order to increase the maximum permissible fuel enrichment for the MPC-24E compared to the MPC-24, the following changes were introduced into the MPC-24E:

- The fuel storage cells and flux traps are arranged in a fully symmetric manner, which allows moving some cells further away from the center of the basket. This results in increased flux traps in some areas of the basket.

- The ^{10}B loading of the Boral is increased from 0.0267 (minimum) to 0.0372 g/cm2 (minimum). This requires a change in the Boral thickness from 0.082 inches to 0.101 inches.

- The cell pitch is slightly increased.

Additionally, four of the peripheral cells have an increased cell ID to accommodate PWR Damaged Fuel Containers. This results in decreased flux traps for these cells.

Overall, this design allows an increase in the maximum permissible fuel enrichment of 0.4 wt% ^{235}U for most fuel classes, while maintaining the same level of margin toward the regulatory requirement of $k_{eff} < 0.95$. The maximum k_{eff} for the bounding assembly in each class is listed in Table 6.1.3 in Section 6.1 of the proposed Revision 11 of the TSAR (see Attachment 5).

Additionally, the MPC-24E is analyzed with credit taken for soluble boron present in the water during wet loading and unloading operations. With a minimum soluble boron concentration in the water of 300 ppmb, a maximum enrichment of 5.0 wt% ^{235}U for all assembly classes is permissible. To ensure that the actual k_{eff} is always below the maximum calculated k_{eff}, the following additional conservative assumptions are applied in the calculations with soluble boron.

- The pellet to clad gap is assumed to be flooded with pure, unborated water.

- The water above and below the active regions is assumed to be pure, unborated water.

The maximum k_{eff} for the bounding assembly in each class for this condition is listed in Table 6.1.4 in Section 6.1 of the Proposed Rev. 11 of the TSAR.

<u>Confinement Evaluation</u>

From a confinement perspective, the evaluation for the MPC-24 and MPC-24E are identical with the exception of the minimum free volume in the MPC cavity. The MPC-24E minimum free volume is slightly less than the MPC-24 due to increased thickness of the basket cell walls and the presence of more basket cell walls. This increases the concentration of radionuclides slightly due to the smaller dilution volume. The resultant doses from the MPC-24E are presented in TSAR Table 7.3.2 in proposed Revision 11 of the TSAR and bound the doses from the MPC-24 (see Attachment 5).

Proposed Change No. 20

Certificate of Compliance, Appendix A, Table 3-1 and Appendix B, Table 2.1-1

New Item V is added to the table for MPC-32.

Reason for Proposed Change

The MPC-32 allows users to place PWR fuel into dry storage using one third fewer casks due to its increased storage capacity over the MPC-24 and MPC-24E. Fewer casks to load decreases the probability of cask handling mishaps, reduces the overall occupational exposure for the fuel loading campaign, and reduces customer cost.

Justification for Proposed Change

The MPC-32 basket design is very similar to the previously approved BWR MPC-68. However, unlike the MPC-24 series PWR basket, no flux traps are used. As such, credit for soluble boron is taken in the MPC-32 criticality analyses for all authorized fuel enrichments. Two ranges of enrichment, with two separate minimum boron concentration requirements have been analyzed (see associated changes to Table 2.1-2 and Proposed Change Number 3). A detailed discussion of this change is provided below, arranged by technical discipline.

Structural Evaluation

The structural analysis of the MPC-32 was considered in the initial versions of the HI-STAR TSAR (Docket 72-1008). The review of the structural analysis of the MPC-32 fuel basket was performed by the NRC staff and all structural questions from the NRC staff resolved. Prior to final approval of the HI-STAR TSAR, however, the MPC-32 basket was removed from the submittal to permit final resolution of some outstanding non-structural issues without a delay in the CoC approval process. The MPC-32 was also removed from the HI-STORM TSAR submittal at the same time.

The re-introduction of the MPC, from a structural point of view, required only the addition back into the text and appendices previously reviewed calculations and results. To that end, all TSAR text, tables, and appendices have been reviewed and updated to include the MPC-32 input data and structural results. The finite element model of the MPC-32 fuel basket was originally prepared at the same time and in the same manner as the currently reviewed and approved MPC-24 and MPC-68 fuel baskets. The analyses of the MPC-32 fuel basket under applied inertia loads, simulating a handling accident was carried out to obtain stresses in the fuel basket structure and in the MPC shell. The safety factors, previously reviewed, after applying the appropriate dynamic amplifier, exceed 1.0 and are re-introduced into the TSAR document in the appropriate tabular form. Since the MPC-32 was (and still is) the heaviest MPC when fully loaded, there has been no change in the bounding loads used as input for other calculations. Appropriate text and tables in Section 3 of the TSAR have been updated to reflect the presence of this new fuel basket (see Attachment 5). The changes to the MPC-32 drawings as described in Section III of this attachment, were reviewed and found to be insignificant with respect to the structural evaluation. No new structural evaluations have been introduced into the TSAR as a result of restoring the MPC-32.

Thermal Evaluation

With respect to thermal performance, the MPC-32 design for PWR fuel is akin to the previously approved MPC-68 for BWR fuel in that the same general construction and the same heat rejection mechanisms are present. The thermal performance of the MPC-32 design has been evaluated, in support of this amendment, using the analysis methods employed to determine the performance of the previously approved MPC-24 and MPC-68. The substantial conservative assumptions embedded in the evaluations of the MPC-24 and MPC-68 designs have also been incorporated in the evaluations of the MPC-32. Allowable decay heat loads have been determined for design-basis (DB) intact Zircaloy and stainless steel clad fuel that ensure safe long-term storage of SNF in the MPC-32. The HI-STORM temperature field for the MPC-32 loaded with design-basis heat emitting fuel was calculated and is reported in proposed revisions to HI-STORM TSAR Chapter 4 (see Attachment 5). This analysis demonstrates substantial cladding thermal margins.

Shielding Evaluation

The MPC-32 was explicitly analyzed in Chapter 5 of the proposed Revision 11 to the HI-STORM TSAR (see Attachment 5). The dose rates around the HI-STORM overpack were conservatively analyzed at a burnup of 45,000 MWD/MTU and 5 year cooling for the MPC-32. Only the 100-ton HI-TRAC was analyzed with the MPC-32 since those dose rates bound the dose rates from the 125-ton HI-TRAC. The burnup and cooling times used for the HI-TRAC analysis are consistent with the burnup and cooling times specified in the proposed changes to the Approved Contents section of Appendix B to the CoC. Since the specified burnups and cooling times for the MPC-32 are considerably lower than the MPC-24, the MPC-

24 was still used for the site-boundary evaluation to demonstrate compliance with 10CFR72.104. In addition, because of the differences in burnup and cooling times between the MPC-32 and the MPC-24, the radial dose rates from the MPC-24 are typically higher than for the MPC-32. Therefore, the MPC-24 was still used for the dose rate evaluations in Chapter 10.

Sections 5.1 and 5.4 of proposed Revision 11 of the HI-STORM TSAR report the calculated dose rates for the MPC-32 and Section 5.2 reports the source terms used for the MPC-32 evaluations.

Criticality Evaluation

The MPC-32 is analyzed with credit for soluble boron present in the water during wet loading and unloading operations. Two soluble boron concentrations are used in the analysis, 1900 ppmb and 2600 ppmb. With a minimum soluble boron concentration in the water of 1900 ppmb, a maximum enrichment of 4.1 wt% ^{235}U for all authorized fuel assembly array/classes is permissible. At 2600 ppmb, a maximum enrichment of 5.0 wt% ^{235}U for all authorized fuel assembly array/classes is permissible. Consistent with the analysis for the MPC-24E, the following additional conservative assumptions are applied to ensure that the actual k_{eff} is always below the maximum calculated k_{eff}.

- The pellet to clad gap is assumed to be flooded with pure, unborated water.
- The water above and below the active regions is assumed to be pure, unborated water.

The maximum k_{eff} for the bounding assembly in each class for the two soluble boron levels is listed in Tables 6.1.5 and 6.16 in Section 6.1 of the Proposed Rev. 11 of the TSAR (see Attachment 5).

Confinement Evaluation

The MPC-32 is explicitly analyzed in Chapter 7 of proposed Revision 11 of the HI-STORM TSAR. The radionuclide inventories were conservatively calculated assuming the design basis assembly at a burnup of 70,000MWD/MTU at a 5 year cooling time. The fuel specifications in the Approved Contents section of Appendix B to the CoC limit the fuel assembly burnup to well below 70,000 MWD/MTU for 5-years cooling time, ensuring that this inventory exceeds that of the actual fuel acceptable for loading into the MPC-32. The resultant doses are summarized in Table 7.3.3 of proposed Revision 11 of the TSAR (see Attachment 5).

Proposed Change No. 21

Certificate of Compliance, Appendix B, Table 2.1-1

New Item VI is added to the table for MPC-68FF.

Reason for Proposed Change

The MPC-68FF allows users to place BWR fuel debris into dry storage where this was previously not authorized beyond Dresden Unit 1 and Humboldt Bay Fuel. User feedback on fuel condition indicates that some fuel assemblies destined for dry storage would be classified as fuel debris in accordance with the CoC.

Justification for Proposed Change

The MPC-68FF combines the thickened top portion of the previously approved MPC-68F shell with the maximized ^{10}B loading in the Boral neutron absorbers of the standard MPC-68, to allow storage of a wide range of damaged BWR fuel or fuel debris, loaded into DFCs. A detailed discussion of this change is provided below, arranged by technical discipline.

Structural Evaluation

With the exception of the thickened top portion of the MPC shell, the MPC-68FF is identical to the previously approved MPC-68F. The thickening of the MPC shell is limited to the closure lid region, and has already been evaluated for structural integrity and approved as part of the HI-STAR 100 Part 71 SAR.

Thermal Evaluation

With the notable exception of the thickened top portion of the MPC shell, the MPC-68FF is identical to the previously approved MPC-68. The thickening of the MPC shell is limited to the closure lid region, and has no impact on the thermal performance of the MPC. The thermal performance of the MPC-68FF is, therefore, identical to that of the previously approved MPC-68.

Shielding Evaluation

The MPC-68FF is identical to the MPC-68 from a shielding perspective. Therefore the analysis of the MPC-68, including damaged fuel, in Chapter 5 of proposed Revision 11 of the HI-STORM TSAR is applicable for the MPC-68FF and no explicit analysis of the MPC-68FF is required.

Criticality Evaluation

The basket structure in the MPC-68FF is identical to the basket structure inside the MPC-68. More specifically, all dimensions relevant for the criticality analysis such as pitch, basket wall thickness and ^{10}B loading in the Boral are identical between MPC-68 and MPC-68FF. Therefore, all criticality results obtained for the MPC-68 are valid for the MPC-68FF and no further analyses are necessary. With regard to the analyses of damaged fuel and fuel debris, see Proposed Change No. 5, Holtec Generic BWR DFC.

Confinement Evaluation

The MPC-68FF confinement analysis is bounded by the evaluation of the MPC-68. The MPC-68FF has a larger MPC lid-to-shell weld, which is necessary for storage and transportation of fuel debris. The smaller MPC lid-to-shell weld in the MPC-68 conservatively overestimates the leakage rate from the MPC-68FF. Therefore, no separate explicit analysis of the MPC-68FF is required.

Proposed Change No. 22

Certificate of Compliance, Appendix B, Table 2.1-1

New Item VII is added to the table for MPC-24EF.

Reason for Proposed Change

The MPC-24EF allows users to place PWR fuel debris into dry storage where this was previously not authorized. User feedback on fuel condition indicates that some fuel assemblies destined for dry storage would be classified as fuel debris in accordance with the CoC.

Justification for Proposed Change

The MPC-24EF combines the thickened top portion of the previously approved MPC-68F shell with the newly proposed optimized MPC-24E fuel basket arrangement, to allow storage of a wide range of damaged PWR fuel or fuel debris, loaded into DFCs. A detailed discussion of this change is provided below, arranged by technical discipline.

Structural Evaluation

With the exception of the thickened top portion of the MPC shell, the MPC-24EF is identical to the proposed MPC-24E design discussed elsewhere in this section

(Proposed Change No. 19). The thickening of the MPC shell is limited to the closure lid region, and has already been evaluated for structural integrity and approved as part of the HI-STAR 100 Part 71 SAR.

Thermal Evaluation

With the notable exception of the thickened top portion of the MPC shell, the MPC-24EF is identical to the proposed MPC-24E design discussed elsewhere in this section (Proposed Change No. 19). The thickening of the MPC shell is limited to the closure lid region, and has no impact on the thermal performance of the MPC. The thermal performance of the MPC-24EF is, therefore, identical to that of the previously approved MPC-24. The evaluations of thermosiphon (MPC convection) and high burnup fuel for the MPC-24E are applicable to the MPC-24EF.

Shielding Evaluation

The MPC-24EF is identical to the MPC-24E from a shielding perspective. Therefore the shielding evaluation for Proposed Change No.19 is applicable here.

Criticality Evaluation

The basket structure in the MPC-24EF is identical to the basket structure inside the MPC-24E. More specifically, all dimensions relevant for the criticality analysis such as pitch, basket wall thickness and ^{10}B loading in the Boral are identical between MPC-24E and MPC-24EF. Therefore, all criticality results obtained for the MPC-24E are valid for the MPC-24EF and no further analyses are necessary. With regard to the analyses of damaged fuel and fuel debris, see Proposed Change No. 10, Holtec Generic PWR DFC.

Confinement Evaluation

The MPC-24EF confinement analysis is bounded by the evaluation of the MPC-24. The MPC-24EF has a larger MPC lid-to-shell weld, which is necessary for storage and transportation of fuel debris. The smaller MPC lid-to-shell weld in the MPC-24 conservatively overestimates the leakage rate from the MPC-24EF. Therefore, no separate explicit analysis of the MPC-24EF is required.

Proposed Change No. 23

Certificate of Compliance, Appendix B, Table 2.1-2:

Table 2.1-2 is revised to indicate two ranges of enrichment for PWR fuel to be stored in the MPC-24, MPC-24E, and MPC-24EF, with and without soluble boron in the MPC water (see also Proposed Change Numbers 5, 19 and 22).

Reason for Proposed Change

This change is proposed to allow higher enriched PWR fuel to be stored in the MPC-24, MPC-24E, and MPC-24EF with credit taken for soluble boron in the MPC water during wet loading and unloading operations.

Justification for Proposed Change

Criticality Evaluation

The MPC-24, MPC-24E, and MPC-24EF are all analyzed with credit taken for the soluble boron present in the water during wet loading and unloading operations. With a minimum soluble boron concentration in the water of 400 ppmb in the MPC-24 or 300 ppmb in the MPC-24E and MPC-24EF, a maximum enrichment of 5.0 wt% ^{235}U for all authorized fuel assembly array/classes is permissible. To ensure that the actual k_{eff} is always below the maximum calculated k_{eff}, the following additional conservative assumptions are applied in the calculations with soluble boron.

- The pellet to clad gap is assumed to be flooded with pure, unborated water.
- The water above and below the active regions is assumed to be pure, unborated water.

The maximum k_{eff} for the bounding assembly in each class for this condition is listed in Tables 6.1.2 (MPC-24) and 6.1.4 (MPC-24E and MPC-24EF) in Section 6.1 of proposed Revision 11 of the TSAR (see Attachment 5).

Proposed Change No. 24

Certificate of Compliance, Appendix B, Tables 2.1-2 and 2.1-3

Notes at the end of Tables 2.1-2 and 2.1-3 are revised/added as shown in the attached marked-up pages of the CoC. Pointers to these notes in the tables are also revised accordingly.

a. Note 3 in both tables is revised to clarify the intent.*

b. New Note 5 is added to Table 2.1-2.

c. New Note 6 is added to Table 2.1-2.

d. New Note 7 is added to Table 2.1-2

e. Note 4 in Table 2.1-3 is revised to increase the allowable weight percent of U-235 in the MOX rods of fuel assembly array/class 6x6B from 0.612 to 0.635. This note is also clarified to state that the weight percentages are to be calculated based on the total fuel weight (i.e., uranium oxide plus plutonium oxide).*

f. Notes 6 and 7 in Table 2.1-3 are swapped.

g. New Note 11 is added to Table 2.1-3.*

h. New Note 12 is added to Table 2.1-3.*

i. New Note 13 is added to Table 2.1-3.*

j. New Note 14 is added to Table 2.1-3.

Reason for Proposed Changes

a. As currently worded, it is unclear whether implementation of the tolerance offered by Note 3 allows adjusting the documented value of the as-delivered uranium mass for a fuel assembly, or adjusting the uranium mass limit specified in the table for comparison against users' fuel records. The intent is to adjust the uranium mass limit up (within the prescribed tolerance), as necessary, for comparison against users fuel records. This eliminates a potential poor practice of users adjusting uranium mass values found on fuel records.

b. This note is required to connect the enrichment level for PWR fuel to be loaded with the LCO for the required boron concentration in the MPC water.

c. This note is necessary to recognize that this array/class (representing only the Indian Point Unit 1 fuel assembly) includes two different fuel rod pitches.

d. This note is required due to the addition of damaged PWR fuel to the authorized contents.

e. User feedback indicates that there are fuel assemblies with MOX rods containing less than 1.578 weight percent fissile plutonium in natural uranium. To bound this situation, the uranium content in the MOX rods is increased slightly. The second change to Note 4 is proposed to improve clarity regarding the intent of the note.

f. These notes are swapped for consistency between the HI-STAR and HI-STORM for these same notes.

g. New Note 11 is proposed in response to user feedback that some assemblies may include non-fuel rods which are filled with zirconium or an alloy of zirconium material in lieu of water.

h. New Note 12 is proposed to be added for information on this new array/class.

i. New Note 13 is proposed to address a situation for the 9x9E fuel assembly array/class where one assembly type in the class (SPC 9x9-5) contains rods of different dimensions within the array.

j. New Note 14 addresses an issue related to the criticality analyses for stainless steel clad fuel from the LaCrosse plant.

Justification for Proposed Changes

a. None. The tolerance in the mass limit allowed by this note is in the current, approved CoC.

b. This note provides required logic for proper implementation of the CoC requirements.

c. The Indian Point Unit 1 (IP-1) fuel assembly is unique and has been analyzed separately to account for the two different pitches. Only the IP-1 assembly fits into this array/class. The criticality analysis for the IP-1 fuel assembly is performed based on the actual configuration with different pitches in different sectors of the assembly. However, as this assembly class does not bound any assemblies other than the IP-1, the pitches are not listed in Table 2.1-2.

d. The addition of damaged fuel and fuel debris in the PWR MPC-24E and MPC-24EF requires that the maximum enrichment of all fuel assemblies in the MPC be no greater than the maximum enrichment for the damaged fuel and fuel debris to preserve the assumptions of the criticality analyses. In the criticality analysis for damaged fuel in the generic PWR damaged fuel container, both intact and damaged fuel loaded into the same MPC are modeled at an enrichment of 4.0 wt% ^{235}U. The results of the analysis demonstrates that this ensures compliance with the regulatory requirement of k_{eff} <0.95. Therefore, limiting the maximum initial enrichment to 4.0 wt% for this loading situation is a requirement to ensure regulatory compliance.

e. All criticality calculations for the 6x6B (MOX) fuel assembly array/class were re-performed (see proposed revised TSAR Table 6.2.38 in Attachment 5). The change in reactivity for this change is small (less than 2σ). This demonstrates that the maximum k_{eff} remains below 0.95 with the increased uranium concentration. The second change is proposed for clarity.

f. Editorial.

g. Replacing water with a non-fissile zirconium material will reduce the amount of moderator without increasing the amount of fissile material. This results in a decreased reactivity. This situation is comparable to the overall reduction of water density analyzed in Section 6.4.2.1 of the TSAR, which shows a decrease of reactivity with decreasing water density (i.e. decreasing the amount of water in the cask). The existing calculations assuming water in the water rods are therefore bounding for rods with non-fissile material in lieu of water.

h. New fuel assembly array/class 8x8F represents a unique fuel assembly type known as the QUAD+. New Note 12 is proposed to describe the unique water rod features of this assembly.

i. The SPC 9x9-5 fuel assembly is configured with two types of fuel rods having differing dimensions. Accordingly, the criticality analyses have been performed considering the varying fuel rod dimensions in the SPC 9x9-5 fuel type. Bounding all fuel rods in the assembly with one set of rod dimensions is not feasible because of excessive dimensional overlap.

j. In the criticality analysis for damaged fuel in the generic BWR damaged fuel container, intact and damaged fuel/fuel debris loaded into the same MPC are modeled at enrichments of 3.7 wt% ^{235}U (intact) and 4.0 wt% ^{235}U (damaged/debris). The results of the analysis demonstrate that this ensures compliance with the regulatory requirement of k_{eff} <0.95. Therefore, limiting the maximum initial enrichment of the intact fuel to

3.7 wt% for this loading situation is a requirement to ensure the assumptions of the criticality analyses are preserved.

Proposed Change No. 25*

Certificate of Compliance, Appendix B, Tables 2.1-2 and 2.1-3 :

The maximum allowed design initial uranium masses for selected fuel assemblies are increased as shown in the marked-up CoC tables. This affects PWR fuel assembly array/classes 14x14A, 14x14B, 14x14C, 15x15A, 16x16A, 17x17A, 17x17B, and 17x17C in Table 2.1-2 and BWR fuel assembly array/classes 6x6A, 6x6B, 6x6C, 8x8B, 8x8C, 8x8D, 8x8E, 9x9A, 9x9B, 9x9C, 9x9D, 9x9E, 9x9F, 10x10A, 10x10B, and 10x10C in Table 2.1-3.

Reason for Proposed Changes

To respond to user feedback describing certain fuel assemblies which have uranium masses slightly above the specified limit (including the tolerance allowed by Note 3 included with Tables 2.1-2 and 2.1-3) for the applicable fuel assembly array/class. These changes are required to ensure users can load all of the fuel they plan to place into dry storage.

Justification for Proposed Changes

Structural Evaluation

There is no effect on the existing structural evaluation. The increased uranium masses do not cause an increase in the overall assembly weight limits in the CoC. These weights (or greater) were used in the structural evaluation. Since the allowed assembly weights are not being changed, the structural evaluation is unaffected.

Thermal Evaluation

There is no effect on the existing thermal evaluation. This is because the allowed heat load for the cask is computed based on the heat transfer characteristics of the cask system and permissible peak cladding temperatures. The increase in uranium mass does not impact any assumption made in determining the heat transfer characteristics of the cask system.

Shielding Evaluation

The uranium mass limit is a value that is determined from the shielding analysis. An increase in the mass of uranium will result in an increase in the neutron and gamma source term and decay heat load for a specified burnup and cooling time. The current CoC developed from the analyses in Revision 10 of the HI-STORM TSAR provides some margin between the analyzed mass of uranium and the approved mass of uranium as listed in the CoC. The allowable burnup and cooling times in the CoC were developed by comparing the calculated decay heat for the design basis assemblies to the allowable decay heat load as determined in the thermal analysis. The decay heat values that are compared against the limits were calculated using the mass of uranium listed in Chapter 5 of the HI-STORM TSAR for the design basis fuel assemblies. Since a lower mass of uranium will result in a lower decay heat, it is conservative, and provides margin, to specify the allowable mass of uranium in the current CoC for the design basis fuel assemblies (B&W 15x15 and 7x7) lower than the values analyzed in TSAR Chapter 5.

As discussed in Section 5.2.5 of the HI-STORM TSAR Revision 10, the design basis assembly was chosen by comparing the source terms for many different types of assemblies. All of the assemblies were shown to have a lower source term than the design basis fuel assemblies. For additional conservatism, the mass of uranium specified in the current CoC for these non-design basis fuel assemblies is also specified lower than the mass used in the comparison in Chapter 5 of TSAR Revision 10. This level of conservatism is unnecessary since the decay heat load used to determine the allowable burnup and cooling times for all assemblies was the decay heat load from the design basis fuel assemblies. Therefore, there was already a significant amount of conservatism for the non-design basis fuel assemblies included by using the design basis decay heat to determine the allowable burnup and cooling times. Section 5.2.5.3 of Revision 10 of the HI-STORM TSAR provides an indication of the level of conservatism associated with using the design basis decay heat for the non-design basis fuel assemblies.

The proposed change in the CoC is to increase the mass of uranium for the non-design basis fuel assemblies up to the value that was used in the analysis in Chapter 5 of the HI-STORM TSAR to determine the design basis fuel assembly. In order to permit a slightly larger increase in the uranium mass loadings relative to Revision 10 of the HI-STORM TSAR, the analysis in Sections 5.2.5.2 and 5.2.5.3, specifically Tables 5.2.26 and 5.2.28, has been modified to use a slightly larger uranium mass loading for the 8x8, 9x9, and 10x10 assemblies. As mentioned above, this change eliminates unnecessary over-conservatism while still maintaining a significant degree of conservatism and margin for the non-design basis fuel assemblies. The design basis fuel assemblies and the allowable mass loading for the design basis fuel assemblies remains unchanged. Therefore, the proposed change does not affect the shielding analysis presented in Revision 10 of the HI-STORM TSAR. Additional clarification has been added to the proposed Revision 11 of the HI-STORM TSAR to discuss this issue (see Attachment 5).

Criticality Evaluation

The criticality analyses are not affected by the proposed changes to the maximum allowed design uranium masses shown in the Certificate of Compliance (CoC). The uranium mass limits in the CoC are determined from the shielding analysis, and are specified as bounding values for groups of fuel classes (e.g. all B&W 15x15). The criticality analyses are based on an independent bounding assumption of a fuel stack density of 96.0% of the theoretical fuel density of 10.96 g/cm^3. The fuel stack density is approximately equal to 98% of the pellet density. Therefore, while the pellet density of some fuels might be slightly greater than 96% of theoretical, the actual stack density will be less. For some fuel classes, this density assumption results in a uranium mass for the criticality analyses that is below the value shown in the CoC. However, this only indicates the conservatism of the shielding analysis for these classes. The criticality analyses are still valid and bounding for all classes, due to the density assumption stated above, which is valid for current and future fuel assemblies.

Confinement Evaluation

As described in the shielding evaluation, the values of uranium mass used in the shielding analyses have not changed. These proposed changes simply increase the allowed uranium masses for non-design basis fuel assemblies to those used in the analysis for the design basis fuel assembly. The source terms used in the confinement analyses were taken from the design basis source terms used in the shielding analyses. Therefore, the existing confinement evaluation is still bounding for the proposed new uranium mass limits.

Proposed Change No. 26*

Certificate of Compliance, Appendix B, Tables 2.1-2 and 2.1-3:

Certain fuel assembly parameter limits are revised as shown in the attached marked-up CoC tables. This affects PWR fuel assembly array/class 14x14C in Table 2.1-2 and BWR fuel assembly array/classes 6x6A, 6x6B, 7x7A, 7x7B, 8x8A, 8x8B, 8x8D, 9x9B, 9x9D, 9x9E, 9x9F, and 10x10C in Table 2.1-3.

Reason for Proposed Changes

To respond to user feedback describing certain fuel assemblies that have parameters outside of the limits in the existing CoC Tables. These changes are required to ensure users can load all of the fuel they plan to place into dry storage.

Justification for Proposed Changes

Structural Evaluation

The proposed changes to fuel parameter limits for some of the existing fuel assembly array/classes have no impact on the structural evaluation because the design basis weights used in the analyses (and provided as limits elsewhere in the CoC) are not changed, the design basis temperatures are not changed, and the geometry of the fuel assemblies (also limited by the CoC) are not changed.

Thermal Evaluation

The active fuel length for array/classes 6x6A and 6x6B is proposed to be increased to 120 inches to bound an earlier variant of Dresden-1 fuel. Among the fuel assemblies included in the 6x6A array/class, one particular fuel type was determined to be fabricated with a thinner cladding (0.026 in.) relative to other fuel in this class (minimum 0.030 in. cladding). In the 7x7A array/class of fuel assemblies, minor adjustments to the fuel parameters[4] was necessary to bound Humboldt Bay fuel. Changes to the 7x7B and 8x8B array/classes were necessary to bound the fuel types at Oyster Creek plant. Accordingly, the thermal analyses for these fuel types were evaluated in support of this amendment and additional analyses performed, as required.

A review of the Oyster Creek fuel parameters against the fuel parameters of other fuel types in the same array/classes has revealed no significant differences. The Oyster Creek 7x7 fuel rod mechanical parameters are identical to an existing member of the 7x7B class. The relatively larger pellet diameter (from 0.491 vs. 0.488 in) necessitates an adjustment to the uranium weight limit for this array/class. The Oyster Creek 8x8 fuel rod diameter is slightly larger than other members in the 8x8B class and has a thicker cladding.

An 8x8 fuel assembly used at Browns Ferry and a 9x9 fuel assembly from Grand Gulf, have been evaluated in support of this amendment request to modify the BWR fuel parameters. Likewise, a Millstone Unit 2 14x14 fuel assembly has been evaluated to support modification of the PWR fuel tables. As explained below, these PWR and other BWR fuel have been evaluated in accordance with

[4] Cladding thickness change from 0.033 inch to 0.0328 inch and active fuel length from 79 in to 80 in.

the NRC-approved HI-STORM thermal analysis methodologies to confirm that the HI-STORM 100 temperature field is bounded by the design basis analyses.

The overall HI-STORM thermal analysis methodology is partitioned into two evaluations. The first evaluation pertains to determining the appropriate peak cladding temperature limits for long term dry storage for each proposed fuel type. For this purpose, theoretical bounding rod gas pressures for the PWR and BWR classes of fuel are employed. In the second evaluation, the temperature field in the HI-STORM 100 cask is computed and the resulting cladding temperatures demonstrated to be below the respective temperature limits. The analytical evaluations for BWR fuel are further sub-divided in two groups of fuel assemblies classified as Low Heat Emitting (LHE) fuel assemblies and Design Basis (DB) fuel assemblies. The LHE fuel assemblies are characterized by low burnup, long cooling time and short active fuel lengths. Consequently, their heat loads are dwarfed by the full active length DB fuel assemblies. The additional Dresden-1 and Humboldt Bay fuel assemblies in the 6x6A and 7x7A array/classes belong to the LHE group of fuel, while the additional Oyster Creek, Browns Ferry, and Grand Gulf fuel assemblies are included in the DB group.

In accordance with the PNL-6189 methodology, peak fuel cladding temperature limits are specified as a function of cladding stress and age of fuel. The cladding stress calculations for the additional fuel are documented in proposed revised TSAR Tables 4.3.2, 4.3.3, 4.3.5 and 4.3.6 in Attachment 5 to this letter. The cladding stress in the additional DB fuel types is bounded by the limiting cladding stress computed previously. An adjustment to the 10x10 SVEA-96 fuel parameters (an O.D. change by 0.001 inch) is insignificant for the cladding stress evaluation as it is bounded by the design basis cladding stress. Consequently, the age-dependent peak fuel cladding temperature limits do not require changes to accommodate the additional fuel. For the LHE fuel group, the thin-clad Dresden-1 fuel type is determined to be the limiting fuel resulting in a downward shift in the applicable fuel cladding temperature limit. The revised temperature limits for LHE and DB fuel are summarized in proposed revised TSAR Tables 4.3.7 and 4.3.8.

The second evaluation pertaining to computation of the HI-STORM 100 cask temperature field is functionally dependent upon the effective conductivity of fuel assemblies loaded in the MPC-68 fuel cells. The LHE fuel assemblies are further analyzed under the assumption that they are loaded while encased in stainless steel DFCs. Due to interruption of radiation heat exchange between the fuel assembly and the fuel basket by the DFC boundary, this configuration is bounding for the thermal evaluation. Two DFC designs are evaluated - a previously approved Holtec design (TSAR Figure 2.1.1) and an existing TN/D-1 DFC in which some of the Dresden-1 fuel is currently stored (TSAR Figure 2.1.2) (see Proposed Change Number 5). The most resistive fuel assembly determined by analytical evaluation is considered for the HI-STORM 100 cask thermal evaluation. The results of the evaluation of additional fuel types performed in support of this amendment request are summarized in proposed revised Table 4.4.6 for LHE and DB fuel (see Attachment 5).

In both groups investigated, the thermal conductivity of the additional fuels is bounded by the limiting fuel types in each group. For the DB group of fuel assemblies, it is shown that the peak cladding temperature limits for the limiting fuel type adequately cover the additional fuel. The most resistive fuel characteristics also bound the additional fuel in the list of DB fuel types authorized for storage in the HI-STORM 100 System. Thus, the design basis thermal analysis envelopes the HI-STORM 100 System thermal response when loaded with the additional BWR and PWR fuel. For the LHE group of assemblies, the low decay heat load burden on the HI-STORM 100 cask (~ 8kW) guarantees large thermal margins to permit safe storage of Dresden-1 and Humboldt Bay fuel. Nevertheless, a conservative analysis was performed and is described in the proposed Revision 11 TSAR and the temperature field determined and reported Subsection 4.4.1.1.13 (see Attachment 5).

Shielding Evaluation

The accuracy of the shielding analysis is dependent upon the calculation of the radiation source term. The source term is dependent on the mass of uranium in the fuel assembly. For a specified burnup and cooling time, the radiation source term will increase as the mass of uranium increases (this is addressed in Proposed Change Number 25). The minor changes proposed for the dimensions of the fuel assembly array/classes will have a negligible impact on the radiation source term. Since the allowable uranium mass loadings are not being changed as a result of these changes in dimensions, it is concluded that these changes will have a negligible effect of the shielding analysis and therefore are not explicitly considered in Revision 11 of Chapter 5 of the HI-STORM TSAR.

Criticality Evaluation

For the criticality evaluation, the fuel assemblies are grouped into assembly array/classes. The proposed CoC modifications to fuel assemblies already included are reflected in proposed revised TSAR Table 6.2.1 (see Attachment 5). For each assembly array/class, a theoretical bounding assembly is defined. The characteristics of the bounding assembly for each affected array/class was amended to reflect the additional fuel types within an array/class.

Criticality calculations were performed for the changed fuel types and the bounding assembly in each array/class to account for the modified dimensions. Table 26.1 below shows the comparison between the maximum k_{eff} for each of the affected array/classes and the corresponding current values (i.e. TSAR Rev. 10). The TSAR table number containing the detailed results is also listed. The comparison demonstrates that, apart from the 10x10C assembly class, the maximum k_{eff} of each affected class only changes slightly as a result of the changes in the fuel assembly characteristics.

For the 10x10C assembly class, the changes are larger due to a change in the material of the internal structures (water tubes) inside the assembly. The initial calculation assumed stainless steel for these structures, whereas the actual material is a zirconium alloy. This results in an increase in reactivity, as the zirconium alloy shows a lower neutron absorption compared to stainless steel. Additionally, some dimensions in the model (Channel ID and sub-assembly spacing) deviated from the fuel manufacturers specification available for this assembly. Adjustment of these values leads to an additional small reduction in reactivity. Overall, for the same initial planar average enrichment of 4.2 wt% ^{235}U, the reactivity of this assembly increases, but still remains below 0.95. Therefore, with the proposed changes, the cask system is still in compliance with the regulatory requirement of $k_{eff} < 0.95$ for all authorized fuel assembly array/classes.

Table 26.1
Comparison of Maximum k_{eff} for TSAR Rev. 10 and Proposed Rev. 11

Assembly Array/Class	Maximum k_{eff} TSAR Rev. 10	Table Number in TSAR Rev. 10	Maximum k_{eff} TSAR Proposed Rev. 11	Table Number in Proposed Rev. 11 of the TSAR
6x6A	0.7602	6.2.35	0.7888	6.2.41
6x6B	0.7611	6.2.36	0.7824	6.2.42
7x7A	0.7973	6.2.38	0.7974	6.2.44
7x7B	0.9375	6.2.19	0.9386	6.2.23
8x8A	0.7685	6.2.39	0.7697	6.2.45
8x8B	0.9368	6.2.20	0.9416	6.2.24
8x8D	0.9366	6.2.22	0.9403	6.2.26
9x9B	0.9388	6.2.25	0.9436	6.2.30
9x9D	0.9392	6.2.27	0.9394	6.2.32

Assembly Array/Class	Maximum k_{eff} TSAR Rev. 10	Table Number in TSAR Rev. 10	Maximum k_{eff} TSAR Proposed Rev. 11	Table Number in Proposed Rev. 11 of the TSAR
9x9E	0.9406	6.2.28	0.9401	6.2.33
9x9F	0.9377	6.2.29	0.9401	6.2.34
10x10C	0.8990	6.2.32	0.9433	6.2.38
14x14C	0.9361	6.2.6	0.9400	6.2.8

Confinement Evaluation

There is no effect of these proposed changes on the confinement evaluation because the source terms used in the confinement analysis are not changed.

Proposed Change No. 27

Certificate of Compliance, Appendix B, Tables 2.1-2 and 2.1-3:

Four new fuel assembly array/classes, 14x14E and 15x15H* (PWR); and 8x8F* and 9x9G (BWR) are added to Appendix B, Tables 2.1-2 and 2.1-3, respectively, as shown in Tables 27.1 and 27.2 below and in the attached marked-up CoC tables. Items II.A.1.d and e and Items VI.a.1.d and e in Table 2.1-1 are also revised to add separate decay heat, cooling time, and burnup limits for the 8x8F array/class (QUAD+ assembly).

Table 27.1
New PWR Fuel Assembly Array/Classes 14x14E and 15x15H

Fuel Assembly Array/Class	**14x14E**	**15x15H**
Clad Material	SS	Zr
Design Initial U (kg/assy.)	≤ 206	≤ 475
Initial Enrichment (wt % ^{235}U)		
MPC-24 without soluble boron credit	≤ 5.0	≤ 3.8
MPC-24E/24EF without soluble boron credit	≤ 5.0	≤ 4.2
Any PWR MPC with soluble boron credit	≤ 5.0	≤ 5.0
No. of Fuel Rod Locations	173	208
Fuel Clad O.D. (in.)	≥ 0.3145	≥ 0.414
Fuel Clad I.D. (in.)	≤ 0.3175	≤ 0.3700
Fuel Pellet Dia. (in.)	≤ 0.3130	≤ 0.3622
Fuel Rod Pitch (in.)	0.441 and 0.453	≤ 0.568
Active Fuel Length (in.)	≤ 102	≤ 150
No. of Guide and/or Instrument Tubes	0	17
Guide/Instrument Tube Thickness (in.)	N/A	≥ 0.0140

Table 27.2
New BWR Fuel Assembly Array/Classes 8x8F and 9x9G

Fuel Assembly Array/Class	**8x8F**	**9x9G**
Clad Material	Zr	Zr
Design Initial U (kg/assy.)	≤ 191	≤ 179
Maximum PLANAR-AVERAGE INITIAL ENRICHMENT (wt.% ^{235}U)	≤ 4.0	≤ 4.2
Initial Maximum Rod Enrichment(wt.% ^{235}U)	≤ 5.0	≤ 5.0
No. of Fuel Rod Locations	64	72
Fuel Clad O.D. (in.)	≥ 0.4576	≥ 0.4240
Fuel Clad I.D. (in.)	≤ 0.3996	≤ 0.3640
Fuel Pellet Dia. (in.)	≤ 0.3913	≤ 0.3565
Fuel Rod Pitch (in.)	≤ 0.609	≤ 0.572
Design Active Fuel Length (in.)	≤ 150	≤ 150
No. of Water Rods	N/A	1
Water Rod Thickness (in.)	≥ 0.0315	≥ 0.320
Channel Thickness (in.)	≤ 0.055	≤ 0.120

Reason for Proposed Changes

Based on user feedback, additional fuel assemblies were identified that did not fit into any of the existing fuel assembly array/classes. Four new assembly array/classes are required to assure all user fuel types can be loaded. The 14x14E array/class represents only Indian Point Unit 1 fuel. The 15x15H includes the B&W Mark B11 fuel design. The 8x8F represents only the "QUAD+" assembly. The 9x9G array/class represents the ANF-9X fuel assembly.

Justification for Proposed Changes

Structural Evaluation

The addition of new fuel types permitted to be stored in the HI-STORM 100 System can have an effect on the structural analyses performed in Chapter 3 if, and only if, one or more of the following occurs because of the new fuel types:

1. The design basis weights of 700 lbs (BWR) or 1680 lbs. (PWR), including non-fuel hardware, channels, and DFCs, as applicable, are exceeded.

2. The design basis temperatures are exceeded because of the presence of the new fuel types.

3. The lengths of the new fuel assemblies cause an increase in the length of the Holtec fuel spacers.

Section 3.0 of the HI-STORM TSAR contains a compliance matrix showing how the structural review requirements of NUREG 1536 have been satisfied by the totality of analyses currently reviewed and reported in Chapter 3. To ascertain whether any of the proposed amendment items require a re-visiting of any or all of the currently approved analyses reported in Chapter 3, the Compliance Matrix was reviewed and the following conclusions reached.

1. The weights of the proposed new fuel types do not exceed the limiting (i.e., design basis) weights specified in Table 2.1-1 of Appendix B to the CoC. Therefore, no structural analysis currently approved needs to be re-visited.

2. The design basis temperatures of all components have not exceeded the values currently licensed. Therefore, no structural analyses or free thermal expansion analyses currently approved needs to be revisited.

3. The lengths of the proposed new fuel types are longer than the minimum length of the fuel assemblies currently approved for the HI-STORM 100. Therefore, the fuel spacer stability analysis in the TSAR remains bounding.

The lengths of the proposed new fuel types are also less than the maximum lengths specified in Table 2.1-1 of Appendix B to the CoC.

Thermal Evaluation

The Indian Point Unit 1, B&W Mark B11, QUAD+, and ANF-9X fuel types have been evaluated along with the changes to the existing 8x8 and 15x15 fuel assembly array/classes as described in Proposed Change No. 26 above.

The B&W Mark B11 and ANF-9X fuel assemblies are bounded by the existing design basis thermal analyses. The QUAD+ fuel assembly is included in the LHE group of BWR fuel assemblies and has been found acceptable for safe storage in proposed Revision 11 of the HI-STORM TSAR Subsection 4.4.1.1.13. The Indian Point Unit 1 fuel assembly is included in the stainless steel group of PWR fuel assemblies and has been found acceptable for safe storage in proposed Revision 11 of the HI-STORM TSAR Subsection 4.4.1.1.13.

Shielding Evaluation

The accuracy of the shielding analysis is dependent upon the calculation of the radiation source term. The source term is dependent on the mass of uranium in the fuel assembly. For a specified burnup and cooling time, the radiation source term will increase as the mass of uranium increases. Minor variations in the dimensions of a fuel assembly will have a negligible impact on the radiation source term if the mass or uranium remains constant. The additional fuel assemblies proposed for the CoC are not significantly different than the currently licensed fuel assemblies to require an assembly-specific source term calculation. These new fuel assemblies are bounded by the current design basis fuel assemblies. In addition, the allowable uranium mass loadings for these new fuel assemblies is specified consistent with similar fuel assemblies in the CoC thereby assuring that these assemblies are bounded by the current design basis fuel assemblies. Therefore, these additions will have a negligible effect of the shielding analysis and therefore are not explicitly considered in proposed Revision 11 of Chapter 5 of the HI-STORM TSAR.

Criticality Evaluation

Criticality calculations were performed for all four new fuel array/classes. The results for these classes in the MPC-24 and MPC-68 are summarized in Table 27.3 below. The two PWR assemblies (14x14E and 15x15H) are also permitted in the MPC-24E, MPC-24EF, MPC-32 and the MPC-24 with credit for soluble boron. Maximum k_{eff} values for these baskets are similar to the values listed in Table 21.3 below, and can be found in Tables 6.1.2 through 6.1.6 in Section 6.1 of the Proposed Rev. 11 of the TSAR (see Attachment 5). For all new PWR and

BWR fuel assemblies, the maximum k_{eff} is below 0.95. Therefore, with the proposed changes, the cask system is still in compliance with the regulatory requirement of $k_{eff} < 0.95$ for all authorized fuel assembly array/classes.

Table 27.3
Maximum k_{eff} for new PWR and BWR Fuel Assembly Array/Classes

Fuel Assembly Array/Class	Basket Type	Maximum k_{eff}	Table Number in Proposed Rev. 11 of the TSAR
14x14E	MPC-24	0.7715	6.2.10
15x15H	MPC-24	0.9411	6.2.18
8x8F	MPC-68/68FF	0.9411	6.2.28
9x9G	MPC-68/68FF	0.9309	6.2.35

Confinement Evaluation

The source terms used for the existing confinement analysis bound those of the new fuel assembly array/classes. Therefore, there is no impact on confinement.

Proposed Change No. 28

Certificate of Compliance, Appendix B, Approved Contents, Tables 2.1-4 through 2.1-7:

The per-assembly limits on fuel burnups, cooling time, and decay heat have been modified to reflect credit being taken for thermosiphon (convection inside the MPC) heat transfer and to allow loading of high burnup fuel (> 45,000 MWD/MTU) into the MPC.

Reason for Proposed Changes

To take appropriate account for a naturally-occurring method of heat transfer inside the MPC vessel that was previously not credited. User feedback indicates a growing inventory of fuel burned to greater than 45,000 MWD/MTU that must be authorized for loading into dry storage casks.

Justification for Proposed Changes

Structural Evaluation

The thermosiphon effect inside the MPC (now included in Chapter 4 thermal evaluations) results in an alteration in the MPC/overpack temperature distributions. The free thermal expansion evaluations, summarized in Subsection 3.4.4.2.1 for the "hot" MPC in HI-STORM or HI-TRAC and in Subsection 3.4.5 for the "hot" MPC inserted into a "cold" overpack have been revisited using the new temperature distributions from Chapter 4 The summary tables have been updated and the applicable appendices showing detailed calculations (Appendices 3.U, 3.V, 3.W, 3.I, 3.AF, and 3.AQ) have been revised to reflect the new temperatures. The revised calculations continue to demonstrate that there is no restraint of free thermal expansion between the fuel basket and the MPC canister, and between the MPC and the HI-STORM or HI-TRAC overpacks.

Thermal Evaluation

In the previous HI-STORM licensing analyses, the thermal models were run with the MPC internal convection heat transfer completely suppressed. Benchmarking studies performed by Holtec on full-size cask data and submitted to the NRC as a topical report showed that a complete neglect of the thermosiphon effect has the result of grossly over-predicting the peak fuel cladding temperature by as much as 200°F. Recent independent work performed by PNNL on the thermal simulation of the HI-STORM 100 System using the COBRA-SFS code has confirmed large conservatisms in the peak cladding temperature results of the current HI-STORM 100 thermal model that does not recognize internal convection in the honeycomb basket-equipped MPC. Including MPC internal convection heat dissipation in the HI-STORM 100 thermal models is found to yield conservative results when compared with the full-scale cask data. However, the extent of the conservatism is not inordinately large.

Accordingly, in the revised thermal model, the effect of internal convection is incorporated. However, to impute added conservatism, the conduction heat transfer contribution of the "aluminum heat conduction elements" (located in the peripheral spaces between the fuel basket and the MPC wall) is neglected. A detailed discussion of the revised thermal model is included in the proposed Chapter 4 TSAR Revision 11 changes.

Shielding Evaluation

The shielding evaluation in Chapter 5 has been revised to reflect the increased permissible heat loads and the increased burnups beyond 45 GWD/MTU. The increased heat loads result in a decreased cooling time for a specific burnup. The increased heat loads in conjunction with higher burnups result in increased dose rates around the HI-STORM 100 System including the HI-TRAC transfer casks. These dose rates are evaluated in Chapter 5 and their effect on occupational exposure are evaluated in Chapter 10. As a result of the increased dose rates, the dose rate limits specified in the LCOs have been increased (see Proposed Change Nos. 3 and 4). While the dose rates and occupational exposure both increased, the HI-STORM 100 system is still in compliance with 10CFR72.104 and 10CFR72.106. The increases in the very conservatively estimated occupational exposures do not pose an ALARA concern, as the user may employ various forms of temporary shielding to reduce the dose rates.

Criticality Evaluation

All criticality analyses are performed assuming fresh fuel, i.e. no credit is taken for the reduction in reactivity due to the burnup of the fuel. An increase of the allowable fuel burnup will therefore increase the inherent safety margin in the criticality evaluation and no further analyses are necessary.

Confinement Evaluation

The confinement evaluation has been modified to account for the increased fuel burnup limits per-assembly. The source terms for the MPC-24, MPC-24E, MPC-24EF, MPC-32, MPC-68 and MPC-68FF have been chosen to ensure that a bounding inventory is chosen for determining the dose due to a leak in the confinement boundary. The inventory for the MPC-24, MPC-24E, MPC-24EF and MPC-32 was conservatively based on the B&W 15x15 fuel assembly with a burnup of 70,000 MWD/MTU, 5 years of cooling time, and an enrichment of 4.8%. The inventory for the MPC-68 and MPC-68FF was based on the GE 7x7 fuel assembly with a burnup of 60,000 MWD/MTU, 5 years of cooling time, and 4.4% enrichment. The CoC limits the fuel assembly burnup below 60,000 MWD/MTU for both BWR and PWR fuel at 5 years of cooling time. This ensures that the inventory used in this calculation exceeds that of the fuel authorized for storage. Additionally, the leakage rate for normal, off-normal and hypothetical accident conditions has been updated to reflect the increased MPC pressure and temperature.

For storage of spent fuel assemblies with burnups in excess of 45 GWD/MTU the source term from the assumed rod breakage fractions of ISG-5 were augmented by the source term from 50% of the rods having peak cladding oxide thicknesses greater than 70 micrometers. ISG-11 recommends that for high burnup fuel assemblies the releasable source term from ISG-5 for normal and off-normal conditions be increased by an additional factor. Therefore, the source term available for release has been revised from 1.0% to 2.5% for normal conditions and from 10.0% to 11.5% for off-normal conditions.

Additionally, the confinement analysis has incorporated the assumption that only 10% of the fines released to the MPC cavity from a cladding breach remain airborne long enough to be available for release from the MPC. It is conservatively assumed that 100% of the volatiles, crud and gases remain airborne and available for release.

Proposed Change No. 29

Certificate of Compliance, Appendix B, Design Features Section 3.2:

New design features important for criticality control are added for the MPC-24E, MPC-24EF, MPC-68FF and the MPC-32.

Reason and Justification for Proposed Changes

These changes are conforming changes in support of the addition of these MPC models to the CoC. The values for Boron-10 loading and flux trap size are consistent with their respective design drawings, including tolerances.

Proposed Change No. 30*

Certificate of Compliance, Appendix B, Table 3-1:

The entry in the "Exception, Justification & Compensatory Measures" column for the exception to Code Section NB-5230 for the closure ring, vent, and drain cover plate welds is clarified as shown in the attached marked-up CoC table to recognize welds which may be single pass welds.

Reason for Proposed Change

To provide clarification and as a conforming change to a proposed drawing change (see Attachment 4).

Justification for Proposed Change

Small welds, such as 1/8 inch will likely be completed in one pass, with no root.

Proposed Change No. 31

Certificate of Compliance, Appendix B, Design Features Section 3.4.3:

Revisions are made to these requirements to distinguish between free-standing casks and casks for deployment in high seismic regions.

Reason for Proposed Changes

Under the current CoC, the seismic acceleration limits for free-standing casks do not envelop the seismic spectra for plants located in so-called "high seismic" regions. The new overpack variant called HI-STORM 100A is designed for use at ISFSIs in high seismic regions. The key design features proposed to be included in the CoC are essential for ensuring deployment of the HI-STORM 100A System is performed within the design and analysis basis for the system.

Justification for Proposed Changes

This is purely a structural design issue. The thermal, shielding, criticality, and confinement evaluations are unaffected by these design changes. The details of the structural evaluation may be found in the proposed Chapter 3 TSAR revisions in Attachment 5.

Proposed Change No. 32

Certificate of Compliance, Appendix B, Design Features Section 3.4.6:

Re-format and revise Design Features Section 3.4.6 to remove the specific ISFSI pad and subgrade design parameters and to distinguish between free-standing overpacks and the HI-STORM 100A. New Design Features Sections 3.4.6.a and 3.4.6.b establish ISFSI pad requirements for the free-standing and HI-STORM 100A overpacks, respectively, as shown in the attached marked-up CoC pages.

Reason for Proposed Change

The current CoC requires that all ISFSI pads be designed to meet a single set of design parameters, including pad thickness, concrete compressive strength, reinforcing bar yield strength, and subgrade modulus of elasticity. This proposed change allows the necessary flexibility for utility licensees to design their ISFSI pads according to their site-specific needs and geological characteristics.

Justification for Proposed Change

The deceleration limit of 45-g's for the HI-STORM 100 System provides assurance that the cask system, including contents, will remain intact and retrievable after a postulated drop event and non-mechanistic tipover event. Therefore, the 45-g deceleration limit is the appropriate safety limit to be included in the CoC, while the specific pad design parameters may be left to the discretion of the general licensee. Any site-specific drop and tipover analyses are required to be performed in accordance with the methodologies described in the Hi-STORM TSAR.

To assist the licensee in designing their ISFSI pad, Holtec has added a second set of "pre-approved" ISFSI pad and subgrade design parameters to TSAR Table 2.2.9. These design parameters were developed using the approved TSAR methodologies for cask drop and tipover analyses. Licensees may choose to design their ISFSI pads using either the Set "A" or Set "B" ISFSI pad and subgrade design parameters in TSAR Table 2.2.9, or design their own pad. Any ISFSI pad design is acceptable provided it is a structurally competent pad for which cask deceleration limits are shown to be met (if required).

SECTION II – PROPOSED CHANGES TO THE TSAR

Nearly all of the proposed TSAR changes included in Attachment 5 are in support of CoC changes discussed in Section I. The changes to Chapters 3 through 7 are referred to in the technical justifications in Section I, as required, and are not listed again as changes here in Section II. However, several TSAR changes have also been included in this LAR for NRC review due to their overall magnitude or potential significance. These changes are listed below by subject (e.g., HI-STORM 100S) or grouped by Chapter. Throughout the proposed Rev. 11 TSAR, text revisions may be found that correct editorial inconsistencies or support other changes not proposed for NRC review and approval (i.e., are being processed under 10 CFR 72.48). These text revisions are left in the chapter sections provided with this submittal for continuity of the chapter content by our chapter authors and are not proposed as changes requiring NRC review and approval.

In summary, if a TSAR change is not referred to from the change justifications in Section I or explicitly listed below, we intend to process the change under 10 CFR 72.48 and NRC review and approval <u>is not</u> requested as part of this LAR. Further, some of the changes below which <u>are</u> submitted for NRC approval <u>may</u> be implemented under 10 CFR 72.48 in parallel with NRC review to meet users' needs.

<u>Proposed Change No. 33</u>

<u>HI-STORM 100S</u>

The text is revised throughout the TSAR and new Bills-of-Material (BM-3065 and 3066) and design drawings (3067 through 3075) describing the HI-STORM 100S are included in Attachment 4. The technical evaluations of HI-STORM 100S contained in the proposed TSAR changes in Attachment 5 are summarized below by affected technical discipline below:

<u>Structural Evaluation</u>

The HI-STORM 100S overpack is a slightly shortened version of HI-STORM 100 overpack that is approximately 12,000 lb. lighter. The weight reduction has been achieved by reduction in the height of the concrete pedestal supporting the MPC and by the shortening of the overpack inner, outer, and shield shell, and the contained concrete. The weight of the HI-STORM 100S lid, however, is increased. Section 3.2 provides the specifics of the weights and center of gravity locations for the HI-STORM 100S loaded with the different MPCs. Detailed evaluations are performed in Chapter 3 to justify that nearly all analysis results previously performed and approved for the HI-STORM 100 bound results for the HI-STORM 100S and need not be repeated. Where justifications could not be provided, the detailed evaluations specific to the HI-STORM 100S are performed. All new safety factors specific to the HI-STORM 100S are greater than 1.0.

Where required to perform specific evaluations for the short HI-STORM, new appendices have been added to Chapter 3. Attachment 5 details all changed text and calculations specific to the introduction of the HI-STORM 100S into the TSAR.

Thermal Evaluation

From the standpoint of thermal performance, the HI-STORM 100S overpack is nearly identical to HI-STORM 100. HI-STORM 100S features a slightly smaller inlet duct-to-outlet duct separation and a slightly enhanced gamma shield cross plate (which acts as a flow straightener) than its older counterpart. The HI-STORM 100S peak fuel cladding temperatures are bounded by the HI-STORM 100 thermal solution. Therefore, HI-STORM 100S and HI-STORM 100 are considered to be interchangeable from the thermal-hydraulic standpoint.

Shielding Evaluation

The HI-STORM 100S overpack is quite similar to the current HI-STORM overpack. The only significant difference from a shielding perspective is that the MPC has been moved closer to the upper and lower air ducts. This results in an increase in the local dose rate at the opening of the ducts. In addition, the lid design has been changed by moving the concrete shielding from below the 4 inch thick steel to above the 4 inch steel plate in the top lid. The radial shielding is identical between the HI-STORM 100 and the HI-STORM 100S overpacks.

Chapter 5 of proposed Revision 11 of the HI-STORM TSAR specifically analyzes the HI-STORM 100S with the MPC-32 and the MPC-68. The MPC-24 analysis in the HI-STORM 100 overpack was unchanged. A comparison of the dose rates between the MPC-32 and MPC-24 indicates that the dose rate at the duct openings has increased in the HI-STORM 100S. This increase in the dose rate does not pose an ALARA concern and does not alter the HI-STORM 100 System's capability of meeting 10CFR72.104 requirements. Since the only significant change in the dose rate between the HI-STORM 100 and the HI-STORM 100S is at the duct opening, the previous analysis of the controlled area boundary dose rates using the HI-STORM 100 overpack was maintained.

For those users that are especially concerned with the dose rate at the duct openings, the HI-STORM 100S offers optional gamma shield cross plates which have more metal than the standard gamma shield cross plates and would therefore further reduce the dose rates at the duct openings.

Proposed Change No. 34

Changes to TSAR Chapter 1

a. The text and figures in Sections 1.0, 1.1, and 1.2 have been revised throughout to address the new MPC models, HI-STORM 100S, HI-STORM 100A, new DFC designs, the increased heat duty of the cask, and other CoC changes from Section I.

b. Table 1.0.1, Section 1.2, and Appendix 1.B have been revised to clarify the text regarding our Holtite-A neutron shielding material, consistent with our docketed correspondence dated August 18, 2000.

c. Several definitions in Table 1.0.1 have been modified or added in support of other proposed changes in the CoC and TSAR document.

d. Section 1.4 has been revised to clarify the requirements for cask spacing and make them more consistent with the thermal analysis basis.

Proposed Change No. 35

Changes to TSAR Chapter 2

a. The text and figures in Sections 2.0, 2.1, 2.2, and 2.3 have been revised throughout to address the new MPC models, HI-STORM 100S, HI-STORM 100A, new DFC design, the increased heat duty of the cask, and other CoC changes from Section I.

b. Table 2.3.1 is revised to be consistent with the latest language in 10 CFR 72.104 and 72.106.

Proposed Change No. 36

Changes to TSAR Chapter 7

The confinement analyses have been revised to reflect new regulations at 10 CFR 72.104 and the revised review guidance in ISG-5 and ISG-11.

Proposed Change No. 37

Changes to TSAR Chapter 8

a. The operating procedures have been revised throughout to include lessons learned from cask loadings at Dresden Unit 1 and Plant Hatch and other enhancements.

b. The chapter introduction has been revised to provide clarification regarding the need for users to develop and implement site-specific procedures that meet the intent of Chapter 8. Additional flexibility is proposed regarding the content of the site-specific procedures and the need for 72.48 evaluations for differences between the procedures and TSAR Chapter 8. In essence, wide flexibility is afforded the general licensees in preparing site-specific procedures outside of the purview of 10 CFR 72.48, provided the intent of the Chapter is met.

c. The ITS categories for several ancillary components are clarified based on lessons learned from the ongoing engineering and manufacturing phase for these components.

Proposed Change No. 38

Changes to TSAR Chapter 9

a. Subsection 9.1.5.1 is revised regarding Holtite-A testing to re-define the frequency of testing to be every manufactured lot instead of every mixed batch.*

b. Subsection 9.1.5.1 is revised to allow the option of installing the lead shielding in the HI-TRAC transfer cask as pre-cast sections in lieu of pouring molten lead. This is for fabrication flexibility. Appropriate cautions are also added to minimize gaps if pre-cast sections are used.

c. Subsection 9.1.5.2 is revised to allow gamma scanning of the HI-TRAC shielding prior to, or after the installation of the water jacket. This is fabrication flexibility.

Proposed Change No. 39

Changes to TSAR Chapter 10

a. The occupational exposure estimates were revised throughout to account for the revised dose rates due to fuel and MPC changes, and the HI-STORM 100S configuration.

b. Table 10.1.2 has been revised to provide general licensees with the flexibility to decide for themselves whether or not temporary shielding is required based on the particular age and burnup of the fuel being loaded, and the actual dose rates measured. Long cooling time and/or low burnup fuel may yield very low dose rates, even with the HI-TRAC 100, potentially obviating the need for any temporary shielding. Users' radiation protection programs will govern the use of temporary shielding.

Proposed Change No. 40

Changes to TSAR Chapter 11

The events and accidents in Subsections 11.1 and 11.2 have been reanalyzed, as necessary to reflect changes made to the authorized contents and heat loads for the MPC in Section I.

Proposed Change No. 41

Changes to TSAR Chapter 12

The list of technical specifications in Table s 12.1.1 and 12.1.2 and the Bases in Appendix 12.A are revised to match the changes to the technical specifications proposed in Section I.

Section III – PROPOSED CHANGES TO DESIGN DRAWINGS

Proposed Change No. 42

Drawings for MPC-24E/24EF, MPC-32, MPC-68FF, HI-STORM 100S and HI-STORM 100A

a. The MPC-24E/24EF required the generation of four (4) new design drawings: 2889, 2890, 2891, 2892 and BOMs 2898 and 2899. Where appropriate, the existing (standard) MPC-24 drawings (1395 and 1396 series) have been revised as required to indicate certain dimensions and details which differ between the MPC-24 and the MPC-24E/24EF.

b. The MPC-32 required the return of the ten (10) drawings previously included in the HI-STORM TSAR in the early phases of the license review (1392 and 1393 series, and BM-1477, Sheets 1 and 2). These drawings have been revised to incorporate the lessons learned from Plant Hatch and the HI-STAR prototype fabrication activities similar to the MPC-24 and MPC-68 drawings.

c. The MPC-68FF required the revision of four (4) existing (standard) MPC-68 drawings (1402 series) to show the enhanced MPC upper shell and deeper MPC lid weld, similar to the MPC-68F.

d. The HI-STORM 100S required the creation of eleven (11) new drawings (3067 through 3077) and a new Bill-of-Materials (3065 and 3066). Only nine of the eleven drawings are included in the TSAR. Drawings 3076 and 3077 depict the cask nameplate and inlet and outlet vent screens, respectively. This equipment is considered ancillary to the cask system. Therefore, the drawings are not considered necessary to be included in the TSAR.

e. The HI-STORM 100A required the creation of one new drawing (3187) and one new Bill-of-Material (BM-3189).

All new and revised drawings are included in Attachment 4 of this submittal. Please note that a number of the changes shown on the MPC drawings (not related to CoC changes) may be implemented under 10 CFR 72.48 prior to NRC approval of this CoC amendment request.

CERTIFICATE OF COMPLIANCE
FOR SPENT FUEL STORAGE CASKS
Supplemental Sheet

9. SPECIAL REQUIREMENTS FOR FIRST SYSTEMS IN PLACE

The heat transfer characteristics of the cask system will be recorded by temperature measurements for the first HI-STORM SFSC Systems (MPC-24, MPC-24E, MPC-24EF, MPC-32, MPC-68, MPC-68F and MPC-68FF) placed into service with a heat load equal to or greater than 10 kW. An analysis shall be performed that demonstrates the temperature measurements validate the analytic methods and predicted thermal behavior described in Chapter 4 of the SAR.

Validation tests shall be performed for each subsequent cask system that has a heat load that exceeds a previously validated heat load by more than 2 kW (e.g., if the initial test was conducted at 10 kW, then no additional testing is needed until the heat load exceeds 12 kW). No additional testing is required for a system after it has been tested at a heat load equal to or greater than 16 kW.

Letter reports summarizing the results of each validation test shall be submitted to the NRC in accordance with 10 CFR 72.4. Cask users may satisfy these requirements by referencing validation test reports submitted to the NRC by other cask users.

~~9.~~ *10.* AUTHORIZATION

The HI-STORM 100 Cask System, which is authorized by this certificate, is hereby approved for general use by holders of 10 CFR Part 50 licenses for nuclear reactors at reactor sites under the general license issued pursuant to 10 CFR 72.210, subject to the conditions specified by 10 CFR 72.212, and the attached Appendix A and Appendix B.

FOR THE U.S. NUCLEAR REGULATORY COMMISSION

E. William Brach, Director
Spent Fuel Project Office
Office of Nuclear Materials Safety
and Safeguards

Attachments:

1. Appendix A
2. Appendix B

TABLE OF CONTENTS

3.1 SFSC INTEGRITY

3.1.1 Multi-Purpose Canister (MPC)

LCO 3.1.1 The MPC shall be dry and helium filled.

APPLICABILITY: During TRANSPORT OPERATIONS and STORAGE OPERATIONS.

ACTIONS

-----NOTE-----
Separate Condition entry is allowed for each MPC.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. MPC cavity vacuum drying pressure limit not met.	A.1 Perform an engineering evaluation to determine the quantity of moisture left in the MPC.	7 days
	<u>AND</u>	
	A.2 Develop and initiate corrective actions necessary to return the MPC to an analyzed condition.	30 days
B. MPC helium backfill ~~density~~ *pressure* limit not met.	B.1 Perform an engineering evaluation to determine the impact of helium differential.	72 hours
	<u>AND</u>	
	B.2 Develop and initiate corrective actions necessary to return the MPC to an analyzed condition.	14 days

ACTIONS
(continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. MPC helium leak rate limit not met.	C.1 Perform an engineering evaluation to determine the impact of increased helium leak rate on heat removal capability and offsite dose.	24 hours
	AND	
	C.2 Develop and initiate corrective actions necessary to return the MPC to an analyzed condition.	7 days
D. Required Actions and associated Completion Times not met.	D.1 Remove all fuel assemblies from the SFSC.	30 days

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.1.1.1	Verify MPC cavity vacuum drying pressure is within the limit specified in Table 3-1 for the applicable MPC model.	Once, prior to TRANSPORT OPERATIONS
SR 3.1.1.2	Verify MPC helium backfill ~~density~~ *pressure* is within the limit specified in Table 3-1 for the applicable MPC model.	Once, prior to TRANSPORT OPERATIONS
SR 3.1.1.3	Verify that the total helium leak rate through the MPC lid confinement weld and the drain and vent port confinement welds is within the limit specified in Table 3-1 for the applicable MPC model.	Once, prior to TRANSPORT OPERATIONS

3.1 SFSC INTEGRITY

3.1.2 SFSC Heat Removal System

LCO 3.1.2 The SFSC Heat Removal System shall be OPERABLE

APPLICABILITY: During STORAGE OPERATIONS.

ACTIONS

---NOTE---
Separate Condition entry is allowed for each SFSC.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. SFSC Heat Removal System inoperable.	A.1 Restore SFSC Heat Removal System to OPERABLE status.	8 hours
B. Required Action A.1 and associated Completion Time not met.	B.1 Perform SR 3.2.3.1.	Immediately and every 12 hours thereafter
	<u>AND</u>	
	B.2.1 Restore SFSC Heat Removal System to OPERABLE status.	48 hours
	<u>OR</u>	
	B.2.2 Transfer the MPC into a TRANSFER CASK.	48 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.1.2.1	Verify all OVERPACK inlet and outlet air ducts are free of blockage.	24 hours
	<u>OR</u>	
	For OVERPACKS with installed temperature monitoring equipment, verify *that* the difference between the average OVERPACK air outlet temperature and ISFSI ambient temperature is $\leq$ *126°F.* ~~99° F (for the MPC-24) and $\leq$ 105° F (for the MPC-68 and MPC-68F)~~	24 hours

3.1 SFSC INTEGRITY

3.1.3 Fuel Cool-Down

LCO 3.1.3 The MPC helium exit temperature shall be ≤ 200° F

---NOTE---
The LCO is only applicable to wet UNLOADING OPERATIONS.

APPLICABILITY: UNLOADING OPERATIONS prior to re-flooding.

ACTIONS

---NOTE---
Separate Condition entry is allowed for each MPC.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. MPC helium gas exit temperature not within limit.	A.1 Establish MPC helium gas exit temperature within limit.	Prior to initiating MPC re-flooding operations
	AND	
	A.2 Ensure adequate heat transfer from the MPC to the environment	~~24~~ *22* hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.1.3.1	Verify MPC helium gas exit temperature within limit.	Prior to MPC re-flooding operations.

3.2 SFSC RADIATION PROTECTION

3.2.1 TRANSFER CASK Average Surface Dose Rates

LCO 3.2.1 The average surface dose rates of each TRANSFER CASK shall not exceed:

a. 125 Ton TRANSFER CASK

i. ~~130~~ *220* mrem/hour (neutron + gamma) on the side; |

ii. ~~40~~ *60* mrem/hour (neutron + gamma) on the top |

b. 100 Ton TRANSFER CASK

i. ~~890~~ *1500* mrem/hour (neutron + gamma) on the side; |

ii. ~~170~~ *315* mrem/hour (neutron + gamma) on the top |

APPLICABILITY: During TRANSPORT OPERATIONS.

ACTIONS

-----NOTE-----
Separate Condition entry is allowed for each TRANSFER CASK.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. TRANSFER CASK average surface dose rate limits not met.	A.1 Administratively verify correct fuel loading.	24 hours
	AND	
	A.2 Perform evaluation to verify compliance with the ISFSI offsite radiation protection requirements of 10 CFR Part 20 and 10 CFR Part 72.	24 hours

(continued)

3.2 SFSC RADIATION PROTECTION

3.2.3 OVERPACK Average Surface Dose Rates

LCO 3.2.3 The average surface dose rates of each OVERPACK shall not exceed:

a. ~~40~~ *50* mrem/hour (neutron + gamma) on the side

b. 10 mrem/hour (neutron + gamma) on the top

c. ~~16~~ *40* mrem/hour (neutron + gamma) at the inlet and outlet vent ducts

APPLICABILITY: During ~~TRANSPORT OPERATIONS AND~~ STORAGE OPERATIONS.

ACTIONS

---NOTE---

Separate Condition entry is allowed for each SFSC.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. OVERPACK average surface dose rate limits not met.	A.1 Administratively verify correct fuel loading. <u>AND</u> A.2 Perform ~~analysis~~ *a written evaluation* to verify compliance with the ISFSI offsite radiation protection requirements of 10 CFR Part 20 and 10 CFR Part 72.	24 hours ~~24~~ *48* hours
B. Required Action and associated Completion Time not met.	B.1 Remove all fuel assemblies from the SFSC.	30 days

3.3 SFSC CRITICALITY CONTROL

3.3.1 Boron Concentration

LCO 3.3.1 *As required by CoC Appendix B, Table 2.1-2, the concentration of boron in the water in the MPC shall meet the following limits for the applicable MPC model:*

a. *MPC-24 with one or more fuel assemblies having an initial enrichment greater than the value in Table 2.1-2 for no soluble boron credit and $\leq$ 5.0 wt% ^{235}U: $\geq$ 400 ppmb*

b. *MPC-24E and MPC-24EFwith one or more fuel assemblies having an initial enrichment greater than the value in Table 2.1-2 for no soluble boron credit and $\leq$ 5.0 wt% ^{235}U: $\geq$ 300 ppmb*

c. *MPC-32 with all fuel assemblies having an initial enrichment $\leq$ 4.1 wt% ^{235}U: $\geq$ 1900 ppmb*

d. *MPC-32 with one or more fuel assemblies having an initial enrichment > 4.1 and $\leq$ 5.0 wt% ^{235}U: $\geq$ 2600 ppmb*

APPLICABILITY: *During PWR fuel LOADING OPERATIONS with fuel and water in the MPC*

AND

During PWR fuel UNLOADING OPERATIONS with fuel and water in the MPC.

ACTIONS

---NOTE---

Separate Condition entry is allowed for each MPC.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Boron concentration not within limit.	A.1 Suspend LOADING OPERATIONS or UNLOADING OPERATIONS.	Immediately
	<u>AND</u>	
	A.2 Suspend positive reactivity additions.	Immediately
	<u>AND</u>	
	A.3 Initiate action to restore boron concentration to within limit.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
---------------------NOTE--------------------- This surveillance 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. --	Within 4 hours of entering the Applicability of this LCO.
	<u>AND</u>
SR 3.3.1.1 Verify boron concentration is within the applicable limit using two independent measurements.	Every 48 hours thereafter.

Table 3-1
MPC Model-Dependent Limits

MPC MODEL	LIMITS
1. MPC-*24/24E/24EF*	
a. MPC Cavity Vacuum Drying Pressure	≤ 3 torr for ≥ 30 min
b. MPC Helium Backfill ~~Density~~ *Pressure*[1]	~~0.1212 +0/-10% g-moles/l~~ *≥ 29.3 psig and ≤ 33.3 psig*
c. MPC Helium Leak Rate	≤ 5.0E-6 atm cc/sec (He)
2. MPC-68*/68F/68FF*	
a. MPC Cavity Vacuum Drying Pressure	≤ 3 torr for ≥ 30 min
b. MPC Helium Backfill ~~Density~~ *Pressure*[1]	~~0.1218 +0/-10% g-moles/l~~ *≥ 29.3 psig and ≤ 33.3 psig*
c. MPC Helium Leak Rate	≤ 5.0E-6 atm cc/sec (He)
3. MPC-32	
a. MPC Cavity Vacuum Drying Pressure	*≤ 3 torr for ≥ 30 min*
b. MPC Helium Backfill Pressure[1]	*≥ 29.3 psig and ≤ 33.3 psig*
c. MPC Helium Leak Rate	*≤ 5.0E-6 atm cc/sec (He)*

[1] Helium used for backfill of MPC shall have a purity of ≥ 99.995%.

5.0 ADMINISTRATIVE CONTROLS AND PROGRAMS

The following programs shall be established, implemented and maintained.

5.1 ~~Training Program~~ *Deleted*

~~A training program for the HI-STORM 100 Cask System shall be developed under the general licensee's systematic approach to training (SAT). Training modules shall include comprehensive instructions for the operation and maintenance of the HI-STORM 100 Cask System and the independent spent fuel storage installation (ISFSI).~~

5.2 ~~Pre-Operational Testing and Training Exercise~~ *Deleted*

~~A dry run training exercise of the loading, closure, handling, unloading, and transfer of the HI-STORM 100 Cask System shall be conducted by the licensee prior to the first use of the system to load spent fuel assemblies. The training exercise shall not be conducted with spent fuel in the MPC. The dry run may be performed in an alternate styep sequence from the actual procedures, but all steps must be performed. The dry run shall include, but is not limited to the following:~~

~~a. Moving the MPC and the TRANSFER CASK into the spent fuel pool.~~

~~b. Preparation of the HI-STORM 100 Cask System for fuel loading.~~

~~c. Selection and verification of specific fuel assemblies to ensure type conformance.~~

~~d. Loading specific assemblies and placing assemblies into the MPC (using a dummy fuel assembly), including appropriate independent verification.~~

~~e. Remote installation of the MPC lid and removal of the MPC and TRANSFER CASK from the spent fuel pool.~~

~~f. MPC welding, NDE inspections, hydrostatic testing, draining, vacuum drying, helium backfilling, and leakage testing (for which a mockup may be used).~~

~~(continued)~~

~~ADMINISTRATIVE CONTROLS AND PROGRAMS~~

~~5.2 Pre-Operational Testing and Training Exercise (continued)~~

~~g. TRANSFER CASK upending/downending on the horizontal transfer trailer or other transfer device, as applicable to the site's cask handling arrangement.~~

~~h. Transfer of the MPC from the TRANSFER CASK to the OVERPACK.~~

~~i. Placement of the HI-STORM 100 SFSC System at the ISFSI.~~

~~j. HI-STORM 100 Cask System unloading, including cooling fuel assemblies, flooding MPC cavity, removing MPC lid welds.~~

5.3 ~~Special Requirements For First Systems In Place~~ *Deleted*

~~The heat transfer characteristics of the cask system will be recorded by temperature measurements for the first HI-STORM SFSC Systems (MPC-24, MPC-68, and MPC-68F) placed into service with a heat load equal to or greater than 10 kW. An analysis shall be performed that demonstrates the temperature measurements validate the analytic methods and predicted thermal behavior described in Chapter 4 of the SAR.~~

~~Validation tests shall be performed for each subsequent cask system that has a heat load that exceeds a previously validated heat load by more than 2 kW (e.g., if the initial test was conducted at 10 kW, then no additional testing is needed until the heat load exceeds 12 kW). No additional testing is required for a system after it has been tested at a heat load equal to or greater than 16 kW.~~

~~Letter reports summarizing the results of each validation test shall be submitted to the NRC in accordance with 10 CFR 72.4. Cask users may satisfy these requirements by referencing validation test reports submitted to the NRC by other cask users.~~

(continued)

5.4 Radioactive Effluent Control Program

This program implements the requirements of 10 CFR 72.44(d).

a. The HI-STORM 100 Cask System does not create any radioactive materials or have any radioactive waste treatment systems. Therefore, specific operating procedures for the control of radioactive effluents are not required. Specification 3.1.1, Multi-Purpose Canister (MPC), provides assurance that there are not radioactive effluents from the SFSC.

b. This program includes an environmental monitoring program. Each general license user may incorporate SFSC operations into their environmental monitoring programs for 10 CFR Part 50 operations.

c. An annual report shall be submitted pursuant to 10 CFR 72.44(d)(3).

(continued)

5.5 Cask Transport Evaluation Program

This program provides a means for evaluating various transport configurations and transport route conditions to ensure that the design basis drop limits are met. For lifting of the loaded TRANSFER CASK or OVERPACK using devices which are integral to a structure governed by 10 CFR Part 50 regulations, 10 CFR 50 requirements apply. This program is not applicable when the TRANSFER CASK or OVERPACK is in the FUEL BUILDING or is being handled by a device providing support from underneath (i.e., on a rail car, heavy haul trailer, air pads, etc.).

Pursuant to 10 CFR 72.212, this program shall evaluate the site-specific transport route conditions.

a. *For free-standing OVERPACKS and the TRANSFER CASK, the following requirements apply:*

1. The lift height above the transport *route* surface*(s)* ~~prescribed in Section 3.4.6 of Appendix B to Certificate of Compliance No. 1014~~ shall not exceed the limits in Table 5-1 *except as provided for in Specification 5.5.a.2*. Also, the program shall ensure that the transport route conditions (i.e., surface hardness and pad thickness) are equivalent to or less limiting than *either Set A or Set B in HI-STORM TSAR Table 2.2.9.* ~~those prescribed for the reference pad surface which forms the basis for the values cited in Section 3.4.6 of Appendix B to Certificate of Compliance No. 1014.~~

2. For site-specific transport *route surfaces that* ~~conditions which~~ are not bounded by *either the Set A or Set B parameters of FSAR Table 2.2.9,* ~~*the surface characteristics in Section 3.4.6 of Appendix B to Certificate of Compliance No. 1014*~~, the program may ~~evaluate~~ *determine lift heights by analysis based on* the site-specific conditions to ensure that the impact loading due to design basis drop events does not exceed 45 g*'s at the top of the MPC fuel basket*. ~~This~~ *These* alternative *analyses* shall be commensurate with the drop analyses described in the Topical Safety Analysis Report for the HI-STORM 100 Cask System. The program shall ensure that these alternative analyses are documented and controlled.

(continued)

ADMINISTRATIVE CONTROLS AND PROGRAMS

5.5 Cask Transport Evaluation Program (continued)

3. *The TRANSFER CASK or OVERPACK, when loaded with spent fuel, may be lifted to any height necessary during transportation between the FUEL BUILDING and the CTF and/or ISFSI pad, provided the lifting device is designed in accordance with ANSI N14.6 and has redundant drop protection features.*

4. The TRANSFER CASK and MPC, when loaded with spent fuel, may be lifted to those heights necessary to perform cask handling operations, including MPC transfer, provided the lifts are made with structures and components designed in accordance with the criteria specified in Section 3.5 of Appendix B to Certificate of Compliance No. 1014, as applicable.

[redacted]

SECTION DELETED IN ORIGINAL DOCUMENT
(NOT WITHHELD, HERE)

(continued)

ADMINISTRATIVE CONTROLS AND PROGRAMS

5.5 Cask Transport Evaluation Program (continued)

Table 5-1

TRANSFER CASK and *Free-Standing* OVERPACK Lifting Requirements

ITEM	ORIENTATION	LIFTING HEIGHT LIMIT (in.)
TRANSFER CASK	Horizontal	42 (Notes 1 *and 2*)
TRANSFER CASK	Vertical	None Established (Note 2)
OVERPACK	Horizontal	Not Permitted
OVERPACK	Vertical	11 (Note ~~2~~ *3*)

Notes:
1. To be measured from the lowest point on the TRANSFER CASK (i.e., the bottom edge of the transfer lid)
2. See Technical Specification 5.5.*a.3 and 4*
3. *See Technical Specification 5.5.a.3.*

1.0 Definitions

---NOTE---
The defined terms of this section appear in capitalized type and are applicable throughout these Technical Specifications and Bases.

Term	Definition
CASK TRANSFER FACILITY (CTF)	The CASK TRANSFER FACILITY includes the following components and equipment: (1) a Cask Transfer Structure used to stabilize the TRANSFER CASK and MPC during lifts involving spent fuel not bounded by the regulations of 10 CFR Part 50, and (2) Either a stationary lifting device or a mobile lifting device used in concert with the stationary structure to lift the OVERPACK, TRANSFER CASK, and MPC
DAMAGED FUEL ASSEMBLY	DAMAGED FUEL ASSEMBLIES are fuel assemblies with known or suspected cladding defects, as determined by a review of records, greater than pinhole leaks or hairline cracks, ~~missing~~ *empty* fuel rod *location*s that are not ~~replaced~~ *filled* with dummy *fuel* rods, or those that cannot be handled by normal means. Fuel assemblies which cannot be handled by normal means due to fuel cladding damage are considered FUEL DEBRIS.
DAMAGED FUEL CONTAINER (DFC)	DFCs are specially designed enclosures for DAMAGED FUEL ASSEMBLIES or FUEL DEBRIS which permit gaseous and liquid media to escape while minimizing dispersal of gross particulates. *DFCs authorized for use in the HI-STORM 100 System are as follows:* *1. Holtec Dresden Unit 1/Humboldt Bay design* *2. Transnuclear Dresden Unit 1 design* *3. Holtec Generic BWR design* *4. Holtec Generic PWR design*
FUEL DEBRIS	FUEL DEBRIS is ruptured fuel rods, severed rods, loose fuel pellets or fuel assemblies with known or suspected defects which cannot be handled by normal means due to fuel cladding damage.

(continued)

1.0 Definitions (continued)

INTACT FUEL ASSEMBLY	INTACT FUEL ASSEMBLIES are fuel assemblies without known or suspected cladding defects greater than pinhole leaks or hairline cracks and which can be handled by normal means. ~~Partial fuel assemblies, that is~~ Fuel assemblies *without fuel rods in fuel rod locations* ~~from which fuel rods are missing,~~ shall not be classified as INTACT FUEL ASSEMBLIES unless dummy fuel rods are used to displace an amount of water greater than or equal to that displaced by the ~~original~~ fuel rod(s).
LOADING OPERATIONS	LOADING OPERATIONS include all licensed activities on an OVERPACK or TRANSFER CASK while it is being loaded with fuel assemblies. LOADING OPERATIONS begin when the first fuel assembly is placed in the MPC and end when the OVERPACK or TRANSFER CASK is suspended from or secured on the transporter. LOADING OPERATIONS does not included MPC transfer between the TRANSFER CASK and the OVERPACK.
MULTI-PURPOSE CANISTER (MPC)	MPCs are the sealed spent nuclear fuel canisters which consist of a honeycombed fuel basket contained in a cylindrical canister shell which is welded to a baseplate, lid with welded port cover plates, and closure ring. The MPC provides the confinement boundary for the contained radioactive materials.
NON-FUEL HARDWARE	*NON-FUEL HARDWARE is defined as Burnable Poison Rod Assemblies (BPRAs), Thimble Plug Devices (TPDs), Control Rod Assemblies (CRAs), Axial Power Shaping Rods (APSRs) and other similarly designed devices with different names.*
OVERPACK	OVERPACKs are the casks which receive and contain the sealed MPCs for interim storage on the ISFSI. They provide gamma and neutron shielding, and provide for ventilated air flow to promote heat transfer from the MPC to the environs. The OVERPACK does not include the TRANSFER CASK.

(continued)

1.0 Definitions (continued)

PLANAR-AVERAGE INITIAL ENRICHMENT	PLANAR-AVERAGE INITIAL ENRICHMENT is the average of the distributed fuel rod initial enrichments within a given axial plane of the assembly lattice.
SPENT FUEL STORAGE CASKS (SFSCs)	An SFSC is a container approved for the storage of spent fuel assemblies at the ISFSI. The HI-STORM 100 SFSC System consists of the OVERPACK and its integral MPC.
TRANSFER CASK	TRANSFER CASKs are containers designed to contain the MPC during and after loading of spent fuel assemblies and to transfer the MPC to or from the OVERPACK. The HI-STORM 100 System employs either the 125-Ton or the 100-Ton HI-TRAC TRANSFER CASK.
TRANSPORT OPERATIONS	TRANSPORT OPERATIONS include all licensed activities performed on an OVERPACK or TRANSFER CASK loaded with one or more fuel assemblies when it is being moved to and from the ISFSI. TRANSPORT OPERATIONS begin when the OVERPACK or TRANSFER CASK is first suspended from or secured on the transporter and end when the OVERPACK or TRANSFER CASK is at its destination and no longer secured on or suspended from the transporter. TRANSPORT OP RATIONS includes transfer of the MPC between the OVERPACK and the TRANSFER CASK.
UNLOADING OPERATIONS	UNLOADING OPERATIONS include all licensed activities on an SFSC to be unloaded of the contained fuel assemblies. UNLOADING OPERATIONS begin when the OVERPACK or TRANSFER CASK is no longer suspended from or secured on the transporter and end when the last fuel assembly is removed from the SFSC. UNLOADING OPERATIONS does not include MPC transfer between the TRANSFER CASK and the OVERPACK.

2.0 APPROVED CONTENTS

2.1 Fuel Specifications and Loading Conditions

2.1.1 Fuel To Be Stored In The HI-STORM 100 SFSC System

a. INTACT FUEL ASSEMBLIES, DAMAGED FUEL ASSEMBLIES, ~~and~~ FUEL DEBRIS, *and NON-FUEL HARDWARE* meeting the limits specified in Table 2.1-1 *and other referenced tables* may be stored in the HI-STORM 100 SFSC System.

b. For MPCs partially loaded with stainless steel clad fuel assemblies, all remaining fuel assemblies in the MPC shall meet the decay heat generation limit for the stainless steel clad fuel assemblies.

c. For MPCs partially loaded with DAMAGED FUEL ASSEMBLIES or FUEL DEBRIS, all remaining Zircaloy clad INTACT FUEL ASSEMBLIES in the MPC shall meet the decay heat generation limits for the DAMAGED FUEL ASSEMBLIES. *This requirement applies only to uniform fuel loading.*

d. For MPC-68's partially loaded with array/class 6x6A, 6x6B, 6x6C, or 8x8A fuel assemblies, all remaining Zircaloy clad INTACT FUEL ASSEMBLIES in the MPC shall meet the decay heat generation limits for the 6x6A, 6x6B, 6x6C, *7x7A* and 8x8A fuel assemblies.

e. All BWR fuel assemblies may be stored with or without Zircaloy channels with the exception of array/class 10x10D and 10x10E fuel assemblies, which may be stored with or without Zircaloy or stainless steel channels.

(continued)

2.0 Approved Contents ~~(continued)~~

2.1 *Fuel Specifications and Loading Conditions (cont'd)*

2.1.2 ~~Preferential~~ *Uniform* Fuel Loading

Preferential fuel loading shall be used *during uniform loading (i.e., any authorized fuel assembly in any fuel storage location)* whenever fuel assemblies with significantly different post-irradiation cooling times ($\geq$ 1 year) are to be loaded in the same MPC. Fuel assemblies with the longest post-irradiation cooling times shall be loaded into fuel storage locations at the periphery of the basket. Fuel assemblies with shorter post-irradiation cooling times shall be placed toward the center of the basket. *Regionalized fuel loading as described in Technical Specification 2.1.3 below meets the intent of preferential fuel loading.*

2.1.3 *Regionalized Fuel Loading*

Users may choose to store fuel using regionalized loading in lieu of uniform loading to allow higher heat emitting fuel assemblies to be stored than would otherwise be able to be stored using uniform loading. Regionalized loading is limited to those fuel assemblies with Zircaloy (or other alloy of zirconium) cladding. Figures 2.1-1 through 2.1-4 define the regions for the MPC-24, MPC-24E, MPC-24EF, MPC-32, MPC-68, and MPC-68FF models, respectively. Fuel assembly burnup, decay heat, and cooling time limits for regionalized loading are specified in Tables 2.1-6 and 2.1-7. Fuel assemblies used in regionalized loading shall meet all other applicable limits specified in Tables 2.1-1 through 2.1-3.

2.2 Violations

If any Fuel Specifications or Loading Conditions of 2.1 are violated, the following actions shall be completed:

2.2.1 The affected fuel assemblies shall be placed in a safe condition.

2.2.2 Within 24 hours, notify the NRC Operations Center.

2.2.3 Within 30 days, submit a special report which describes the cause of the violation, and actions taken to restore compliance and prevent recurrence.

LEGEND:

REGION 1:

REGION 2:

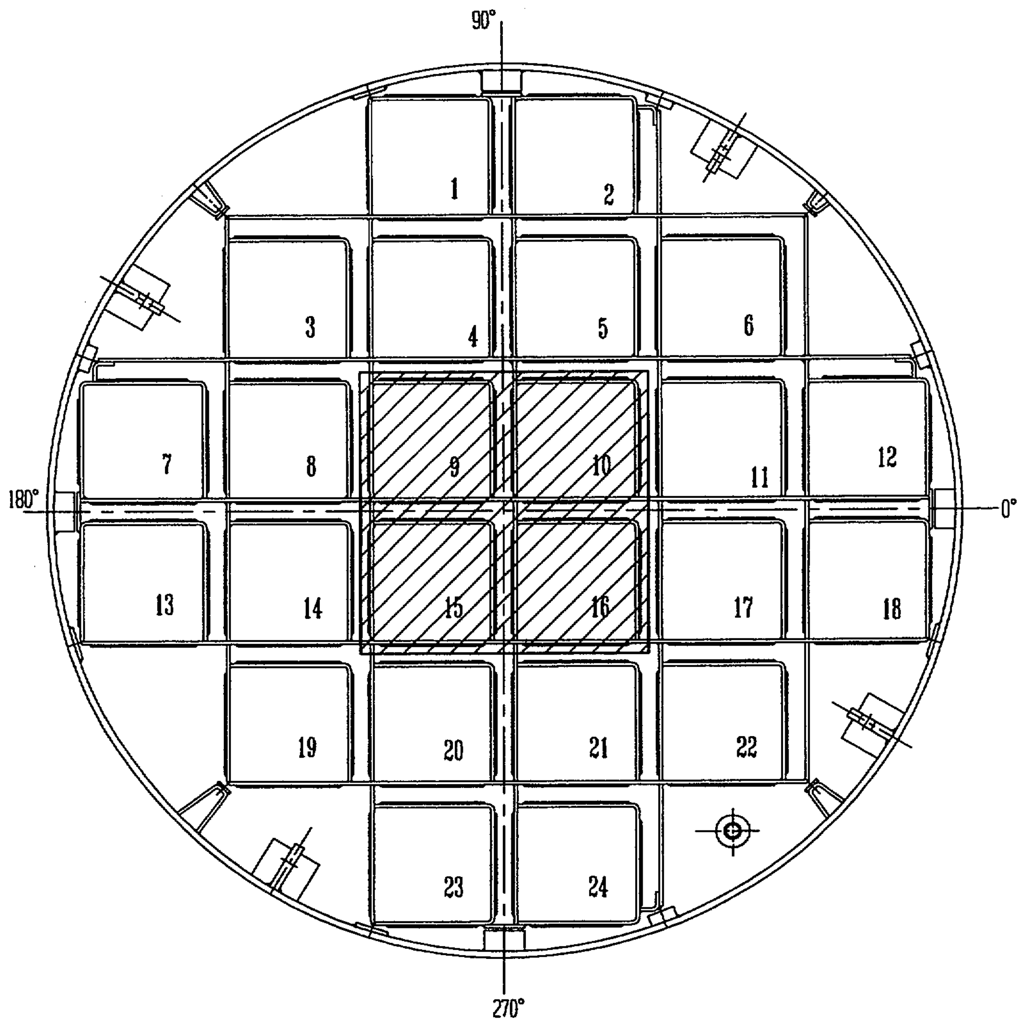

FIGURE 2.1-1
FUEL LOADING REGIONS - MPC-24

LEGEND:

REGION 1: (hatched)

REGION 2: (blank)

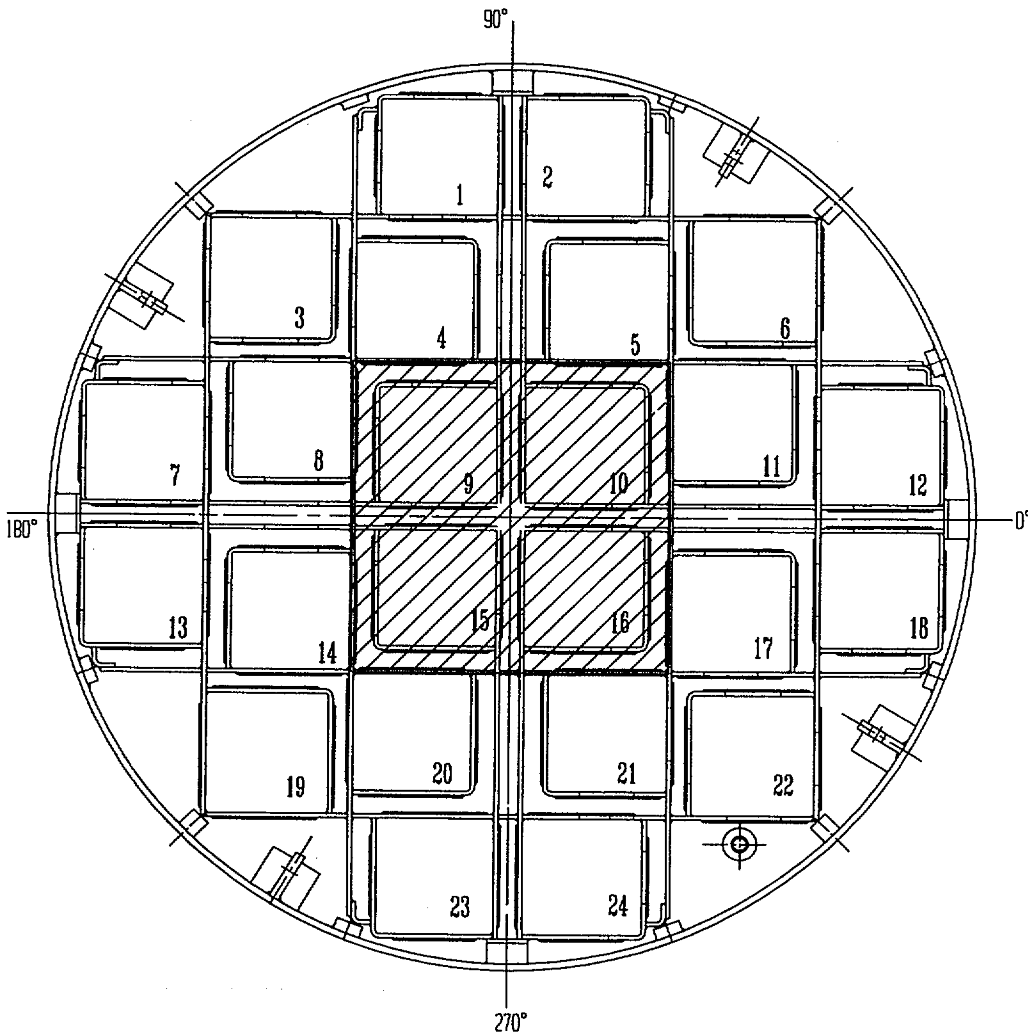

FIGURE 2.1-2

FUEL LOADING REGIONS - MPC-24E/24EF

LEGEND:

REGION 1: (hatched)

REGION 2: (blank)

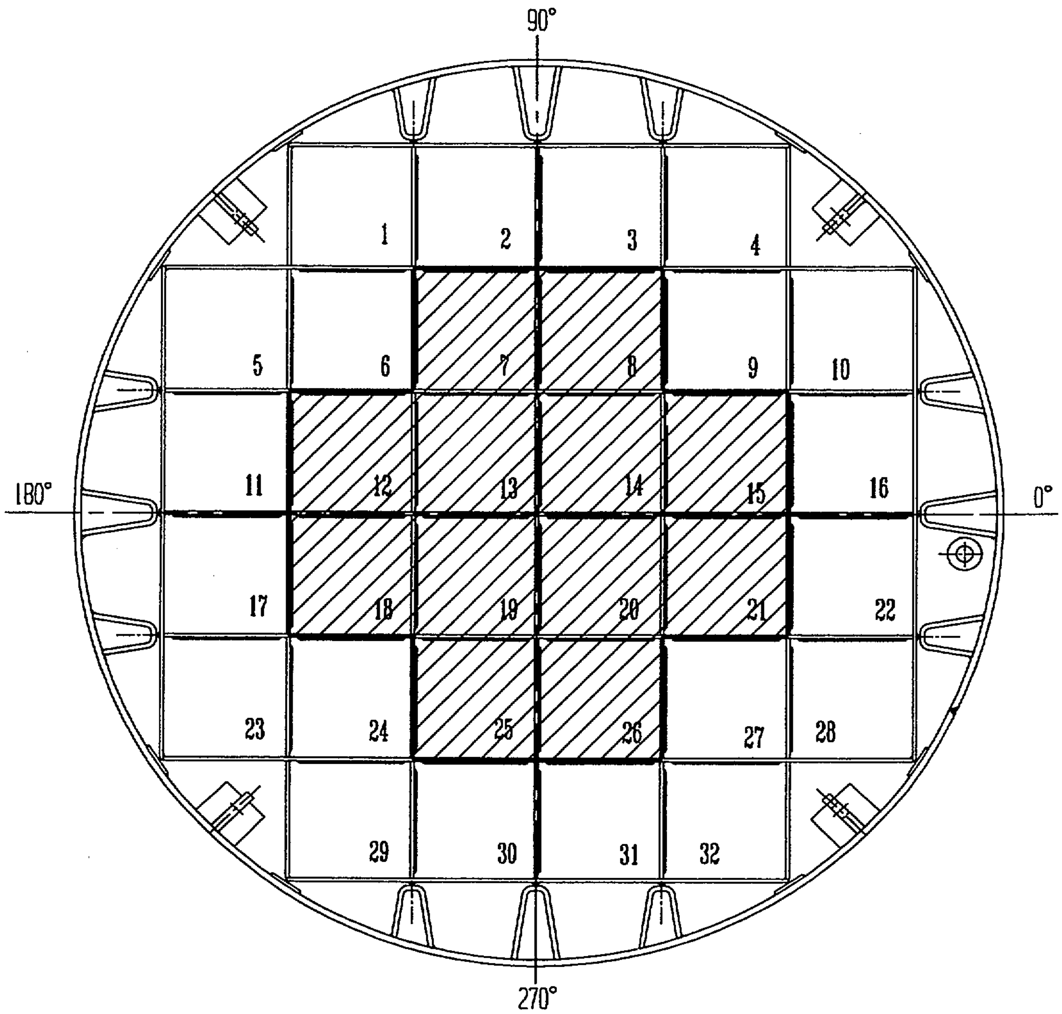

FIGURE 2.1-3

FUEL LOADING REGIONS - MPC-32

LEGEND:

REGION 1: (hatched)

REGION 2: (blank)

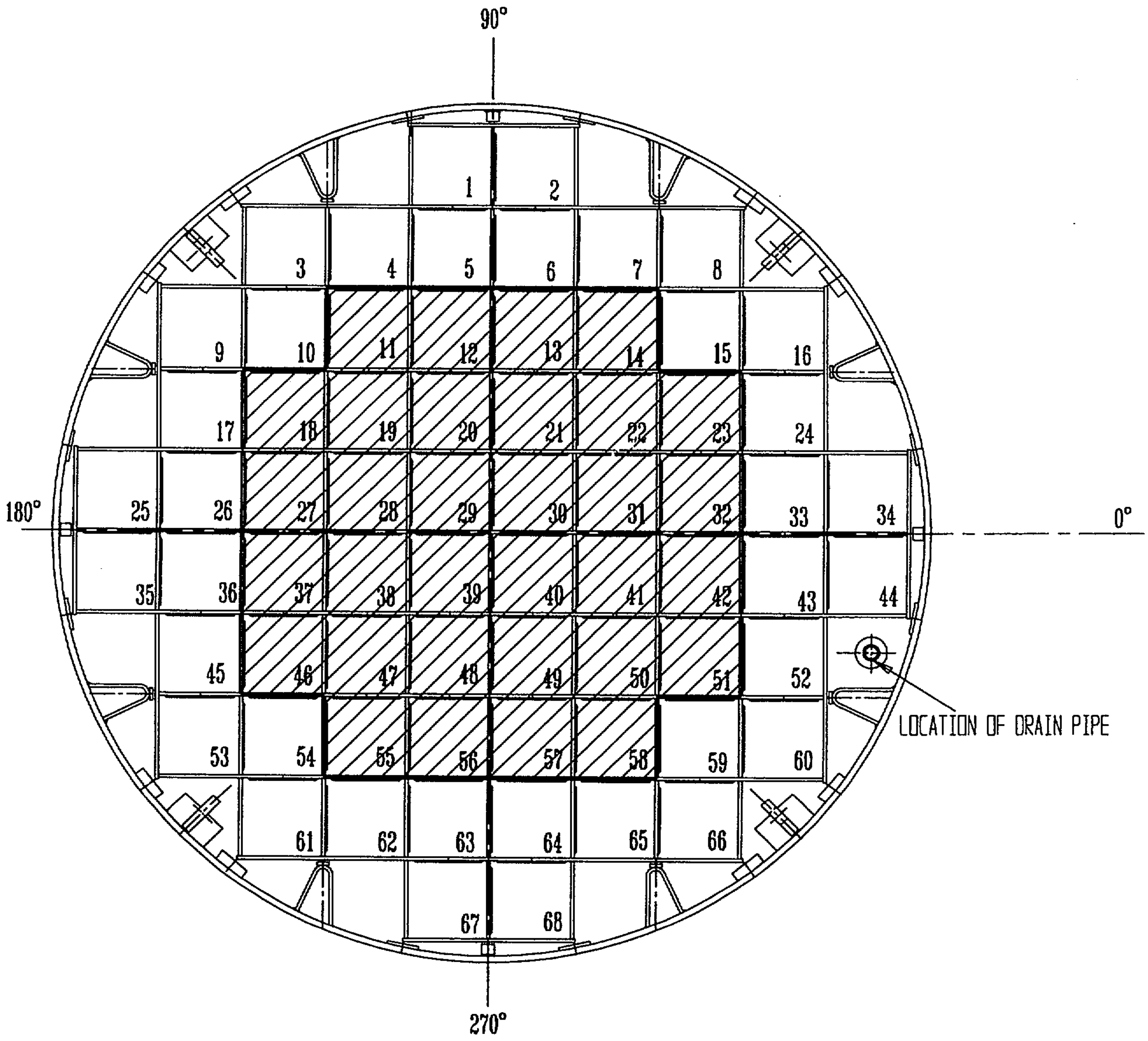

FIGURE 2.1-4

FUEL LOADING REGIONS - MPC-68/68FF

Table 2.1-1 (page 1 of ~~14~~ *33*)
Fuel Assembly Limits

I. MPC MODEL: MPC-24

A. Allowable Contents

1. Uranium oxide, PWR INTACT FUEL ASSEMBLIES listed in Table 2.1-2, *with or without NON-FUEL HARDWARE* and meeting the following specifications *(Note 1)*:

a. Cladding Type:	Zircaloy (Zr) or Stainless Steel (SS) as specified in Table 2.1-2 for the applicable fuel assembly array/class.
b. Initial Enrichment:	As specified in Table 2.1-2 for the applicable fuel assembly array/class.
c. Post-irradiation Cooling Time and Average Burnup Per Assembly:	
i. ~~Zr Clad:~~ *Array/Classes 14x14D, 14x14E, and 15x15G*	~~An assembly post-irradiation~~ Cooling time ≥ 8 years and an average burnup ≤ 40,000 MWD/MTU.
ii. ~~SS Clad:~~ *All Other Array/Classes*	~~An assembly post-irradiation~~ Cooling time and average burnup as specified in Tables 2.1-4 *or 2.1-6*.
iii. *NON-FUEL HARDWARE*	*As specified in Table 2.1-8.*

Table 2.1-1 (page 2 of ~~14~~ *33*)
Fuel Assembly Limits

I. MPC MODEL: MPC-24 (continued)

A. Allowable Contents (continued)

d. Decay Heat Per Assembly:	
i. ~~Zr Clad~~ *Array/Classes 14x14D, 14x14E, and 15x15G*	≤ 710 Watts
ii ~~SS Clad~~ All Other Array/Classes	~~An assembly decay heat~~ As specified in Tables 2.1-5 *or 2.1-7* ~~for the applicable post-irradiation cooling time~~.
e. Fuel Assembly Length:	≤ 176.8 inches (nominal design)
f. Fuel Assembly Width:	≤ 8.54 inches (nominal design)
g. Fuel Assembly Weight:	≤ 1,680 lbs *(including NON-FUEL HARDWARE)*

B. Quantity per MPC: Up to 24 fuel assemblies.

C. ~~Fuel assemblies shall not contain control components.~~ *Deleted.*

D. DAMAGED FUEL ASSEMBLIES and FUEL DEBRIS are not authorized for loading into the MPC-24.

Note 1: Fuel assemblies containing BPRAs or TPDs may be stored in any fuel cell location. Fuel assemblies containing CRAs or APSRs may only be loaded in fuel storage locations 9, 10, 15, and/or 16. These requirements are in addition to any other requirements specified for uniform or regionalized fuel loading.

Table 2.1-1 (page 3 of ~~14~~ *33*)
Fuel Assembly Limits

II. MPC MODEL: MPC-68

A. Allowable Contents

1. Uranium oxide, BWR INTACT FUEL ASSEMBLIES listed in Table 2.1-3, with or without ~~Zircaloy~~ channels, and meeting the following specifications:

a. Cladding Type:	Zircaloy (Zr) or Stainless Steel (SS) as specified in Table 2.1-3 for the applicable fuel assembly array/class.
b. Maximum PLANAR-AVERAGE INITIAL ENRICHMENT:	As specified in Table 2.1-3 for the applicable fuel assembly array/class.
c. Initial Maximum Rod Enrichment:	As specified in Table 2.1-3 for the applicable fuel assembly array/class.
d. Post-irradiation Cooling Time and Average Burnup Per Assembly:	
i. ~~Zr Clad:~~ *Array/Classes 6x6A, 6x6C, 7x7A, and 8x8A*:	~~An assembly post-irradiation~~ Cooling time ≥ 18 years and an average burnup ≤ 30,000 MWD/MTU
ii. ~~SS Clad:~~ *Array/Class 8x8F*	*Cooling time ≥ 10 years and an average burnup ≤ 27,500 MWD/MTU.*
iii. Array/Classes 10x10D and 10x10E	~~An assembly post-irradiation~~ Cooling time ≥ 10 years and an average burnup ≤ 22,500 MWD/MTU.
iv. All Other Array/Classes	~~An assembly post-irradiation cooling time and average burnup~~ As specified in Tables 2.1-4 *or 2.1-6.*

Table 2.1-1 (page 4 of ~~14~~ *33*)
Fuel Assembly Limits

II. MPC MODEL: MPC-68 (continued)

A. Allowable Contents (continued)

e. Decay Heat Per Assembly:

i. ~~Zr Clad:~~ *Array/Classes 6x6A, 6x6C, 7x7A, and 8x8A*	≤ 115 Watts
ii. ~~SS Clad:~~ *Array/Class 8x8F*	≤ *183.5 Watts.*
iii. *Array/Classes 10x10D and 10x10E*	≤ 95 Watts
iv. All Other Array/Classes	~~An assembly maximum decay heat~~ As specified in Tables 2.1-5 *or 2.1-7.*

f. Fuel Assembly Length:	≤ ~~176.2~~ *176.5* inches (nominal design)
g. Fuel Assembly Width:	≤ 5.85 inches (nominal design)
h. Fuel Assembly Weight:	≤ 700 lbs, including channels

Table 2.1-1 (page 5 of ~~14~~ *33*)
Fuel Assembly Limits

II. MPC MODEL: MPC-68 (continued)

A. Allowable Contents (continued)

2. Uranium oxide, BWR DAMAGED FUEL ASSEMBLIES, with or without ~~Zircaloy~~ channels, placed in DAMAGED FUEL CONTAINERS. Uranium oxide BWR DAMAGED FUEL ASSEMBLIES shall meet the criteria specified in Table 2.1-3 ~~for fuel assembly array/class 6x6A, 6x6C, 7x7A, or 8x8A,~~ and meet the following specifications:

a. Cladding Type:	Zircaloy (Zr) *or Stainless Steel (SS) as specified in Table 2.1-3 for the applicable fuel assembly array/class.*
b. Maximum PLANAR-AVERAGE INITIAL ENRICHMENT:	
i. Array/Classes 6x6A, 6x6C, 7x7A, and 8x8A	As specified in Table 2.1-3 for the applicable fuel assembly array/class.
ii. All Other Array/Classes specified in Table 2.1-3	*4.0 wt% ^{235}U*
c. Initial Maximum Rod Enrichment:	As specified in Table 2.1-3 for the applicable fuel assembly array/class.
d. Post-irradiation Cooling Time and Average Burnup Per Assembly:	
i. Array/Classes 6x6A, 6x6C, 7x7A,and 8x8A	*Cooling time ≥ 18 years and an average burnup ≤ 30,000 MWD/MTU.*
ii. Array/Class 8x8F	*Cooling time ≥ 10 years and an average burnup ≤ 27,500 MWD/MTU.*
iii. Array/Classes 10x10D and 10x10E	*Cooling time ≥ 10 years and an average burnup ≤ 22,500 MWD/MTU.*
iv. All Other Array Classes	*As specified in Tables 2.1-4 or 2.1-6.*

Table 2.1-1 (page 6 of 33)
Fuel Assembly Limits

II. MPC MODEL: MPC-68 (continued)

A. Allowable Contents (continued)

e. Decay Heat Per Assembly:

i.	*Array/Class 6x6A, 6x6C, 7x7A, and 8x8A*	*≤ 115 Watts*
ii.	*Array/Class 8x8F*	*≤ 183.5 Watts*
iii.	*Array/Classes 10x10D and 10x10E*	*≤ 95 Watts*
iv.	*All Other Array/Classes*	*As specified in Tables 2.1-5 or 2.1-7*

f. Fuel Assembly Length:

i.	*Array/Class 6x6A, 6x6C, 7x7A, or 8x8A*	*≤ 135.0 inches (nominal design)*
ii.	*All Other Array/Classes*	*≤ 176.5 inches (nominal design)*

g. Fuel Assembly Width:

i.	*Array/Class 6x6A, 6x6C, 7x7A, or 8x8A*	*≤ 4.70 inches (nominal design)*
ii.	*All Other Array/Classes*	*≤ 5.85 inches (nominal design)*

h. Fuel Assembly Weight:

i.	*Array/Class 6x6A, 6x6C, 7x7A, or 8x8A*	*≤ 550 lbs, including channels and DFC*
ii.	*All Other Array/Classes*	*≤ 700 lbs, including channels and DFC*

Table 2.1-1 (page 7 of ~~14~~ *33*)
Fuel Assembly Limits

II. MPC MODEL: MPC-68 (continued)

A. Allowable Contents (continued)

3. Mixed oxide (MOX), BWR INTACT FUEL ASSEMBLIES, with or without ~~Zircaloy~~ channels. MOX BWR INTACT FUEL ASSEMBLIES shall meet the criteria specified in Table 2.1-3 for fuel assembly array/class 6x6B, and meet the following specifications:

a. Cladding Type:	Zircaloy (Zr)
b. Maximum PLANAR-AVERAGE INITIAL ENRICHMENT:	As specified in Table 2.1-3 for fuel assembly array/class 6x6B.
c. Initial Maximum Rod Enrichment:	As specified in Table 2.1-3 for fuel assembly array/class 6x6B.
d. Post-irradiation Cooling Time and Average Burnup Per Assembly:	~~An assembly post-irradiation~~ Cooling time $\geq$ 18 years and an average burnup $\leq$ 30,000 MWD/MTIHM.
e. Decay Heat Per Assembly:	$\leq$ 115 Watts
f. Fuel Assembly Length:	$\leq$ 135.0 inches (nominal design)
g. Fuel Assembly Width:	$\leq$ 4.70 inches (nominal design)
h. Fuel Assembly Weight:	$\leq$ 400 lbs, including channels

Table 2.1-1 (page 8 of ~~14~~ *33*)
Fuel Assembly Limits

II. MPC MODEL: MPC-68 (continued)

A. Allowable Contents (continued)

4. Mixed oxide (MOX), BWR DAMAGED FUEL ASSEMBLIES, with or without ~~Zircaloy~~ channels, placed in DAMAGED FUEL CONTAINERS. MOX BWR DAMAGED FUEL ASSEMBLIES shall meet the criteria specified in Table 2.1-3 for fuel assembly array/class 6x6B, and meet the following specifications:

a. Cladding Type:	Zircaloy (Zr)
b. Maximum PLANAR-AVERAGE INITIAL ENRICHMENT:	As specified in Table 2.1-3 for array/class 6x6B.
c. Initial Maximum Rod Enrichment:	As specified in Table 2.1-3 for array/class 6x6B.
d. Post-irradiation Cooling Time and Average Burnup Per Assembly:	~~An assembly post-irradiation~~ Cooling time $\geq$ 18 years and an average burnup $\leq$ 30,000 MWD/MTIHM.
e. Decay Heat Per Assembly:	$\leq$ 115 Watts
f. Fuel Assembly Length:	$\leq$ 135.0 inches (nominal design)
g. Fuel Assembly Width:	$\leq$ 4.70 inches (nominal design)
h. Fuel Assembly Weight:	$\leq$ ~~400~~ *550* lbs, including channels *and DFC*

Table 2.1-1 (page 9 of 33)
Fuel Assembly Limits

II. MPC MODEL: MPC-68 (continued)

A. Allowable Contents (continued)

5. *Thoria rods (ThO_2 and UO_2) placed in Dresden Unit 1 Thoria Rod Canisters and meeting the following specifications:*

a. Cladding Type:	Zircaloy (Zr)
b. Composition:	98.2 wt.% ThO_2, 1.8 wt. % UO_2 with an enrichment of 93.5 wt. % ^{235}U.
c. Number of Rods Per Thoria Rod Canister:	≤ 18
d. Decay Heat Per Thoria Rod Canister:	≤ 115 Watts
e. Post-irradiation Fuel Cooling Time and Average Burnup Per Thoria Rod Canister:	A fuel post-irradiation cooling time ≥ 18 years and an average burnup ≤ 16,000 MWD/MTIHM.
f. Initial Heavy Metal Weight:	≤ 27 kg/canister
g. Fuel Cladding O.D.:	≥ 0.412 inches
h. Fuel Cladding I.D.:	≤ 0.362 inches
i. Fuel Pellet O.D.:	≤ 0.358 inches
j. Active Fuel Length:	≤ 111 inches
k. Canister Weight:	≤ 550 lbs, including fuel

Table 2.1-1 (page 10 of 33)
Fuel Assembly Limits

II. MPC MODEL: MPC-68 (continued)

B. Quantity per MPC:

1. *Up to one (1) Dresden Unit 1 Thoria Rod Canister;*

2. *Up to 68 array/class 6x6A, 6x6B, 6x6C, 7x7A, or 8x8A DAMAGED FUEL ASSEMBLIES in DAMAGE FUEL CONTAINERS;*

3. *Up to sixteen (16) other BWR DAMAGED FUEL ASSEMBLIES in DAMAGED FUEL CONTAINERS in fuel storage locations 1, 2, 3, 8, 9, 16, 25, 34, 35, 44, 53, 60, 61, 66, 67, and/or 68; and/or*

4. *Any number of BWR INTACT FUEL ASSEMBLIES up to a total of 68.*

C. ~~Fuel assemblies with stainless steel channels are not authorized for loading in the MPC-68.~~ *Array/Class 10x10D and 10x10E fuel assemblies in stainless steel channels must be stored in fuel storage locations 19 - 22, 28 - 31, 38 -41, and/or 47 - 50.*

D. Dresden Unit 1 fuel assemblies with one Antimony-Beryllium neutron source are authorized for loading in the MPC-68. The Antimony-Beryllium source material shall be in a water rod location.

Table 2.1-1 (page ~~8~~ *11* of ~~14~~ *33*)
Fuel Assembly Limits

III. MPC MODEL: MPC-68F

A. Allowable Contents

1. Uranium oxide, BWR INTACT FUEL ASSEMBLIES, with or without Zircaloy channels. Uranium oxide BWR INTACT FUEL ASSEMBLIES shall meet the criteria *specified* in Table 2.1-3 for fuel assembly array class 6x6A, 6x6C, 7x7A or 8x8A, and meet the following specifications:

a. Cladding Type:	Zircaloy (Zr)
b Maximum PLANAR-AVERAGE INITIAL ENRICHMENT:	As specified in Table 2.1-3 for the applicable fuel assembly array/class.
c. Initial Maximum Rod Enrichment:	As specified in Table 2.1-3 for the applicable fuel assembly array/class.
d. Post-irradiation Cooling Time and Average Burnup Per Assembly:	~~An assembly post-irradiation~~ Cooling time ≥ 18 years and an average burnup ≤ 30,000 MWD/MTU.
e. Decay Heat Per Assembly	≤ 115 Watts
f. Fuel Assembly Length:	≤ ~~176.2~~ *135.0* inches (nominal design)
g. Fuel Assembly Width:	≤ ~~5.85~~ *4.70* inches (nominal design)
h. Fuel Assembly Weight:	≤ ~~700~~ *400* lbs, including channels

Table 2.1-1 (page ~~9~~ *12* of ~~14~~ *33*)
Fuel Assembly Limits

III. MPC MODEL: MPC-68F (continued)

A. Allowable Contents (continued)

2. Uranium oxide, BWR DAMAGED FUEL ASSEMBLIES, with or without Zircaloy channels, placed in DAMAGED FUEL CONTAINERS. Uranium oxide BWR DAMAGED FUEL ASSEMBLIES shall meet the criteria specified in Table 2.1-3 for fuel assembly array/class 6x6A, 6x6C, 7x7A, or 8x8A, and meet the following specifications:

a. Cladding Type:	Zircaloy (Zr)
b. Maximum PLANAR-AVERAGE INITIAL ENRICHMENT:	As specified in Table 2.1-3 for the applicable fuel assembly array/class.
c. Initial Maximum Rod Enrichment:	As specified in Table 2.1-3 for the applicable fuel assembly array/class.
d. Post-irradiation Cooling Time and Average Burnup Per Assembly:	*Cooling time* ≥ 18 years and an average burnup ≤ 30,000 MWD/MTU.
e. Decay Heat Per Assembly:	≤ 115 Watts
f. Fuel Assembly Length:	≤ 135.0 inches (nominal design)
g. Fuel Assembly Width:	≤ 4.70 inches (nominal design)
h. Fuel Assembly Weight:	≤ ~~400~~ *550* lbs, including channels *and DFC*

Table 2.1-1 (page ~~10~~ *13* of ~~14~~ *33*)
Fuel Assembly Limits

III. MPC MODEL: MPC-68F (continued)

A. Allowable Contents (continued)

3. Uranium oxide, BWR FUEL DEBRIS, with or without Zircaloy channels, placed in DAMAGED FUEL CONTAINERS. The original fuel assemblies for the uranium oxide BWR FUEL DEBRIS shall meet the criteria specified in Table 2.1-3 for fuel assembly array/class 6x6A, 6x6C, 7x7A, or 8x8A, and meet the following specifications:

a. Cladding Type:	Zircaloy (Zr)
b. Maximum PLANAR-AVERAGE INITIAL ENRICHMENT:	As specified in Table 2.1-3 for the applicable original fuel assembly array/class.
c Initial Maximum Rod Enrichment:	As specified in Table 2.1-3 for the applicable original fuel assembly array/class.
d. Post-irradiation Cooling Time and Average Burnup Per Assembly	~~A post-irradiation~~ Cooling time ~~after discharge~~ $\geq$ 18 years and an average burnup $\leq$ 30,000 MWD/MTU for the original fuel assembly.
e. Decay Heat Per Assembly	$\leq$ 115 Watts
f. Original Fuel Assembly Length	$\leq$ 135.0 inches (nominal design)
g. Original Fuel Assembly Width	$\leq$ 4.70 inches (nominal design)
h. Fuel Debris Weight	$\leq$ ~~400~~ *550* lbs, including channels *and DFC*

Table 2.1-1 (page ~~11~~ *14* of ~~14~~ *33*)
Fuel Assembly Limits

III. MPC MODEL: MPC-68F (continued)

A. Allowable Contents (continued)

4. Mixed oxide (MOX), BWR INTACT FUEL ASSEMBLIES, with or without Zircaloy channels. MOX BWR INTACT FUEL ASSEMBLIES shall meet the criteria specified in Table 2.1-3 for fuel assembly array/class 6x6B, and meet the following specifications:

a. Cladding Type:	Zircaloy (Zr)
b. Maximum PLANAR-AVERAGE INITIAL ENRICHMENT:	As specified in Table 2.1-3 for fuel assembly array/class 6x6B.
c. Initial Maximum Rod Enrichment:	As specified in Table 2.1-3 for fuel assembly array/class 6x6B.
d. Post-irradiation Cooling Time and Average Burnup Per Assembly:	~~An assembly post-irradiation~~ Cooling time ~~after discharge~~ $\geq$ 18 years and an average burnup $\leq$ 30,000 MWD/MTIHM.
e. Decay Heat Per Assembly	$\leq$ 115 Watts
f. Fuel Assembly Length:	$\leq$ 135.0 inches (nominal design)
g. Fuel Assembly Width:	$\leq$ 4.70 inches (nominal design)
h. Fuel Assembly Weight:	$\leq$ 400 lbs, including channels

Table 2.1-1 (page ~~12~~ *15* of ~~14~~ *33*)
Fuel Assembly Limits

III. MPC MODEL: MPC-68F (continued)

A. Allowable Contents (continued)

5. Mixed oxide (MOX), BWR DAMAGED FUEL ASSEMBLIES, with or without Zircaloy channels, placed in DAMAGED FUEL CONTAINERS. MOX BWR DAMAGED FUEL ASSEMBLIES shall meet the criteria specified in Table 2.1-3 for fuel assembly array/class 6x6B, and meet the following specifications:

a. Cladding Type:	Zircaloy (Zr)
b. Maximum PLANAR-AVERAGE INITIAL ENRICHMENT:	As specified in Table 2.1-3 for fuel assembly array/class 6x6B.
c. Initial Maximum Rod Enrichment:	As specified in Table 2.1-3 for fuel assembly array/class 6x6B.
d. Post-irradiation Cooling Time and Average Burnup Per Assembly:	~~A post-irradiation~~ Cooling time ~~after discharge~~ $\geq$ 18 years and an average burnup $\leq$ 30,000 MWD/MTIHM.
e. Decay Heat Per Assembly	$\leq$ 115 Watts
f. Fuel Assembly Length:	$\leq$ 135.0 inches (nominal design)
g. Fuel Assembly Width:	$\leq$ 4.70 inches (nominal design)
h. Fuel Assembly Weight:	$\leq$ ~~400~~ *550* lbs, including channels *and DFC*

Table 2.1-1 (page ~~13~~ *16* of ~~14~~ *33*)
Fuel Assembly Limits

III. MPC MODEL: MPC-68F (continued)

A. Allowable Contents (continued)

6. Mixed Oxide (MOX), BWR FUEL DEBRIS, with or without Zircaloy channels, placed in DAMAGED FUEL CONTAINERS. The original fuel assemblies for the MOX BWR FUEL DEBRIS shall meet the criteria specified in Table 2.1-3 for fuel assembly array/class 6x6B, and meet the following specifications:

a. Cladding Type:	Zircaloy (Zr)
b. Maximum PLANAR-AVERAGE INITIAL ENRICHMENT:	As specified in Table 2.1-3 for original fuel assembly array/class 6x6B.
c. Initial Maximum Rod Enrichment:	As specified in Table 2.1-3 for original fuel assembly array/class 6x6B.
d. Post-irradiation Cooling Time and Average Burnup Per Assembly:	~~A post-irradiation~~ Cooling time ~~after discharge~~ ≥ 18 years and an average burnup ≤ 30,000 MWD/MTIHM for the original fuel assembly.
e. Decay Heat Per Assembly	≤ 115 Watts
f. Original Fuel Assembly Length:	≤ 135.0 inches (nominal design)
g. Original Fuel Assembly Width:	≤ 4.70 inches (nominal design)
h. Fuel Debris Weight:	≤ ~~400~~ *550* lbs, including channels *and DFC*

Table 2.1-1 (page 17 of 33)
Fuel Assembly Limits

III. MPC MODEL: MPC-68F (continued)

A. Allowable Contents (continued)

7. Thoria rods (ThO_2 and UO_2) placed in Dresden Unit 1 Thoria Rod Canisters and meeting the following specifications:

a. Cladding Type:	Zircaloy (Zr)
b. Composition:	98.2 wt.% ThO_2, 1.8 wt. % UO_2 with an enrichment of 93.5 wt. % ^{235}U.
c. Number of Rods Per Thoria Rod Canister:	$\leq$ 18
d. Decay Heat Per Thoria Rod Canister:	$\leq$ 115 Watts
e. Post-irradiation Fuel Cooling Time and Average Burnup Per Thoria Rod Canister:	A fuel post-irradiation cooling time $\geq$ 18 years and an average burnup $\leq$ 16,000 MWD/MTIHM.
f. Initial Heavy Metal Weight:	$\leq$ 27 kg/canister
g. Fuel Cladding O.D.:	$\geq$ 0.412 inches
h. Fuel Cladding I.D.:	$\leq$ 0.362 inches
i. Fuel Pellet O.D.:	$\leq$ 0.358 inches
j. Active Fuel Length:	$\leq$ 111 inches
k. Canister Weight:	$\leq$ 550 lbs, including fuel

Table 2.1-1 (page ~~14~~ *18* of ~~14~~ *33*)
Fuel Assembly Limits

III. MPC MODEL: MPC-68F (continued)

B. Quantity per MPC (up to a total of 68 assemblies):
(All fuel assemblies must be array/class 6x6A, 6x6B, 6x6C, 7x7A, or 8x8A):

Up to four (4) DFCs containing uranium oxide BWR FUEL DEBRIS or MOX BWR FUEL DEBRIS. The remaining MPC-68F fuel storage locations may be filled with ~~array/class 6x6A, 6x6B, 6x6C, 7x7A, and 8x8A~~ fuel assemblies of the following type, as applicable:

1. Uranium oxide BWR INTACT FUEL ASSEMBLIES;
2. MOX BWR INTACT FUEL ASSEMBLIES;
3. Uranium oxide BWR DAMAGED FUEL ASSEMBLIES placed in DFCs; ~~or~~
4. MOX BWR DAMAGED FUEL ASSEMBLIES placed in DFCs; *or*
5. *Up to one (1) Dresden Unit 1 Thoria Rod Canister.*

C. Fuel assemblies with stainless steel channels are not authorized for loading in the MPC-68F.

D. Dresden Unit 1 fuel assemblies with one Antimony-Beryllium neutron source are authorized for loading in the MPC-68F. The Antimony-Beryllium source material shall be in a water rod location.

Table 2.1-1 (page 19 of 33)
Fuel Assembly Limits

IV. MPC MODEL: MPC-24E

A. Allowable Contents

1. Uranium oxide, PWR INTACT FUEL ASSEMBLIES listed in Table 2.1-2, with or without NON-FUEL HARDWARE and meeting the following specifications (Note 1):

a. Cladding Type:	*Zircaloy (Zr) or Stainless Steel (SS) as specified in Table 2.1-2 for the applicable fuel assembly array/class*
b. Initial Enrichment:	*As specified in Table 2.1-2 for the applicable fuel assembly array/class.*
c. Post-irradiation Cooling Time and Average Burnup Per Assembly:	
i. Array/Classes 14x14D, 14x14E, and 15x15G	*Cooling time ≥ 8 years and an average burnup ≤ 40,000 MWD/MTU.*
ii. All Other Array/Classes	*As specified in Tables 2.1-4 or 2.1-6.*
iii. NON-FUEL HARDWARE	*As specified in Table 2.1-8.*

Table 2.1-1 (page 20 of 33)
Fuel Assembly Limits

IV. MPC MODEL: MPC-24E (continued)

A. Allowable Contents (continued)

d. Decay Heat Per Assembly:	
i. Array/Classes 14x14D, 14x14E, and 15x15G	*≤ 710 Watts.*
ii. All other Array/Classes	*As specified in Tables 2.1-5 or 2.1-7.*
e. Fuel Assembly Length:	*≤ 176.8 inches (nominal design)*
f. Fuel Assembly Width:	*≤ 8.54 inches (nominal design)*
g. Fuel Assembly Weight:	*≤ 1,680 lbs (including NON-FUEL HARDWARE)*

Table 2.1-1 (page 21 of 33)
Fuel Assembly Limits

IV. MPC MODEL: MPC-24E (continued)

A. Allowable Contents (continued)

2. Uranium oxide, PWR DAMAGED FUEL ASSEMBLIES, with or without NON-FUEL HARDWARE, placed in DAMAGED FUEL CONTAINERS. Uranium oxide PWR DAMAGED FUEL ASSEMBLIES shall meet the criteria specified in Table 2.1-2 and meet the following specifications (Note 1):

a. Cladding Type:	*Zircaloy (Zr) or Stainless Steel (SS) as specified in Table 2.1-2 for the applicable fuel assembly array/class*
b. Initial Enrichment:	*$\leq$ 4.0 wt% ^{235}U.*
c. Post-irradiation Cooling Time and Average Burnup Per Assembly:	
i. Array/Classes 14x14D, 14x14E, and 15x15G	*Cooling time $\geq$ 8 years and an average burnup $\leq$ 40,000 MWD/MTU.*
ii. All Other Array/Classes	*As specified in Tables 2.1-4 or 2.1-6.*
iii. NON-FUEL HARDWARE	*As specified in Table 2.1-8.*

Table 2.1-1 (page 22 of 33)
Fuel Assembly Limits

IV. MPC MODEL: MPC-24E (continued)

A. Allowable Contents (continued)

d. Decay Heat Per Assembly	
i. Array/Classes 14x14D, 14x14E, and 15x15G	≤ 710 Watts.
ii. All Other Array/Classes	As specified in Tables 2.1-5 or 2.1-7.
e. Fuel Assembly Length	≤ 176.8 inches (nominal design)
f. Fuel Assembly Width	≤ 8.54 inches (nominal design)
g. Fuel Assembly Weight	≤ 1,680 lbs (including NON-FUEL HARDWARE and DFC)

B. Quantity per MPC: Up to four (4) DAMAGED FUEL ASSEMBLIES in DAMAGED FUEL CONTAINERS, stored in fuel storage locations 3, 6, 19 and/or 22. The remaining MPC-24E fuel storage locations may be filled with PWR INTACT FUEL ASSEMBLIES meeting the applicable specifications.

C. FUEL DEBRIS is not authorized for loading in the MPC-24E.

Note 1: Fuel assemblies containing BPRAs or TPDs may be stored in any fuel storage location. Fuel assemblies containing CRAs or APSRs must be loaded in fuel storage locations 9,10,15 and/or 16. These requirements are in addition to any other requirements specified for uniform or regionalized fuel loading.

Table 2.1-1 (page 23 of 33)
Fuel Assembly Limits

V. MPC MODEL: MPC-32

A. Allowable Contents

1. Uranium oxide, PWR INTACT FUEL ASSEMBLIES listed in Table 2.1-2, with or without NON-FUEL HARDWARE and meeting the following specifications (Note 1):

a. Cladding Type:	*Zircaloy (Zr) or Stainless Steel (SS) as specified in Table 2.1-2 for the applicable fuel assembly array/class*
b. Initial Enrichment:	*As specified in Table 2.1-2 for the applicable fuel assembly array/class.*
c. Post-irradiation Cooling Time and Average Burnup Per Assembly	
i. Array/Classes 14x14D, 14x14E, and 15x15G	*Cooling time ≥ 9 years and an average burnup ≤ 30,000 MWD/MTU or cooling time ≥ 20 years and an average burnup ≤ 40,000 MWD/MTU.*
ii. All Other Array/Classes	*As specified in Tables 2.1-4 or 2.1-6.*
iii. NON-FUEL HARDWARE	*As specified in Table 2.1-8.*

Table 2.1-1 (page 24 of 33)
Fuel Assembly Limits

V. MPC MODEL: MPC-32 (continued)

A. Allowable Contents (continued)

d. Decay Heat Per Assembly	
i. Array/Classes 14x14D, 14x14E, and 15x15G	*≤ 500 Watts*
ii. All Other Array/Classes	*As specified in Tables 2.1-5 or 2.1-7.*
e. Fuel Assembly Length	*≤ 176.8 inches (nominal design)*
f. Fuel Assembly Width	*≤ 8.54 inches (nominal design)*
g. Fuel Assembly Weight	*≤ 1,680 lbs (including NON-FUEL HARDWARE)*

B. Quantity per MPC: Up to 32 PWR INTACT FUEL ASSEMBLIES.

C. DAMAGED FUEL ASSEMBLIES and FUEL DEBRIS are not authorized for loading in the MPC-32.

Note 1: Fuel assemblies containing BPRAs or TPDs may be stored in any fuel storage location. Fuel assemblies containing CRAs or APSRs must be loaded in fuel storage locations 13, 14, 19, and/or 20. These requirements are in addition to any other requirements specified for uniform or regionalized fuel loading.

Table 2.1-1 (page 25 of 33)
Fuel Assembly Limits

VI. MPC MODEL: MPC-68FF

A. Allowable Contents

1. Uranium oxide or MOX BWR INTACT FUEL ASSEMBLIES listed in Table 2.1-2, with or without channels and meeting the following specifications:

a. Cladding Type:	*Zircaloy (Zr) or Stainless Steel (SS) as specified in Table 2.1-3 for the applicable fuel assembly array/class*
c. Maximum PLANAR-AVERAGE INITIAL ENRICHMENT:	*As specified in Table 2.1-3 for the applicable fuel assembly array/class.*
d. Initial Maximum Rod Enrichment	*As specified in Table 2.1-3 for the applicable fuel assembly array/class.*
e. Post-irradiation Cooling Time and Average Burnup Per Assembly	
i. Array/Classes 6x6A, 6x6C, 7x7A, and 8x8A	*Cooling time ≥ 18 years and an average burnup ≤ 30,000 MWD/MTU.*
ii. Array/Class 8x8F	*Cooling time ≥ 10 years and an average burnup ≤ 27,500 MWD/MTU.*
iii. Array/Classes 10x10D and 10x10E	*Cooling time ≥ 10 years and an average burnup ≤ 22,500 MWD/MTU.*
iv. All Other Array/Classes	*As specified in Tables 2.1-4 or 2.1-6.*

Table 2.1-1 (page 26 of 33)
Fuel Assembly Limits

VI. MPC MODEL: MPC-68FF (continued)

A. Allowable Contents (continued)

e. Decay Heat Per Assembly

i. Array/Classes 6x6A, 6x6C, 7x7A, and 8x8A	*≤ 115 Watts*
ii. Array/Class 8x8F	*≤ 183.5 Watts*
iii. Array/Classes 10x10D and 10x10E	*≤ 95 Watts*
iv. All Other Array/Classes	*As specified in Tables 2.1-5 or 2.1-7.*
f. Fuel Assembly Length	*≤ 176.5 inches (nominal design)*
g. Fuel Assembly Width	*≤ 5.85 inches (nominal design)*
h. Fuel Assembly Weight	*≤ 700 lbs, including channels*

Table 2.1-1 (page 27 of 33)
Fuel Assembly Limits

VI. MPC MODEL: MPC-68FF (continued)

A. Allowable Contents (continued)

2. Uranium oxide or MOX BWR DAMAGED FUEL ASSEMBLIES or FUEL DEBRIS, with or without channels, placed in DAMAGED FUEL CONTAINERS. Uranium oxide and MOX BWR DAMAGED FUEL ASSEMBLIES and FUEL DEBRIS shall meet the criteria specified in Table 2.1-3, and meet the following specifications:

a. Cladding Type:	*Zircaloy (Zr) or Stainless Steel (SS) in accordance with Table 2.1-3 for the applicable fuel assembly array/class.*
b. Maximum PLANAR-AVERAGE INITIAL ENRICHMENT:	
i. Array/Classes 6x6A, 6x6B, 6x6C, 7x7A, and 8x8A.	*As specified in Table 2.1-3 for the applicable fuel assembly array/class.*
ii. All Other Array Classes	*$\leq$ 4.0 wt.% ^{235}U.*
c. Initial Maximum Rod Enrichment	*As specified in Table 2.1-3 for the applicable fuel assembly array/class.*
d. Post-irradiation Cooling Time and Average Burnup Per Assembly:	
i. Array/Class 6x6A, 6x6B, 6x6C, 7x7A, or 8x8A	*Cooling time $\geq$ 18 years and an average burnup $\leq$ 30,000 MWD/MTU (or MWD/MTIHM).*
ii. Array/Class 8x8F	*Cooling time $\geq$ 10 years and an average burnup $\leq$ 27,500 MWD/MTU.*
iii. Array/Class 10x10D and 10x10E	*Cooling time $\geq$ 10 years and an average burnup $\leq$ 22,500 MWD/MTU.*
iv. All Other Array/Classes	*As specified in Tables 2.1-4 or 2.1-6.*

Table 2.1-1 (page 28 of 33)
Fuel Assembly Limits

VI. MPC MODEL: MPC-68FF (continued)

A. Allowable Contents (continued)

e. Decay Heat Per Assembly

i.	Array/Class 6x6A, 6x6B, 6x6C, 7x7A, or 8x8A	≤ 115 Watts
ii.	Array/Class 8x8F	≤ 183.5 Watts
iii.	Array/Classes 10x10D and 10x10E	≤ 95 Watts
iv.	All Other Array/Classes	As specified in Tables 2.1-5 or 2.1-7

f. Fuel Assembly Length

i.	Array/Class 6x6A, 6x6B, 6x6C, 7x7A, or 8x8A	≤ 135.0 inches (nominal design)
ii.	All Other Array/Classes	≤ 176.5 inches (nominal design)

g. Fuel Assembly Width

i.	Array/Class 6x6A, 6x6B, 6x6C, 7x7A, or 8x8A	≤ 4.70 inches (nominal design)
ii.	All Other Array/Classes	≤ 5.85 inches (nominal design)

h. Fuel Assembly Weight

i.	Array/Class 6x6A, 6x6B, 6x6C, 7x7A, or 8x8A	≤ 550 lbs, including channels and DFC
ii.	All Other Array/Classes	≤ 700 lbs, including channels and DFC

Table 2.1-1 (page 33 of 33)
Fuel Assembly limits

VI. MPC MODEL: MPC-68FF (continued)

B. Quantity per MPC (up to a total of 68 assemblies)

Up to sixteen (16) DFCs containing BWR DAMAGED FUEL ASSEMBLIES and/or up to eight (8) DFCs containing FUEL DEBRIS. DFCs shall be located only in fuel storage locations 1, 2, 3, 8, 9, 16, 25, 34, 35, 44, 53, 60, 61, 66, 67, and/or 68. The remaining MPC-68FF fuel storage locations may be filled with fuel assemblies of the following type:

3. Uranium Oxide BWR INTACT FUEL ASSEMBLIES; or

4. MOX BWR INTACT FUEL ASSEMBLIES;

C. Dresden Unit 1 fuel assemblies with one Antimony-Beryllium neutron source are authorized for loading in the MPC-68FF. The Antimony-Beryllium source material shall be in a water rod location.

D. Array/Class 10x10D and 10x10E fuel assemblies in stainless steel channels must be stored in fuel storage locations 19 - 22, 28 - 31, 38 -41, and/or 47 - 50.

Table 2.1-1 (page 30 of 33)
Fuel Assembly Limits

VII. MPC MODEL: MPC-24EF

A. Allowable Contents

1. Uranium oxide, PWR INTACT FUEL ASSEMBLIES listed in Table 2.1-2, with or without NON-FUEL HARDWARE and meeting the following specifications (Note 1):

a. Cladding Type:	Zircaloy (Zr) or Stainless Steel (SS) as specified in Table 2.1-2 for the applicable fuel assembly array/class
b. Initial Enrichment:	As specified in Table 2.1-2 for the applicable fuel assembly array/class.
c. Post-irradiation Cooling Time and Average Burnup Per Assembly:	
i. Array/Classes 14x14D, 14x14E, and 15x15G	Cooling time ≥ 8 years and an average burnup ≤ 40,000 MWD/MTU.
ii. All Other Array/Classes	As specified in Tables 2.1-4 or 2.1-6.
iii. NON-FUEL HARDWARE	As specified in Table 2.1-8.

Table 2.1-1 (page 31 of 33)
Fuel Assembly Limits

VII. MPC MODEL: MPC-24EF (continued)

A. Allowable Contents (continued)

d. Decay Heat Per Assembly:	
i. Array/Classes 14x14D, 14x14E, and 15x15G	≤ 710 Watts.
ii. All other Array/Classes	As specified in Tables 2.1-5 or 2.1-7.
e. Fuel Assembly Length:	≤ 176.8 inches (nominal design)
f. Fuel Assembly Width:	≤ 8.54 inches (nominal design)
g. Fuel Assembly Weight:	≤ 1,680 lbs (including NON-FUEL HARDWARE)

Table 2.1-1 (page 32 of 33)
Fuel Assembly Limits

VII. MPC MODEL: MPC-24EF (continued)

A. Allowable Contents (continued)

2. Uranium oxide, PWR DAMAGED FUEL ASSEMBLIES and FUEL DEBRIS, with or without NON-FUEL HARDWARE, placed in DAMAGED FUEL CONTAINERS. Uranium oxide PWR DAMAGED FUEL ASSEMBLIES and FUEL DEBRIS shall meet the criteria specified in Table 2.1-2 and meet the following specifications (Note 1):

a. Cladding Type:	*Zircaloy (Zr) or Stainless Steel (SS) as specified in Table 2.1-2 for the applicable fuel assembly array/class*
b. Initial Enrichment:	*≤ 4.0 wt% ^{235}U.*
c. Post-irradiation Cooling Time and Average Burnup Per Assembly:	
i. Array/Classes 14x14D, 14x14E, and 15x15G	*Cooling time ≥ 8 years and an average burnup ≤ 40,000 MWD/MTU.*
ii. All Other Array/Classes	*As specified in Tables 2.1-4 or 2.1-6.*
iii. NON-FUEL HARDWARE	*As specified in Table 2.1-8.*

Table 2.1-1 (page 33 of 33)
Fuel Assembly Limits

VII. MPC MODEL: MPC-24EF (continued)

A. Allowable Contents (continued)

d. Decay Heat Per Assembly	
i. Array/Classes 14x14D, 14x14E, and 15x15G	*≤ 710 Watts.*
ii. All Other Array/Classes	*As specified in Tables 2.1-5 or 2.1-7.*
e. Fuel Assembly Length	*≤ 176.8 inches (nominal design)*
f. Fuel Assembly Width	*≤ 8.54 inches (nominal design)*
g. Fuel Assembly Weight	*≤ 1,680 lbs (including NON-FUEL HARDWARE and DFC)*

B. Quantity per MPC: Up to four (4) DAMAGED FUEL ASSEMBLIES and/or FUEL DEBRIS in DAMAGED FUEL CONTAINERS, stored in fuel storage locations 3, 6, 19 and/or 22. The remaining MPC-24E fuel storage locations may be filled with PWR INTACT FUEL ASSEMBLIES meeting the applicable specifications.

Note 1: Fuel assemblies containing BPRAs or TPDs may be stored in any fuel storage location. Fuel assemblies containing CRAs or APSRs must be loaded in fuel storage locations 9,10,15 and/or 16. These requirements are in addition to any other requirements specified for uniform or regionalized fuel loading.

Table 2.1-2 (page 1 of 4)
PWR FUEL ASSEMBLY CHARACTERISTICS (Note 1)

Fuel Assembly Array/Class	**14x14A**	**14x14B**	**14x14C**	**14x14D**	***14x14E***
Clad Material (Note 2)	Zr	Zr	Zr	SS	*SS*
Design Initial U (kg/assy.) (Note 3)	~~≤ 402~~ *≤ 407*	~~≤ 402~~ *≤ 407*	~~≤ 410~~ *≤ 425*	≤ 400	*≤ 206*
Initial Enrichment *(MPC-24, 24E and 24EF without soluble boron credit)* (wt % ^{235}U) *(Note 7)*	≤ 4.6 *(24)* *≤ 5.0 (24E/24EF)*	≤ 4.6 *(24)* *≤ 5.0 (24E/24EF)*	≤ 4.6 *(24)* *≤ 5.0 (24E/24EF)*	≤ 4.0 *(24)* *≤ 5.0 (24E/24EF)*	*≤ 5.0 (24)* *≤ 5.0 (24E/24EF)*
Initial Enrichment (MPC-24, 24E, 24EF, or 32 with soluble boron credit - see Notes 5 and 7) (wt % ^{235}U)	*≤ 5.0*	*≤ 5.0*	*≤ 5.0*	*≤ 5.0*	*≤ 5.0*
No. of Fuel Rod *Locations*	179	179	176	180	*173*
Fuel Rod Clad O.D. (in.)	≥ 0.400	≥ 0.417	≥ 0.440	≥ 0.422	*≥ 0.3415*
Fuel Rod Clad I.D. (in.)	≤ 0.3514	≤ 0.3734	~~≤ 0.3840~~ *≤ 0.3880*	≤ 0.3890	*≤ 0.3175*
Fuel Pellet Dia. (in.)	≤ 0.3444	≤ 0.3659	~~≤ 0.3770~~ *≤ 0.3805*	≤ 0.3835	*≤ 0.3130*
Fuel Rod Pitch (in.)	≤ 0.556	≤ 0.556	≤ 0.580	≤ 0.556	*Note 6*
Active Fuel Length (in.)	≤ 150	≤ 150	≤ 150	≤ 144	*≤ 102*
No. of Guide *and/or Instrument* Tubes	17	17	5 (Note 4)	16	*0*
Guide/*Instrument* Tube Thickness (in.)	≥ 0.017	≥ 0.017	~~≥ 0.040~~ *≥ 0.038*	≥ 0.0145	*N/A*

Table 2.1-2 (page 2 of 4)
PWR FUEL ASSEMBLY CHARACTERISTICS (Note 1)

Fuel Assembly Array/Class	**15x15A**	**15x15B**	**15x15C**	**15x15D**	**15x15E**	**15x15F**
Clad Material (Note 2)	Zr	Zr	Zr	Zr	Zr	Zr
Design Initial U (kg/assy.) (Note 3)	~~≤ 420~~ *≤ 464*	≤ 464	≤ 464	≤ 475	≤ 475	≤ 475
Initial Enrichment *(MPC-24, 24E and 24EF without soluble boron credit)* (wt % ^{235}U) *(Note 7)*	≤ 4.1 *(24)* *≤ 4.5 (24E/24EF)*	≤ 4.1 *(24)* *≤ 4.5 (24E/24EF)*	≤ 4.1 *(24)* *≤ 4.5 (24E/24EF)*	≤ 4.1 *(24)* *≤ 4.5 (24E/24EF)*	≤ 4.1 *(24)* *≤ 4.5 (24E/24EF)*	≤ 4.1 *(24)* *≤ 4.5 (24E/24EF)*
Initial Enrichment (MPC-24, 24E, 24EF, or 32 with soluble boron credit - see Notes 5 and 7) (wt % ^{235}U)	*≤ 5.0*	*≤ 5.0*	*≤ 5.0*	*≤ 5.0*	*≤ 5.0*	*≤ 5.0*
No. of Fuel Rod *Locations*	204	204	204	208	208	208
Fuel Rod Clad O.D. (in.)	≥ 0.418	≥ 0.420	≥ 0.417	≥ 0.430	≥ 0.428	≥ 0.428
Fuel Rod Clad I.D. (in.)	≤ 0.3660	≤ 0.3736	≤ 0.3640	≤ 0.3800	≤ 0.3790	≤ 0.3820
Fuel Pellet Dia. (in.)	≤ 0.3580	≤ 0.3671	≤ 0.3570	≤ 0.3735	≤ 0.3707	≤ 0.3742
Fuel Rod Pitch (in.)	≤ 0.550	≤ 0.563	≤ 0.563	≤ 0.568	≤ 0.568	≤ 0.568
Active Fuel Length (in.)	≤ 150	≤ 150	≤ 150	≤ 150	≤ 150	≤ 150
No. of Guide *and/or Instrument* Tubes	21	21	21	17	17	17
Guide/*Instrument* Tube Thickness (in.)	≥ 0.0165	≥ 0.015	≥ 0.0165	≥ 0.0150	≥ 0.0140	≥ 0.0140

Table 2.1-2 (page 3 of 4)
PWR FUEL ASSEMBLY CHARACTERISTICS (Note 1)

Fuel Assembly Array/ Class	**15x15G**	***15x15H***	**16x16A**	**17x17A**	**17x17B**	**17x17C**
Clad Material (Note 2)	SS	*Zr*	Zr	Zr	Zr	Zr
Design Initial U (kg/assy.) (Note 3)	≤ 420	*≤ 475*	~~≤ 430~~ *≤ 443*	~~≤ 450~~ *≤ 467*	~~≤ 464~~ *≤ 467*	~~≤ 460~~ *≤ 474*
Initial Enrichment *(MPC-24 24E, and 24EF without soluble boron credit)* (wt % ^{235}U) *(Note 7)*	≤ 4.0 *(24)* *≤ 4.5 (24E/24EF)*	*≤ 3.8 (24)* *≤ 4.2 (24E/24EF)*	≤ 4.6 *(24)* *≤ 5.0 (24E/24EF)*	≤ 4.0 *(24)* *≤ 4.4 (24E/24EF)*	≤ 4.0 *(24)* *≤ 4.4 (24E/24EF)*	≤ 4.0 *(24)* *≤ 4.4 (24E/24EF)*
Initial Enrichment (MPC-24, 24E, 24EF, or 32 with soluble boron credit - see Notes 5 and 7) (wt % ^{235}U)	*≤ 5.0*	*≤ 5.0*	*≤ 5.0*	*≤ 5.0*	*≤ 5.0*	*≤ 5.0*
No. of Fuel Rod *Locations*	204	*208*	236	264	264	264
Fuel Rod Clad O.D. (in.)	≥ 0.422	*≥ 0.414*	≥ 0.382	≥ 0.360	≥ 0.372	≥ 0.377
Fuel Rod Clad I.D. (in.)	≤ 0.3890	*≤ 0.3700*	≤ 0.3320	≤ 0.3150	≤ 0.3310	≤ 0.3330
Fuel Pellet Dia. (in.)	≤ 0.3825	*≤ 0.3622*	≤ 0.3255	≤ 0.3088	≤ 0.3232	≤ 0.3252
Fuel Rod Pitch (in.)	≤ 0.563	*≤ 0.568*	≤ 0.506	≤ 0.496	≤ 0.496	≤ 0.502
Active Fuel Length (in.)	≤ 144	*≤ 150*	≤ 150	≤ 150	≤ 150	≤ 150
No. of Guide *and/or Instrument* Tubes	21	*17*	5 (Note 4)	25	25	25
Guide/*Instrument* Tube Thickness (in.)	≥ 0.0145	*≥ 0.0140*	≥ 0.0400	≥ 0.016	≥ 0.014	≥ 0.020

Table 2.1-2 (page 4 of 4)
PWR FUEL ASSEMBLY CHARACTERISTICS

Notes:

1. All dimensions are design nominal values. Maximum and minimum dimensions are specified to bound variations in design nominal values among fuel assemblies within a given array/class.

2. Zr designates cladding material made of zirconium or zirconium alloys.

3. Design initial uranium weight is the nominal uranium weight specified for each assembly by the fuel manufacturer or reactor user. For each PWR fuel assembly, the total ~~initial~~ uranium weight *limit specified in this table* may be *increased* up to 2.0 percent ~~higher than the design initial uranium weight due~~ *for comparison with users' fuel records* to *account for* manufacturer's tolerances.

4. Each guide tube replaces four fuel rods.

5. *Soluble boron concentration per LCO 3.3.1.*

6. *This fuel assembly array/class includes only the Indian Point Unit 1 fuel assembly. This fuel assembly has two pitches in different sectors of the assembly.*

7. *For those MPCs loaded with both INTACT FUEL ASSEMBLIES and DAMAGED FUEL ASSEMBLIES or FUEL DEBRIS, the maximum initial enrichment of the INTACT FUEL ASSEMBLIES is limited to the maximum initial enrichment of the DAMAGED FUEL ASSEMBLIES and FUEL DEBRIS (i.e., 4.0 wt.% ^{235}U).*

Table 2.1-3 (page 1 of 5)
BWR FUEL ASSEMBLY CHARACTERISTICS (Note 1)

Fuel Assembly Array/Class	**6x6A**	**6x6B**	**6x6C**	**7x7A**	**7x7B**	**8x8A**
Clad Material (Note 2)	Zr	Zr	Zr	Zr	Zr	Zr
Design Initial U (kg/assy.) (Note 3)	~~≤ 108~~ *110*	~~≤ 108~~ *110*	~~≤ 108~~ *110*	≤ 100	≤ 195	≤ 120
Maximum PLANAR-AVERAGE INITIAL ENRICHMENT (wt.% ^{235}U) *(Note 14)*	≤ 2.7	≤ 2.7 for the UO_2 rods. See Note 4 for MOX rods	≤ 2.7	≤ 2.7	≤ 4.2	≤ 2.7
Initial Maximum Rod Enrichment (wt.% ^{235}U)	≤ 4.0	≤ 4.0	≤ 4.0	~~≤ 4.0~~ *≤ 5.5*	≤ 5.0	≤ 4.0
No. of Fuel Rod *Locations*	*35 or* 36	*35 or* 36 (up to 9 MOX rods)	36	49	49	*63 or* 64
Fuel Rod Clad O.D. (in.)	≥ 0.5550	≥ 0.5625	≥ 0.5630	≥ 0.4860	≥ 0.5630	≥ 0.4120
Fuel Rod Clad I.D. (in.)	~~≤ 0.4945~~ *≤ 0.5105*	≤ 0.4945	≤ 0.4990	~~≤ 0.4200~~ *≤ 0.4204*	≤ 0.4990	≤ 0.3620
Fuel Pellet Dia. (in.)	~~≤ 0.4940~~ *≤ 0.4980*	≤ 0.4820	≤ 0.4880	≤ 0.4110	~~≤ 0.4880~~ *≤ 0.4910*	≤ 0.3580
Fuel Rod Pitch (in.)	~~≤ 0.694~~ *≤ 0.710*	~~≤ 0.694~~ *≤ 0.710*	≤ 0.740	≤ 0.631	≤ 0.738	≤ 0.523
Active Fuel Length (in.)	~~≤ 110~~ *≤ 120*	~~≤ 110~~ *≤ 120*	≤ 77.5	~~≤ 79~~ *≤ 80*	≤ 150	~~≤ 110~~ *≤ 120*
No. of Water Rods *(Note 11)*	*1 or* 0	*1 or* 0	0	0	0	*1 or* 0
Water Rod Thickness (in.)	~~N/A~~ *> 0*	~~N/A~~ *> 0*	N/A	N/A	N/A	~~N/A~~ *≥ 0*
Channel Thickness (in.)	≤ 0.060	≤ 0.060	≤ 0.060	≤ 0.060	≤ 0.120	≤ 0.100

Table 2.1-3 (2 of 5)
BWR FUEL ASSEMBLY CHARACTERISTICS (Note 1)

Fuel Assembly Array/Class	**8x8B**	**8x8C**	**8x8D**	**8x8E**	***8x8F***	**9x9A**
Clad Material (Note 2)	Zr	Zr	Zr	Zr	*Zr*	Zr
Design Initial U (kg/assy.) (Note 3)	~~≤ 185~~ ≤ *191*	~~≤ 185~~ ≤ *191*	~~≤ 185~~ ≤ *191*	~~≤ 180~~ ≤ *191*	≤ *191*	~~≤ 173~~ ≤ *179*
Maximum PLANAR-AVERAGE INITIAL ENRICHMENT (wt.% ^{235}U) *(Note 14)*	≤ 4.2	≤ 4.2	≤ 4.2	≤ 4.2	≤ *4.0*	≤ 4.2
Initial Maximum Rod Enrichment (wt.% ^{235}U)	≤ 5.0	≤ 5.0	≤ 5.0	≤ 5.0	≤ *5.0*	≤ 5.0
No. of Fuel Rod *Locations*	63 *or 64*	62	60 *or 61*	59	*64*	74/66 (Note 5)
Fuel Rod Clad O.D. (in.)	≥ 0.4840	≥ 0.4830	≥ 0.4830	≥ 0.4930	≥ *0.4576*	≥ 0.4400
Fuel Rod Clad I.D. (in.)	~~≤ 0.4250~~ ≤ *0.4295*	≤ 0.4250	~~≤ 0.4190~~ ≤ *0.4230*	≤ 0.4250	≤ *0.3996*	≤ 0.3840
Fuel Pellet Dia. (in.)	~~≤ 0.4160~~ ≤ *0.4195*	≤ 0.4160	~~≤ 0.4110~~ ≤ *0.4140*	≤ 0.4160	≤ *0.3913*	≤ 0.3760
Fuel Rod Pitch (in.)	~~≤ 0.641~~ ≤ *0.642*	≤ 0.641	≤ 0.640	≤ 0.640	≤ *0.609*	≤ 0.566
Design Active Fuel Length (in.)	≤ 150	≤ 150	≤ 150	≤ 150	≤ *150*	≤ 150
No. of Water Rods *(Note 11)*	1 *or 0*	2	1 - 4 (Note ~~6~~ *7*)	5	*N/A* *(Note 12)*	2
Water Rod Thickness (in.)	≥ 0.034	> 0.00	> 0.00	≥ 0.034	≥ *0.0315*	> 0.00
Channel Thickness (in.)	≤ 0.120	≤ 0.120	≤ 0.120	≤ 0.100	≤ *0.055*	≤ 0.120

Table 2.1-3 (page 3 of 5)
BWR FUEL ASSEMBLY CHARACTERISTICS (Note 1)

Fuel Assembly Array/Class	**9x9B**	**9x9C**	**9x9D**	**9x9E *(Note 13)***	**9x9F *(Note 13)***	***9x9G***
Clad Material (Note 2)	Zr	Zr	Zr	Zr	Zr	*Zr*
Design Initial U (kg/assy.) (Note 3)	~~≤ 173~~ *≤ 179*	~~≤ 173~~ *≤ 179*	~~≤ 170~~ *≤ 179*	~~≤ 170~~ *≤ 179*	~~≤ 170~~ *≤ 179*	*≤ 179*
Maximum PLANAR-AVERAGE INITIAL ENRICHMENT (wt.% ^{235}U) *(Note 14)*	≤ 4.2	≤ 4.2	≤ 4.2	~~≤ 4.2~~ *≤ 4.0*	~~≤ 4.2~~ *≤ 4.0*	*≤ 4.2*
Initial Maximum Rod Enrichment (wt.% ^{235}U)	≤ 5.0	≤ 5.0	≤ 5.0	≤ 5.0	≤ 5.0	*≤ 5.0*
No. of Fuel Rod *Locations*	72	80	79	76	76	*72*
Fuel Rod Clad O.D. (in.)	≥ 0.4330	≥ 0.4230	≥ 0.4240	≥ 0.4170	≥ 0.4430	*≥ 0.4240*
Fuel Rod Clad I.D. (in.)	≤ 0.3810	≤ 0.3640	≤ 0.3640	~~≤ 0.3590~~ *≤ 0.3640*	~~≤ 0.3810~~ *≤ 0.3860*	*≤ 0.3640*
Fuel Pellet Dia. (in.)	≤ 0.3740	≤ 0.3565	≤ 0.3565	~~≤ 0.3525~~ *≤ 0.3530*	≤ 0.3745	*≤ 0.3565*
Fuel Rod Pitch (in.)	~~≤ 0.569~~ *≤ 0.572*	≤ 0.572	≤ 0.572	≤ 0.572	≤ 0.572	*≤ 0.572*
Design Active Fuel Length (in.)	≤ 150	≤ 150	≤ 150	≤ 150	≤ 150	*≤ 150*
No. of Water Rods *(Note 11)*	1 (Note ~~7~~ *6*)	1	2	5	5	*1 (Note 6)*
Water Rod Thickness (in.)	> 0.00	≥ 0.020	~~≥ 0.0305~~ *≥ 0.0300*	~~≥ 0.0305~~ *≥ 0.0120*	~~≥ 0.0305~~ *≥ 0.0120*	*≥ 0.0320*
Channel Thickness (in.)	≤ 0.120	≤ 0.100	≤ 0.100	~~≤ 0.100~~ *≤ 0.120*	~~≤ 0.100~~ *≤ 0.120*	*≤ 0.120*

Table 2.1-3 (page 4 of 5)
BWR FUEL ASSEMBLY CHARACTERISTICS (Note 1)

Fuel Assembly Array/Class	**10x10A**	**10x10B**	**10x10C**	**10x10D**	**10x10E**
Clad Material (Note 2)	Zr	Zr	Zr	SS	SS
Design Initial U (kg/assy.) (Note 3)	~~≤ 182~~ *≤ 188*	~~≤ 182~~ *≤ 188*	~~≤ 180~~ *≤ 188*	≤ 125	≤ 125
Maximum PLANAR-AVERAGE INITIAL ENRICHMENT (wt.% ^{235}U) *(Note 14)*	≤ 4.2	≤ 4.2	≤ 4.2	≤ 4.0	≤ 4.0
Initial Maximum Rod Enrichment (wt.% ^{235}U)	≤ 5.0	≤ 5.0	≤ 5.0	≤ 5.0	≤ 5.0
No. of Fuel Rod *Locations*	92/78 (Note 8)	91/83 (Note 9)	96	100	96
Fuel Rod Clad O.D. (in.)	≥ 0.4040	≥ 0.3957	~~≥ 0.3790~~ *≥ 0.3780*	≥ 0.3960	≥ 0.3940
Fuel Rod Clad I.D. (in.)	≤ 0.3520	≤ 0.3480	≤ 0.3294	≤ 0.3560	≤ 0.3500
Fuel Pellet Dia. (in.)	≤ 0.3455	≤ 0.3420	≤ 0.3224	≤ 0.3500	≤ 0.3430
Fuel Rod Pitch (in.)	≤ 0.510	≤ 0.510	≤ 0.488	≤ 0.565	≤ 0.557
Design Active Fuel Length (in.)	≤ 150	≤ 150	≤ 150	≤ 83	≤ 83
No. of Water Rods *(Note 11)*	2	1 (Note ~~7~~ *6*)	5 (Note 10)	0	4
Water Rod Thickness (in.)	≥ 0.0300	> 0.00	~~≥ 0.034~~ *≥ 0.031*	N/A	≥ 0.022
Channel Thickness (in.)	≤ 0.120	≤ 0.120	≤ 0.055	≤ 0.080	≤ 0.080

Table 2.1-3 (page 5 of 5)
BWR FUEL ASSEMBLY CHARACTERISTICS

Notes:

1. All dimensions are design nominal values. Maximum and minimum dimensions are specified to bound variations in design nominal values among fuel assemblies within a given array/class.

2. Zr designates cladding material made of zirconium or zirconium alloys.

3. Design initial uranium weight is the nominal uranium weight specified for each assembly by the fuel manufacturer or reactor user. For each BWR fuel assembly, the total ~~initial~~ uranium weight *limit specified in this table* may be *increased* up to 1.5 percent ~~higher than the design initial uranium weight due~~ *for comparison with users' fuel records* to *account for* manufacturer tolerances.

4. $\leq$ ~~0.612~~ *0.635* wt. % ^{235}U and $\leq$ 1.578 wt. % total fissile plutonium (^{239}Pu and ^{241}Pu), *(wt. % of total fuel weight, i.e., UO_2 plus PuO_2)*.

5. This assembly class contains 74 total rods; 66 full length rods and 8 partial length rods.

6. *Square, replacing nine fuel rods.*

7. *Variable.*

8. This assembly contains 92 total fuel rods; 78 full length rods and 14 partial length rods.

9. This assembly class contains 91 total fuel rods; 83 full length rods and 8 partial length rods.

10. One diamond-shaped water rod replacing the four center fuel rods and four rectangular water rods dividing the assembly into four quadrants.

11. *These rods may also be sealed at both ends and contain Zr material in lieu of water.*

12. *This assembly is known as "QUAD+." It has four rectangular water cross segments dividing the assembly into four quadrants.*

13. *For the SPC 9x9-5 fuel assembly, each fuel rod must meet either the 9x9E or the 9x9F set of limits for clad O.D., clad I.D., and pellet diameter.*

14. *For those MPCs loaded with both INTACT FUEL ASSEMBLIES and DAMAGED FUEL ASSEMBLIES or FUEL DEBRIS, the maximum PLANAR AVERAGE INITIAL ENRICHMENT for the INTACT FUEL ASSEMBLIES is limited to 3.7 wt.% ^{235}U, as applicable.*

Table 2.1-4

FUEL ASSEMBLY COOLING AND *MAXIMUM* AVERAGE BURNUP ~~(Note 1)~~
(UNIFORM FUEL LOADING)

Post-irradiation Cooling Time (years)	MPC-24 PWR Assembly Burnup *(INTACT FUEL ASSEMBLIES)* (MWD/MTU)	*MPC-24E/24EF PWR Assembly Burnup (INTACT FUEL ASSEMBLIES) (MWD/MTU)*	*MPC-24E/24EF PWR Assembly Burnup (DAMAGED FUEL ASSEMBLIES AND FUEL DEBRIS) (MWD/MTU)*	*MPC-32 PWR Assembly Burnup (INTACT FUEL ASSEMBLIES (MWD/MTU)*	MPC-68/68FF BWR Assembly Burnup *(INTACT FUEL ASSEMBLIES) (MWD/MTU)*	*MPC-68/68FF BWR Assembly Burnup (DAMAGED FUEL ASSEMBLIES AND FUEL DEBRIS) (MWD/MTU)*
≥ 5	*40,600*	*41,100*	*39,200*	*32,200*	*38,300*	*33,400*
≥ 6	*45,000*	*45,000*	*43,700*	*36,500*	*41,600*	*36,600*
≥ 7	*45,900*	*46,300*	*45,200*	*37,500*	*42,300*	*37,000*
≥ 8	*48,300*	*48,900*	*47,300*	*39,900*	*44,800*	*39,100*
≥ 9	*50,300*	*50,700*	*49,000*	*41,500*	*46,600*	*40,700*
≥ 10	*51,600*	*52,100*	*50,100*	*42,900*	*48,000*	*41,900*
≥ 11	*53,100*	*53,700*	*51,500*	*44,100*	*49,600*	*43,000*
≥ 12	*54,500*	*55,100*	*52,600*	*45,000*	*50,800*	*44,100*
≥ 13	*55,600*	*56,100*	*53,800*	*45,700*	*51,800*	*45,000*
≥ 14	*56,500*	*57,100*	*54,900*	*46,500*	*52,700*	*45,800*
≥ 15	*57,400*	*58,000*	*55,800*	*47,200*	*53,900*	*46,500*

Note: 1. Linear interpolation between points is permitted.

Table 2.1-5

FUEL ASSEMBLY COOLING AND *MAXIMUM* DECAY HEAT ~~(Note 1)~~
(UNIFORM FUEL LOADING)

Post-irradiation Cooling Time (years)	MPC-24 PWR Assembly Decay Heat *(INTACT FUEL ASSEMBLIES)* (Watts)	*MPC-24E/24EF PWR Assembly Decay Heat (INTACT FUEL ASSEMBLIES) (Watts)*	*MPC-24E/24EF PWR Assembly Decay Heat (DAMAGED FUEL ASSEMBLIES AND FUEL DEBRIS) (Watts)*	*MPC-32 PWR Assembly Decay Heat (INTACT FUEL ASSEMBLIES (Watts)*	MPC-68/68FF BWR Assembly Decay Heat *(INTACT FUEL ASSEMBLIES) (Watts)*	*MPC-68/68FF BWR Assembly Decay Heat (DAMAGED FUEL ASSEMBLIES AND FUEL DEBRIS) (Watts)*
≥ 5	*1157*	*1173*	*1115*	*898*	*414*	*356*
≥ 6	*1123*	*1138*	*1081*	*873*	*394*	*337*
≥ 7	*1030*	*1043*	*1009*	*805*	*363*	*308*
≥ 8	*1020*	*1033*	*993*	*800*	*360*	*305*
≥ 9	*1010*	*1023*	*977*	*794*	*358*	*303*
≥ 10	*1000*	*1012*	*962*	*789*	*355*	*300*
≥ 11	*996*	*1008*	*958*	*785*	*353*	*299*
≥ 12	*992*	*1004*	*954*	*782*	*352*	*297*
≥ 13	*987*	*999*	*949*	*773*	*350*	*296*
≥ 14	*983*	*995*	*945*	*769*	*348*	*294*
≥ 15	*979*	*991*	*941*	*766*	*347*	*293*

Notes:
1. Linear interpolation between points is permitted.
2. *Includes all sources of heat (i.e., fuel and NON-FUEL HARDWARE).*

Table 2.1-6 (page 1 of 2)

FUEL ASSEMBLY COOLING AND MAXIMUM AVERAGE BURNUP (REGIONALIZED FUEL LOADING)

Post-irradiation Cooling Time (years)	MPC-24 PWR Assembly Burnup for Region 1 (MWD/MTU)	MPC-24 PWR Assembly Burnup for Region 2 (MWD/MTU)	MPC-24E/24EF PWR Assembly Burnup for Region 1 (MWD/MTU)	MPC-24E/24EF PWR Assembly Burnup for Region 2 (MWD/MTU)
≥ 5	49,800	32,200	51,600	32,200
≥ 6	56,100	37,400	58,400	37,400
≥ 7	56,400	41,100	58,500	41,100
≥ 8	58,800	43,800	60,900	43,800
≥ 9	60,400	45,800	62,300	45,800
≥ 10	61,200	47,500	63,300	47,500
≥ 11	62,400	49,000	64,900	49,000
≥ 12	63,700	50,400	65,900	50,400
≥ 13	64,800	51,500	66,800	51,500
≥ 14	65,500	52,500	67,500	52,500
≥ 15	66,200	53,700	68,200	53,700
≥ 16	-	55,000	-	55,000
≥ 17	-	55,900	-	55,900
≥ 18	-	56,800	-	56,800
≥ 19	-	57,800	-	57,800
≥ 20	-	58,800	-	58,800

Note: 1. Linear interpolation between points is permitted.

2. *These limits apply to INTACT FUEL ASSEMBLIES, DAMAGED FUEL ASSEMBLIES, and FUEL DEBRIS.*

Table 2.1-6 (page 2 of 2)

FUEL ASSEMBLY COOLING AND MAXIMUM AVERAGE BURNUP
(REGIONALIZED FUEL LOADING)

Post-irradiation Cooling Time (years)	MPC-32 PWR Assembly Burnup for Region 1 (MWD/MTU)	MPC-32 PWR Assembly Burnup for Region 2 (MWD/MTU)	MPC-68/68FF BWR Assembly Burnup for Region 1 (MWD/MTU)	MPC-68/68FF BWR Assembly Burnup for Region 2 (MWD/MTU)
≥ 5	39,800	22,100	45,100	26,200
≥ 6	43,400	26,200	47,400	30,500
≥ 7	44,500	29,100	47,400	33,600
≥ 8	46,700	31,200	50,400	35,900
≥ 9	48,400	32,700	52,100	37,600
≥ 10	49,600	34,100	53,900	39,000
≥ 11	50,900	35,200	55,500	40,200
≥ 12	51,900	36,200	56,500	41,200
≥ 13	52,900	37,000	57,500	42,300
≥ 14	53,800	37,800	58,800	43,300
≥ 15	54,700	38,600	59,900	44,200
≥ 16	-	39,400	-	45,000
≥ 17	-	40,200	-	45,900
≥ 18	-	40,800	-	46,700
≥ 19	-	41,500	-	47,500
≥ 20	-	42,200	-	48,500

Note:
1. Linear interpolation between points is permitted.
2. *These limits apply to INTACT FUEL ASSEMBLIES, DAMAGED FUEL ASSEMBLIES, and FUEL DEBRIS.*

Table 2.1-7 (page 1 of 2)

FUEL ASSEMBLY COOLING AND MAXIMUM DECAY HEAT
(REGIONALIZED FUEL LOADING)

Post-irradiation Cooling Time (years)	**MPC-24 PWR Assembly Decay Heat for Region 1 (Watts)**	**MPC-24 PWR Assembly Decay Heat for Region 2 (Watts)**	**MPC-24E/24EF PWR Assembly Decay Heat for Region 1 (Watts)**	**MPC-24E/24EF PWR Assembly Decay Heat for Region 2 (Watts)**
≥ 5	1470	900	1540	900
≥ 6	1470	900	1540	900
≥ 7	1335	900	1395	900
≥ 8	1,301	900	1360	900
≥ 9	1268	900	1325	900
≥ 10	1235	900	1290	900
≥ 11	1221	900	1275	900
≥ 12	1207	900	1260	900
≥ 13	1193	900	1245	900
≥ 14	1179	900	1230	900
≥ 15	1165	900	1215	900
≥ 16	-	900	-	900
≥ 17	-	900	-	900
≥ 18	-	900	-	900
≥ 19	-	900	-	900
≥ 20	-	900	-	900

Notes:
1. Linear interpolation between points is permitted.
2. *Includes all sources of decay heat (i.e., fuel and NON-FUEL HARDWARE).*
3. *These limits apply to INTACT FUEL ASSEMBLIES, DAMAGED FUEL ASSEMBLIES, and FUEL DEBRIS.*

Table 2.1-7 (page 2 of 2)

FUEL ASSEMBLY COOLING AND MAXIMUM DECAY HEAT (REGIONALIZED FUEL LOADING)

Post-irradiation Cooling Time (years)	**MPC-32 PWR Assembly Decay Heat for Region 1 (Watts)**	**MPC-32 PWR Assembly Decay Heat for Region 2 (Watts)**	**MPC-68/68FF BWR Assembly Decay Heat for Region 1 (Watts)**	**MPC-68/68FF BWR Assembly Decay Heat for Region 2 (Watts)**
≥ 5	1131	600	500	275
≥ 6	1072	600	468	275
≥ 7	993	600	418	275
≥ 8	978	600	414	275
≥ 9	964	600	410	275
≥ 10	950	600	405	275
≥ 11	943	600	403	275
≥ 12	937	600	400	275
≥ 13	931	600	397	275
≥ 14	924	600	394	275
≥ 15	918	600	391	275
≥ 16	-	600	-	275
≥ 17	-	600	-	275
≥ 18	-	600	-	275
≥ 19	-	600	-	275
≥ 20	-	600	-	275

Notes: 1. Linear interpolation between points is permitted.

2. *Includes all sources of decay heat (i.e., fuel and NON-FUEL HARDWARE).*

3. *These limits apply to INTACT FUEL ASSEMBLIES, DAMAGED FUEL ASSEMBLIES, and FUEL DEBRIS.*

Table 2.1-8
NON-FUEL HARDWARE COOLING AND AVERAGE BURNUP

Post-irradiation Cooling Time (years)	BPRA BURNUP (MWD/MTU)	TPD BURNUP (MWD/MTU)	CRA BURNUP (MWD/MTU)	APSR BURNUP (MWD/MTU)
≥ 3	≤ 20,000	NA (Note 3)	NA	NA
≥ 4	≤ 25000	≤ 20,000	NA	NA
≥ 5	≤ 30,000	≤ 25,000	≤ 630,000	≤ 45,000
≥ 6	≤ 40,000	≤ 30,000	-	≤ 54,500
≥ 7	≤ 45,000	≤ 40,000	-	≤ 68,000
≥ 8	≤ 50,000	≤ 45,000	-	≤ 83,000
≥ 9	≤ 60,000	≤ 50,000	-	≤ 111,000
≥ 10	-	≤ 60,000	-	≤ 180,000
≥ 11	-	≤ 75,000	-	≤ 630,000
≥ 12	-	≤ 90,000	-	-
≥ 13	-	≤ 180,000	-	-
≥ 14	-	≤ 630,000	-	-

Notes:

1. *Linear interpolation between points is permitted, except that TPD and APSR burnups > 180,000 MWD/MTU and ≤ 630,000 MWD/MTU must be cooled ≥ 14 years and ≥ 11 years, respectively.*
2. *Applicable to uniform loading and regionalized loading.*
3. *NA means not authorized for loading.*

3.0 DESIGN FEATURES

3.1 Site

3.1.1 Site Location

The HI-STORM 100 Cask System is authorized for general use by 10 CFR Part 50 license holders at various site locations under the provisions of 10 CFR 72, Subpart K.

3.2 Design Features Important for Criticality Control

3.2.1 MPC-24

1. Flux trap size: ≥ 1.09 in.

2. ^{10}B loading in the Boral neutron absorbers: ≥ 0.0267 g/cm^2

3.2.2 MPC-68 *and MPC-68FF*

1. Fuel cell pitch: ≥ 6.43 in.

2. ^{10}B loading in the Boral neutron absorbers: ≥ 0.0372 g/cm^2

3.2.3 MPC-68F

1. Fuel cell pitch: ≥ 6.43 in.

2. ^{10}B loading in the Boral neutron absorbers: ≥ 0.01 g/cm^2

3.2.4 MPC-24E and MPC-24EF

1. *Flux trap size:*

   i. Cells 3, 6, 19, and 22: ≥ 0.776 inch

   ii. All Other Cells: ≥ 1.076 inches

2. *^{10}B loading in the Boral neutron absorbers: ≥ 0.0372 g/cm^2*

3.2.5 MPC-32

1. *Fuel cell pitch: ≥ 9.158 inches*

2. *^{10}B loading in the Boral neutron absorbers: ≥ 0.0372 g/cm^2*

DESIGN FEATURES

Table 3-1 (page 1 of 5)

LIST OF ASME CODE EXCEPTIONS FOR HI-STORM 100 CASK SYSTEM

Component	Reference ASME Code Section/Article	Code Requirement	Exception, Justification & Compensatory Measures
MPC	NB-1100	Statement of requirements for Code stamping of components.	MPC enclosure vessel is designed and will be fabricated in accordance with ASME Code, Section III, Subsection NB to the maximum practical extent, but Code stamping is not required.
MPC	NB-2000	Requires materials to be supplied by ASME-approved material supplier.	Materials will be supplied by Holtec-approved suppliers with Certified Material Test Reports (CMTRs) in accordance with NB-2000 requirements.
MPC Lid and Closure Ring Welds	NB-4243	Full penetration welds required for Category C Joints (flat head to main shell per NB-3352.3).	MPC lid and closure ring are not full penetration welds. They are welded independently to provide a redundant seal. Additionally, a weld efficiency factor of 0.45 has been applied to the analyses of these welds.
MPC Lid to Shell Weld	NB-5230	Radiographic (RT) or ultrasonic (UT) examination required	Only UT or multi-layer liquid penetrant (PT) examination is permitted. If PT alone is used, at a minimum, it will include the root and final weld layers and each approximately 3/8 inch of weld depth.
MPC Closure Ring, Vent and Drain Cover Plate Welds	NB-5230	Radiographic (RT) or ultrasonic (UT) examination required	Root *(if more than one weld pass is required)* and final liquid penetrant examination to be performed in accordance with NB-5245. The MPC vent and drain cover plate welds are leak tested. The closure ring provides independent redundant closure for vent and drain cover plates.

(continued)

DESIGN FEATURES (continued)

3.4 Site-Specific Parameters and Analyses

Site-specific parameters and analyses that will require verification by the system user are, as a minimum, as follows:

1. The temperature of 80° F is the maximum average yearly temperature.

2. The allowed temperature extremes, averaged over a 3-day period, shall be greater than -40° F and less than 125° F.

3. a. *For free-standing casks,* the resultant horizontal acceleration (vectorial sum of two horizontal *Zero Period Accelerations (*ZPAs) at a three-dimensional seismic site), G_H, and vertical ~~acceleration~~ *ZPA*, G_V, expressed as fractions of 'g', shall satisfy the following inequality:

$$G_H + \mu G_V \leq \mu$$

where μ is the Coulomb friction coefficient for the HI-STORM 100/ISFSI pad interface. Unless demonstrated by appropriate testing that a higher value of μ is appropriate for a specific ISFSI, the value of μ used shall be 0.53. Representative values of G_H and G_V combinations for μ = 0.53 are provided in Table 3-2.

Table 3-2

Representative DBE Acceleration Values to Prevent HI-STORM 100 Sliding (μ = 0.53)

Equivalent Vectorial Sum of Two Horizontal ZPA's (G_H in g's)	**Corresponding Vertical ZPA (G_V in g's)**
0.445	0.160
0.424	0.200
0.397	0.250

(continued)

DESIGN FEATURES

3.4 Site-Specific Parameters and Analyses (continued)

[redacted]

(continued)

DESIGN FEATURES

3.4 Site-Specific Parameters and Analyses (continued)

4. The analyzed flood condition of 15 fps water velocity and a height of 125 feet of water (full submergence of the loaded cask) are not exceeded.

5. The potential for fire and explosion shall be addressed, based on site-specific considerations. This includes the condition that the on-site transporter fuel tank will contain no more than 50 gallons of diesel fuel while handling a loaded OVERPACK or TRANSFER CASK.

6. *ISFSI Pad Design*

   a. *For free-standing casks, the ISFSI pad shall be verified by analysis to limit cask deceleration during a design basis drop and/or non-mechanistic tip-over event to ≤ 45 g's at the top of the MPC fuel basket. Analyses shall be performed using methodologies consistent with those described in the HI-STORM FSAR. A lift height above the ISFSI pad is not required to be established if the cask is lifted with a device designed in accordance with ANSI N14.6 and having redundant drop protection features.* ~~In addition to the requirements of 10CFR72.212(b)(2)(ii), the cask storage pads and foundation shall include the following characteristics as applicable to the drop and tipover analyses.~~

[redacted]

~~a. Concrete Thickness: ≤ 36 inches~~

~~b. Concrete Compressive Strength: ≤ 4,200 psi at 28 days~~

~~c. Reinforcement top and bottom (both directions):~~

~~Reinforcement area and spacing determined by analysis~~

~~Reinforcing bar shall be 60 ksi Yield Strength ASTM Material~~

~~d. Soil Effective Modulus of Elasticity: ≤ 28,000 psi (Measured prior to ISFSI pad installation)~~

~~An acceptable method of defining the soil effective modulus of elasticity applicable to the drop and tipover analyses is provided in Table 13 of NUREG/CR-6608 (February, 1998) with soil classification in accordance with ASTM D2487-93, Standard Classification of Soils for Engineering Purposes (Unified Soil Classification System, USCS) and density determination in accordance with ASTM D1586-84, Standard Test Method for Penetration Test and Split/Barrel Sampling of Soils.~~

(continued)

CERTIFICATE OF COMPLIANCE
FOR SPENT FUEL STORAGE CASKS
Supplemental Sheet

9. SPECIAL REQUIREMENTS FOR FIRST SYSTEMS IN PLACE

The heat transfer characteristics of the cask system will be recorded by temperature measurements for the first HI-STORM SFSC Systems (MPC-24, MPC-24E, MPC-24EF, MPC-32, MPC-68, MPC-68F and MPC-68FF) placed into service with a heat load equal to or greater than 10 kW. An analysis shall be performed that demonstrates the temperature measurements validate the analytic methods and predicted thermal behavior described in Chapter 4 of the SAR.

Validation tests shall be performed for each subsequent cask system that has a heat load that exceeds a previously validated heat load by more than 2 kW (e.g., if the initial test was conducted at 10 kW, then no additional testing is needed until the heat load exceeds 12 kW). No additional testing is required for a system after it has been tested at a heat load equal to or greater than 16 kW.

Letter reports summarizing the results of each validation test shall be submitted to the NRC in accordance with 10 CFR 72.4. Cask users may satisfy these requirements by referencing validation test reports submitted to the NRC by other cask users.

10. AUTHORIZATION

The HI-STORM 100 Cask System, which is authorized by this certificate, is hereby approved for general use by holders of 10 CFR Part 50 licenses for nuclear reactors at reactor sites under the general license issued pursuant to 10 CFR 72.210, subject to the conditions specified by 10 CFR 72.212, and the attached Appendix A and Appendix B.

FOR THE U.S. NUCLEAR REGULATORY COMMISSION

E. William Brach, Director
Spent Fuel Project Office
Office of Nuclear Materials Safety
and Safeguards

Attachments:

1. Appendix A
2. Appendix B

TABLE OF CONTENTS

3.1 SFSC INTEGRITY

3.1.1 Multi-Purpose Canister (MPC)

LCO 3.1.1 The MPC shall be dry and helium filled.

APPLICABILITY: During TRANSPORT OPERATIONS and STORAGE OPERATIONS.

ACTIONS

-----NOTE-----
Separate Condition entry is allowed for each MPC.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. MPC cavity vacuum drying pressure limit not met.	A.1 Perform an engineering evaluation to determine the quantity of moisture left in the MPC.	7 days
	<u>AND</u>	
	A.2 Develop and initiate corrective actions necessary to return the MPC to an analyzed condition.	30 days
B. MPC helium backfill pressure limit not met.	B.1 Perform an engineering evaluation to determine the impact of helium differential.	72 hours
	<u>AND</u>	
	B.2 Develop and initiate corrective actions necessary to return the MPC to an analyzed condition.	14 days

ACTIONS
(continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. MPC helium leak rate limit not met.	C.1 Perform an engineering evaluation to determine the impact of increased helium leak rate on heat removal capability and offsite dose.	24 hours
	AND	
	C.2 Develop and initiate corrective actions necessary to return the MPC to an analyzed condition.	7 days
D. Required Actions and associated Completion Times not met.	D.1 Remove all fuel assemblies from the SFSC.	30 days

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.1.1.1	Verify MPC cavity vacuum drying pressure is within the limit specified in Table 3-1 for the applicable MPC model.	Once, prior to TRANSPORT OPERATIONS
SR 3.1.1.2	Verify MPC helium backfill pressure is within the limit specified in Table 3-1 for the applicable MPC model.	Once, prior to TRANSPORT OPERATIONS
SR 3.1.1.3	Verify that the total helium leak rate through the MPC lid confinement weld and the drain and vent port confinement welds is within the limit specified in Table 3-1 for the applicable MPC model.	Once, prior to TRANSPORT OPERATIONS

3.1 SFSC INTEGRITY

3.1.2 SFSC Heat Removal System

LCO 3.1.2 The SFSC Heat Removal System shall be OPERABLE

APPLICABILITY: During STORAGE OPERATIONS.

ACTIONS

---NOTE---

Separate Condition entry is allowed for each SFSC.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. SFSC Heat Removal System inoperable.	A.1 Restore SFSC Heat Removal System to OPERABLE status.	8 hours
B. Required Action A.1 and associated Completion Time not met.	B.1 Perform SR 3.2.3.1.	Immediately and every 12 hours thereafter
	<u>AND</u>	
	B.2.1 Restore SFSC Heat Removal System to OPERABLE status.	48 hours
	<u>OR</u>	
	B.2.2 Transfer the MPC into a TRANSFER CASK.	48 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.1.2.1	Verify all OVERPACK inlet and outlet air ducts are free of blockage.	24 hours
	<u>OR</u>	
	For OVERPACKS with installed temperature monitoring equipment, verify that the difference between the average OVERPACK air outlet temperature and ISFSI ambient temperature is $\leq$ 126°F.	24 hours

3.1 SFSC INTEGRITY

3.1.3 Fuel Cool-Down

LCO 3.1.3 The MPC helium exit temperature shall be ≤ 200° F

--NOTE--
The LCO is only applicable to wet UNLOADING OPERATIONS.

APPLICABILITY: UNLOADING OPERATIONS prior to re-flooding.

ACTIONS

--NOTE--
Separate Condition entry is allowed for each MPC.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. MPC helium gas exit temperature not within limit.	A.1 Establish MPC helium gas exit temperature within limit.	Prior to initiating MPC re-flooding operations
	<u>AND</u>	
	A.2 Ensure adequate heat transfer from the MPC to the environment	22 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.1.3.1	Verify MPC helium gas exit temperature within limit.	Prior to MPC re-flooding operations.

3.2 SFSC RADIATION PROTECTION

3.2.1 TRANSFER CASK Average Surface Dose Rates

LCO 3.2.1 The average surface dose rates of each TRANSFER CASK shall not exceed:

a. 125 Ton TRANSFER CASK

i. 220 mrem/hour (neutron + gamma) on the side;

ii. 60 mrem/hour (neutron + gamma) on the top

b. 100 Ton TRANSFER CASK

i. 1500 mrem/hour (neutron + gamma) on the side;

ii. 315 mrem/hour (neutron + gamma) on the top

APPLICABILITY: During TRANSPORT OPERATIONS.

ACTIONS

---NOTE---

Separate Condition entry is allowed for each TRANSFER CASK.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. TRANSFER CASK average surface dose rate limits not met.	A.1 Administratively verify correct fuel loading. AND A.2 Perform evaluation to verify compliance with the ISFSI offsite radiation protection requirements of 10 CFR Part 20 and 10 CFR Part 72.	24 hours 24 hours

(continued)

3.2 SFSC RADIATION PROTECTION

3.2.3 OVERPACK Average Surface Dose Rates

LCO 3.2.3 The average surface dose rates of each OVERPACK shall not exceed:

a. 50 mrem/hour (neutron + gamma) on the side

b. 10 mrem/hour (neutron + gamma) on the top

c. 40 mrem/hour (neutron + gamma) at the inlet and outlet vent ducts

APPLICABILITY: During STORAGE OPERATIONS.

ACTIONS

-----NOTE-----
Separate Condition entry is allowed for each SFSC.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. OVERPACK average surface dose rate limits not met.	A.1 Administratively verify correct fuel loading.	24 hours
	<u>AND</u>	
	A.2 Perform a written evaluation to verify compliance with the ISFSI offsite radiation protection requirements of 10 CFR Part 20 and 10 CFR Part 72.	48 hours
B. Required Action and associated Completion Time not met.	B.1 Remove all fuel assemblies from the SFSC.	30 days

3.3 SFSC CRITICALITY CONTROL

3.3.1 Boron Concentration

LCO 3.3.1

As required by CoC Appendix B, Table 2.1-2, the concentration of boron in the water in the MPC shall meet the following limits for the applicable MPC model:

a. MPC-24 with one or more fuel assemblies having an initial enrichment greater than the value in Table 2.1-2 for no soluble boron credit and $\leq$ 5.0 wt% ^{235}U: $\geq$ 400 ppmb

b. MPC-24E with one or more fuel assemblies having an initial enrichment greater than the value in Table 2.1-2 for no soluble boron credit and $\leq$ 5.0 wt% ^{235}U: $\geq$ 300 ppmb

c. MPC-32 with all fuel assemblies having an initial enrichment $\leq$ 4.1 wt% ^{235}U: $\geq$ 1900 ppmb

d. MPC-32 with one or more fuel assemblies having an initial enrichment > 4.1 and $\leq$ 5.0 wt% ^{235}U: $\geq$ 2600 ppmb

APPLICABILITY:

During PWR fuel LOADING OPERATIONS with fuel and water in the MPC

AND

During PWR fuel UNLOADING OPERATIONS with fuel and water in the MPC.

ACTIONS

---NOTE---

Separate Condition entry is allowed for each MPC.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Boron concentration not within limit.	A.1 Suspend LOADING OPERATIONS or UNLOADING OPERATIONS.	Immediately
	<u>AND</u>	
	A.2 Suspend positive reactivity additions.	Immediately
	<u>AND</u>	
	A.3 Initiate action to restore boron concentration to within limit.	Immediately

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
---------------------------NOTE--------------------------- This surveillance 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. --		Within 4 hours of entering the Applicability of this LCO. <u>AND</u> Every 48 hours thereafter.
SR 3.3.1.1	Verify boron concentration is within the applicable limit using two independent measurements.	

Table 3-1
MPC Model-Dependent Limits

MPC MODEL	LIMITS
1. MPC-24/24E/24EF	
a. MPC Cavity Vacuum Drying Pressure	≤ 3 torr for ≥ 30 min
b. MPC Helium Backfill Pressure[1]	≥ 29.3 psig and ≤ 33.3 psig
c. MPC Helium Leak Rate	≤ 5.0E-6 atm cc/sec (He)
2. MPC-68/68F/68FF	
a. MPC Cavity Vacuum Drying Pressure	≤ 3 torr for ≥ 30 min
b. MPC Helium Backfill Pressure[1]	≥ 29.3 psig and ≤ 33.3 psig
c. MPC Helium Leak Rate	≤ 5.0E-6 atm cc/sec (He)
3. MPC-32	
a. MPC Cavity Vacuum Drying Pressure	≤ 3 torr for ≥ 30 min
b. MPC Helium Backfill Pressure[1]	≥ 29.3 psig and ≤ 33.3 psig
c. MPC Helium Leak Rate	≤ 5.0E-6 atm cc/sec (He)

[1] Helium used for backfill of MPC shall have a purity of ≥ 99.995%.

5.0 ADMINISTRATIVE CONTROLS AND PROGRAMS

The following programs shall be established, implemented and maintained.

5.1 Deleted

5.2 Deleted

5.3 Deleted

5.4 Radioactive Effluent Control Program

This program implements the requirements of 10 CFR 72.44(d).

a. The HI-STORM 100 Cask System does not create any radioactive materials or have any radioactive waste treatment systems. Therefore, specific operating procedures for the control of radioactive effluents are not required. Specification 3.1.1, Multi-Purpose Canister (MPC), provides assurance that there are not radioactive effluents from the SFSC.

b. This program includes an environmental monitoring program. Each general license user may incorporate SFSC operations into their environmental monitoring programs for 10 CFR Part 50 operations.

c. An annual report shall be submitted pursuant to 10 CFR 72.44(d)(3).

(continued)

5.5 Cask Transport Evaluation Program

This program provides a means for evaluating various transport configurations and transport route conditions to ensure that the design basis drop limits are met. For lifting of the loaded TRANSFER CASK or OVERPACK using devices which are integral to a structure governed by 10 CFR Part 50 regulations, 10 CFR 50 requirements apply. This program is not applicable when the TRANSFER CASK or OVERPACK is in the FUEL BUILDING or is being handled by a device providing support from underneath (i.e., on a rail car, heavy haul trailer, air pads, etc.).

Pursuant to 10 CFR 72.212, this program shall evaluate the site-specific transport route conditions.

a. For free-standing OVERPACKS and the TRANSFER CASK, the following requirements apply:

   1. The lift height above the transport route surface(s) shall not exceed the limits in Table 5-1 except as provided for in Specification 5.5.a.2. Also, the program shall ensure that the transport route conditions (i.e., surface hardness and pad thickness) are equivalent to or less limiting than either Set A or Set B in HI-STORM TSAR Table 2.2.9.

   2. For site-specific transport route surfaces that are not bounded by either the Set A or Set B parameters of TSAR Table 2.2.9, the program may determine lift heights by analysis based on the site-specific conditions to ensure that the impact loading due to design basis drop events does not exceed 45 g's at the top of the MPC fuel basket. These alternative analyses shall be commensurate with the drop analyses described in the Topical Safety Analysis Report for the HI-STORM 100 Cask System. The program shall ensure that these alternative analyses are documented and controlled.

(continued)

ADMINISTRATIVE CONTROLS AND PROGRAMS

5.5 Cask Transport Evaluation Program (continued)

3. The TRANSFER CASK or OVERPACK, when loaded with spent fuel, may be lifted to any height necessary during transportation between the FUEL BUILDING and the CTF and/or ISFSI pad, provided the lifting device is designed in accordance with ANSI N14.6 and has redundant drop protection features.

4. The TRANSFER CASK and MPC, when loaded with spent fuel, may be lifted to those heights necessary to perform cask handling operations, including MPC transfer, provided the lifts are made with structures and components designed in accordance with the criteria specified in Section 3.5 of Appendix B to Certificate of Compliance No. 1014, as applicable.

[redacted]

(continued)

ADMINISTRATIVE CONTROLS AND PROGRAMS

5.5 Cask Transport Evaluation Program (continued)

Table 5-1

TRANSFER CASK and Free-Standing OVERPACK Lifting Requirements

ITEM	ORIENTATION	LIFTING HEIGHT LIMIT (in.)
TRANSFER CASK	Horizontal	42 (Notes 1 and 2)
TRANSFER CASK	Vertical	None Established (Note 2)
OVERPACK	Horizontal	Not Permitted
OVERPACK	Vertical	11 (Note 3)

Notes:
1. To be measured from the lowest point on the TRANSFER CASK (i.e., the bottom edge of the transfer lid)
2. See Technical Specification 5.5.a.3 and 4
3. See Technical Specification 5.5.a.3.

1.0 Definitions

---NOTE---
The defined terms of this section appear in capitalized type and are applicable throughout these Technical Specifications and Bases.

Term	Definition
CASK TRANSFER FACILITY (CTF)	The CASK TRANSFER FACILITY includes the following components and equipment: (1) a Cask Transfer Structure used to stabilize the TRANSFER CASK and MPC during lifts involving spent fuel not bounded by the regulations of 10 CFR Part 50, and (2) Either a stationary lifting device or a mobile lifting device used in concert with the stationary structure to lift the OVERPACK, TRANSFER CASK, and MPC
DAMAGED FUEL ASSEMBLY	DAMAGED FUEL ASSEMBLIES are fuel assemblies with known or suspected cladding defects, as determined by a review of records, greater than pinhole leaks or hairline cracks, empty fuel rod locations that are not filled with dummy fuel rods, or those that cannot be handled by normal means. Fuel assemblies which cannot be handled by normal means due to fuel cladding damage are considered FUEL DEBRIS.
DAMAGED FUEL CONTAINER (DFC)	DFCs are specially designed enclosures for DAMAGED FUEL ASSEMBLIES or FUEL DEBRIS which permit gaseous and liquid media to escape while minimizing dispersal of gross particulates. DFCs authorized for use in the HI-STORM 100 System are as follows: 1. Holtec Dresden Unit 1/Humboldt Bay design 2. Transnuclear Dresden Unit 1 design 3. Holtec Generic BWR design 4. Holtec Generic PWR design
FUEL DEBRIS	FUEL DEBRIS is ruptured fuel rods, severed rods, loose fuel pellets or fuel assemblies with known or suspected defects which cannot be handled by normal means due to fuel cladding damage.

(continued)

1.0 Definitions (continued)

INTACT FUEL ASSEMBLY	INTACT FUEL ASSEMBLIES are fuel assemblies without known or suspected cladding defects greater than pinhole leaks or hairline cracks and which can be handled by normal means. Fuel assemblies without fuel rods in fuel rod locations shall not be classified as INTACT FUEL ASSEMBLIES unless dummy fuel rods are used to displace an amount of water greater than or equal to that displaced by the fuel rod(s).
LOADING OPERATIONS	LOADING OPERATIONS include all licensed activities on an OVERPACK or TRANSFER CASK while it is being loaded with fuel assemblies. LOADING OPERATIONS begin when the first fuel assembly is placed in the MPC and end when the OVERPACK or TRANSFER CASK is suspended from or secured on the transporter. LOADING OPERATIONS does not included MPC transfer between the TRANSFER CASK and the OVERPACK.
MULTI-PURPOSE CANISTER (MPC)	MPCs are the sealed spent nuclear fuel canisters which consist of a honeycombed fuel basket contained in a cylindrical canister shell which is welded to a baseplate, lid with welded port cover plates, and closure ring. The MPC provides the confinement boundary for the contained radioactive materials.
NON-FUEL HARDWARE	NON-FUEL HARDWARE is defined as Burnable Poison Rod Assemblies (BPRAs), Thimble Plug Devices (TPDs), Control Rod Assemblies (CRAs), Axial Power Shaping Rods (APSRs) and other similarly designed devices with different names.
OVERPACK	OVERPACKs are the casks which receive and contain the sealed MPCs for interim storage on the ISFSI. They provide gamma and neutron shielding, and provide for ventilated air flow to promote heat transfer from the MPC to the environs. The OVERPACK does not include the TRANSFER CASK.

(continued)

1.0 Definitions (continued)

PLANAR-AVERAGE INITIAL ENRICHMENT	PLANAR-AVERAGE INITIAL ENRICHMENT is the average of the distributed fuel rod initial enrichments within a given axial plane of the assembly lattice.
SPENT FUEL STORAGE CASKS (SFSCs)	An SFSC is a container approved for the storage of spent fuel assemblies at the ISFSI. The HI-STORM 100 SFSC System consists of the OVERPACK and its integral MPC.
TRANSFER CASK	TRANSFER CASKs are containers designed to contain the MPC during and after loading of spent fuel assemblies and to transfer the MPC to or from the OVERPACK. The HI-STORM 100 System employs either the 125-Ton or the 100-Ton HI-TRAC TRANSFER CASK.
TRANSPORT OPERATIONS	TRANSPORT OPERATIONS include all licensed activities performed on an OVERPACK or TRANSFER CASK loaded with one or more fuel assemblies when it is being moved to and from the ISFSI. TRANSPORT OPERATIONS begin when the OVERPACK or TRANSFER CASK is first suspended from or secured on the transporter and end when the OVERPACK or TRANSFER CASK is at its destination and no longer secured on or suspended from the transporter. TRANSPORT OP RATIONS includes transfer of the MPC between the OVERPACK and the TRANSFER CASK.
UNLOADING OPERATIONS	UNLOADING OPERATIONS include all licensed activities on an SFSC to be unloaded of the contained fuel assemblies. UNLOADING OPERATIONS begin when the OVERPACK or TRANSFER CASK is no longer suspended from or secured on the transporter and end when the last fuel assembly is removed from the SFSC. UNLOADING OPERATIONS does not include MPC transfer between the TRANSFER CASK and the OVERPACK.

2.0 APPROVED CONTENTS

2.1 Fuel Specifications and Loading Conditions

2.1.1 Fuel To Be Stored In The HI-STORM 100 SFSC System

a. INTACT FUEL ASSEMBLIES, DAMAGED FUEL ASSEMBLIES, FUEL DEBRIS, and NON-FUEL HARDWARE meeting the limits specified in Table 2.1-1 and other referenced tables may be stored in the HI-STORM 100 SFSC System.

b. For MPCs partially loaded with stainless steel clad fuel assemblies, all remaining fuel assemblies in the MPC shall meet the decay heat generation limit for the stainless steel clad fuel assemblies.

c. For MPCs partially loaded with DAMAGED FUEL ASSEMBLIES or FUEL DEBRIS, all remaining Zircaloy clad INTACT FUEL ASSEMBLIES in the MPC shall meet the decay heat generation limits for the DAMAGED FUEL ASSEMBLIES. This requirement applies only to uniform fuel loading.

d. For MPC-68's partially loaded with array/class 6x6A, 6x6B, 6x6C, or 8x8A fuel assemblies, all remaining Zircaloy clad INTACT FUEL ASSEMBLIES in the MPC shall meet the decay heat generation limits for the 6x6A, 6x6B, 6x6C, *7x7A* and 8x8A fuel assemblies.

e. All BWR fuel assemblies may be stored with or without Zircaloy channels with the exception of array/class 10x10D and 10x10E fuel assemblies, which may be stored with or without Zircaloy or stainless steel channels.

(continued)

2.0 Approved Contents ~~(continued)~~

2.1 Fuel Specifications and Loading Conditions (cont'd)

2.1.2 Uniform Fuel Loading

Preferential fuel loading shall be used during uniform loading (i.e., any authorized fuel assembly in any fuel storage location) whenever fuel assemblies with significantly different post-irradiation cooling times ($\geq$ 1 year) are to be loaded in the same MPC. Fuel assemblies with the longest post-irradiation cooling times shall be loaded into fuel storage locations at the periphery of the basket. Fuel assemblies with shorter post-irradiation cooling times shall be placed toward the center of the basket. Regionalized fuel loading as described in Technical Specification 2.1.3 below meets the intent of preferential fuel loading.

2.1.3 Regionalized Fuel Loading

Users may choose to store fuel using regionalized loading in lieu of uniform loading to allow higher heat emitting fuel assemblies to be stored than would otherwise be able to be stored using uniform loading. Regionalized loading is limited to those fuel assemblies with Zircaloy (or other alloy of zirconium) cladding. Figures 2.1-1 through 2.1-4 define the regions for the MPC-24, MPC-24E, MPC-24EF, MPC-32, MPC-68, and MPC-68FF models, respectively. Fuel assembly burnup, decay heat, and cooling time limits for regionalized loading are specified in Tables 2.1-6 and 2.1-7. Fuel assemblies used in regionalized loading shall meet all other applicable limits specified in Tables 2.1-1 through 2.1-3.

2.2 Violations

If any Fuel Specifications or Loading Conditions of 2.1 are violated, the following actions shall be completed:

2.2.1 The affected fuel assemblies shall be placed in a safe condition.

2.2.2 Within 24 hours, notify the NRC Operations Center.

2.2.3 Within 30 days, submit a special report which describes the cause of the violation, and actions taken to restore compliance and prevent recurrence.

LEGEND:

REGION 1: [hatched box]

REGION 2: [open box]

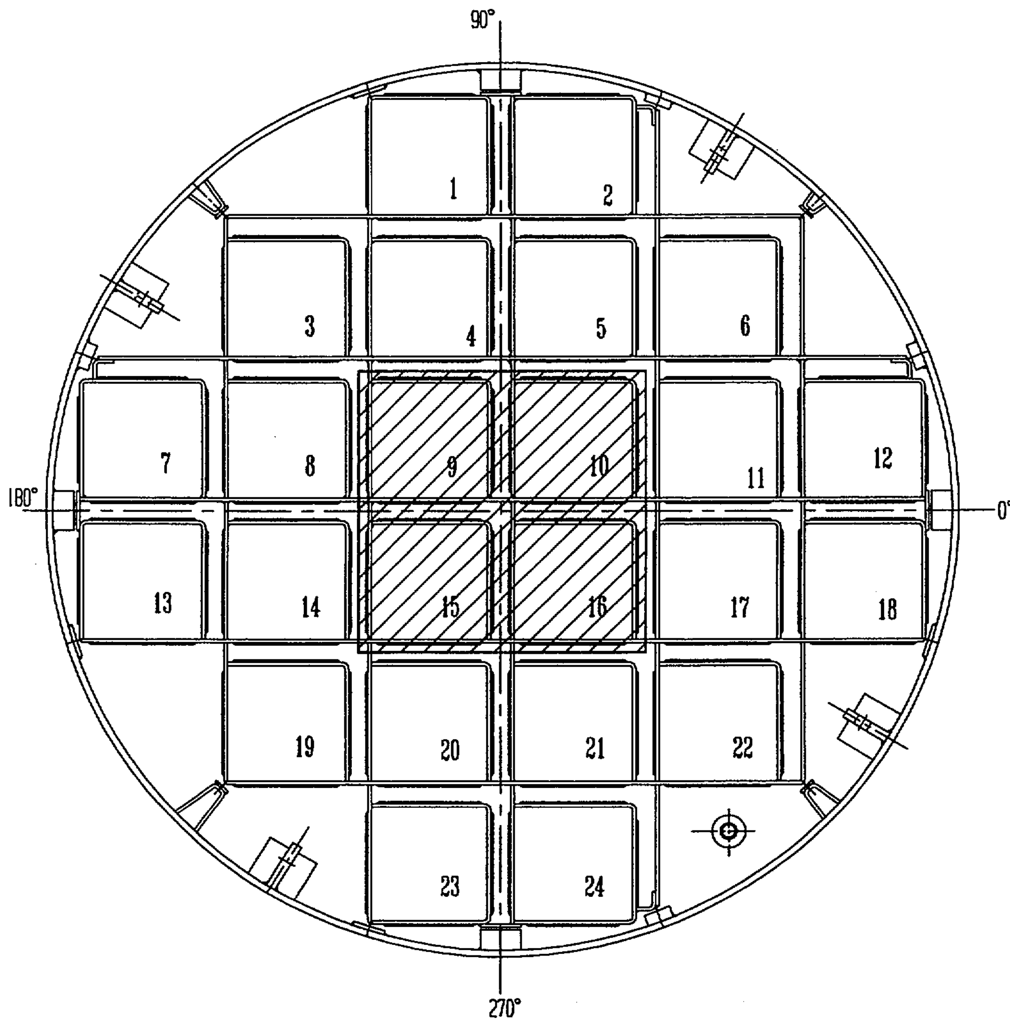

FIGURE 2.1-1
FUEL LOADING REGIONS - MPC-24

LEGEND:

REGION 1: (hatched)

REGION 2: (blank)

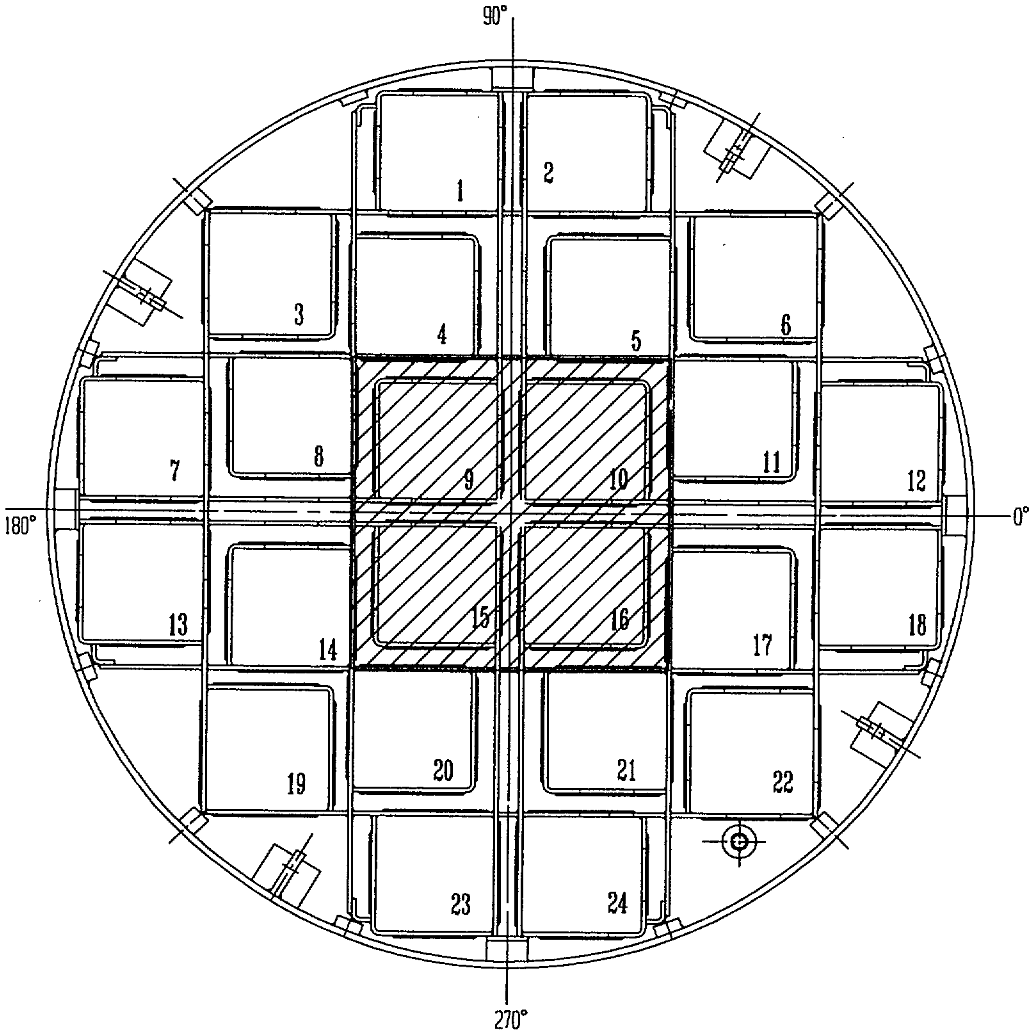

FIGURE 2.1-2

FUEL LOADING REGIONS - MPC-24E/24EF

LEGEND:

REGION 1: (hatched)

REGION 2: (blank)

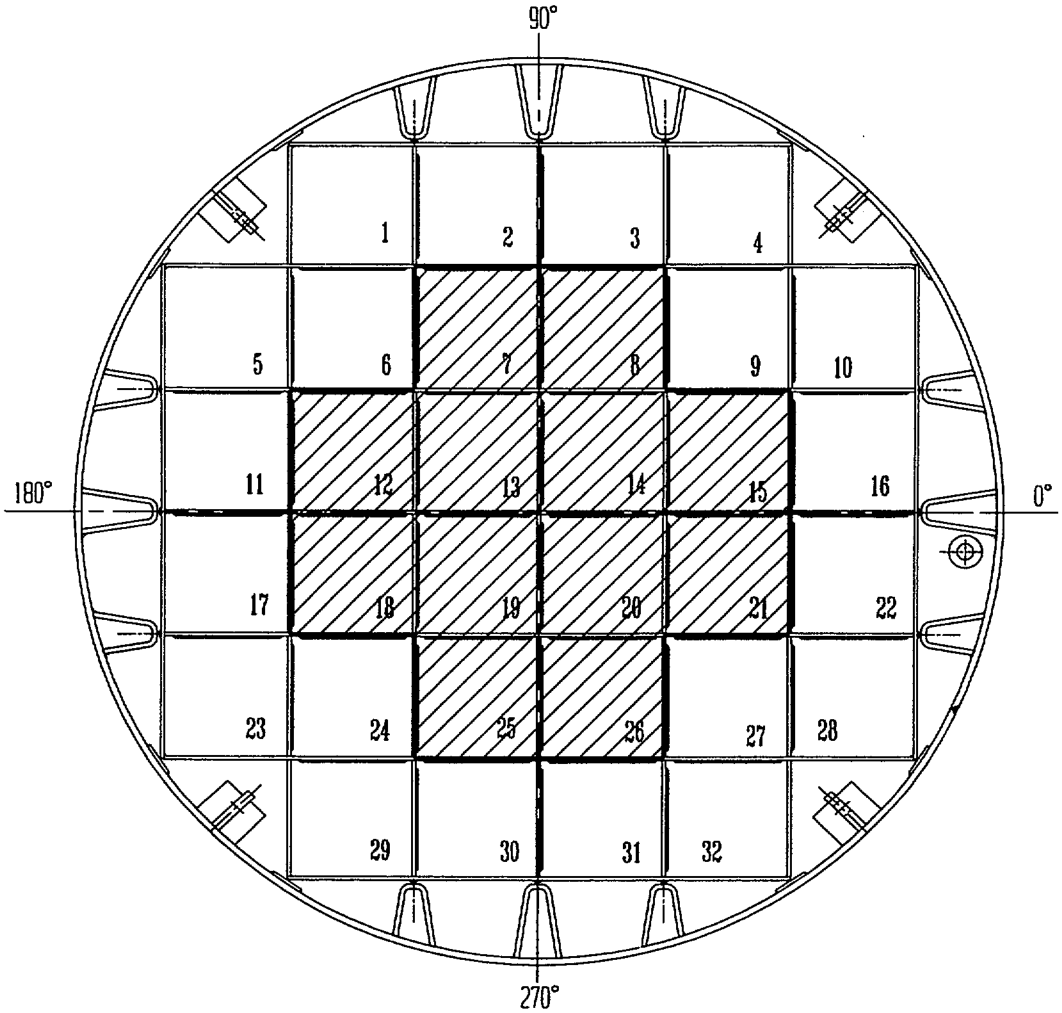

FIGURE 2.1-3

FUEL LOADING REGIONS - MPC-32

LEGEND:

REGION 1: (hatched)

REGION 2: (blank)

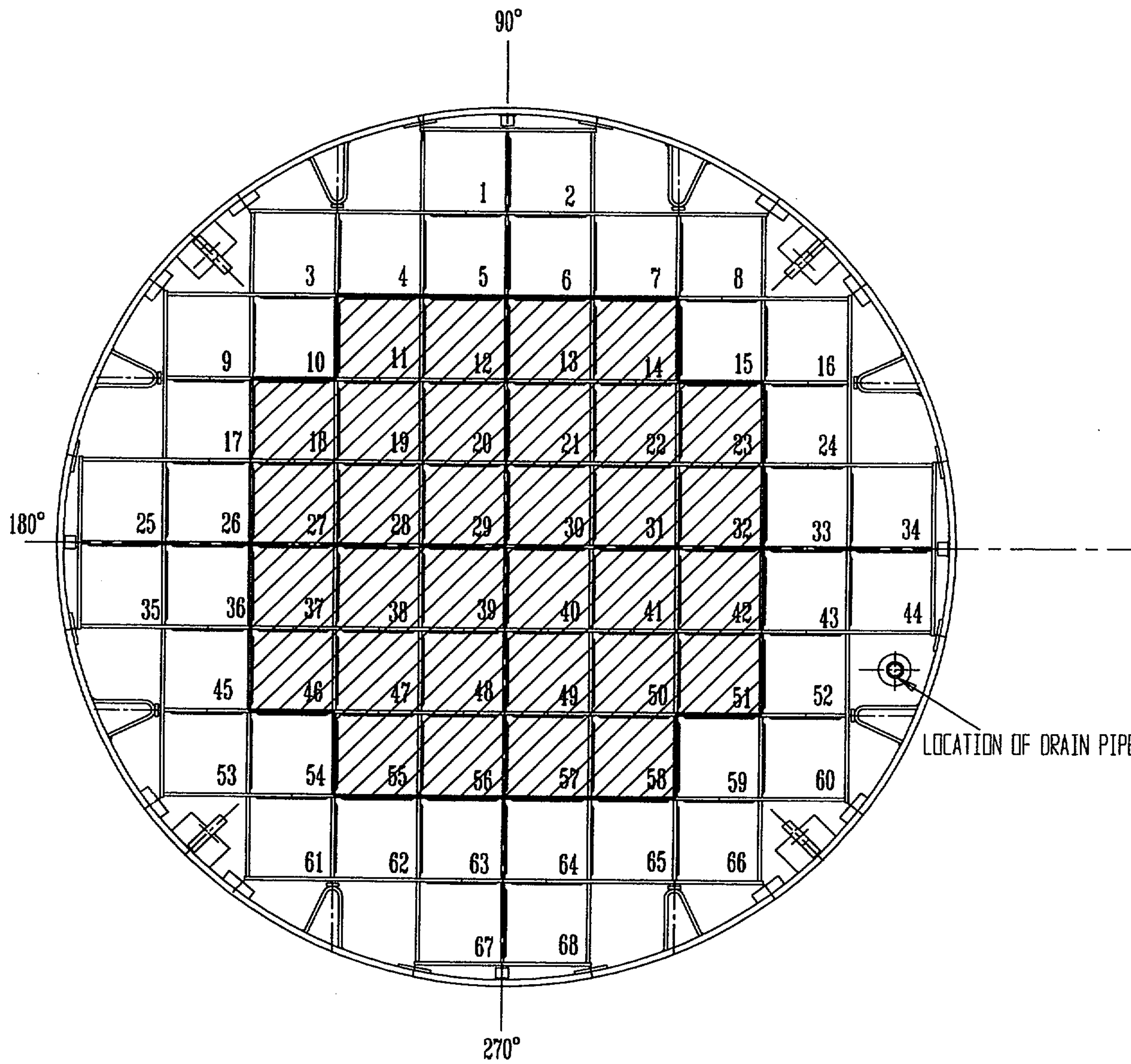

FIGURE 2.1-4

FUEL LOADING REGIONS - MPC-68/68FF

Table 2.1-1 (page 1 of 33)
Fuel Assembly Limits

I. MPC MODEL: MPC-24

A. Allowable Contents

1. Uranium oxide, PWR INTACT FUEL ASSEMBLIES listed in Table 2.1-2, with or without NON-FUEL HARDWARE and meeting the following specifications (Note 1):

a. Cladding Type:	Zircaloy (Zr) or Stainless Steel (SS) as specified in Table 2.1-2 for the applicable fuel assembly array/class.
b. Initial Enrichment:	As specified in Table 2.1-2 for the applicable fuel assembly array/class.
c. Post-irradiation Cooling Time and Average Burnup Per Assembly:	
i. Array/Classes 14x14D,14x14E, and 15x15G	Cooling time $\geq$ 8 years and an average burnup $\leq$ 40,000 MWD/MTU.
ii. All Other Array/Classes	Cooling time and average burnup as specified in Tables 2.1-4 or 2.1-6.
iii. NON-FUEL HARDWARE	As specified in Table 2.1-8.

Table 2.1-1 (page 2 of 33)
Fuel Assembly Limits

I. MPC MODEL: MPC-24 (continued)

A. Allowable Contents (continued)

d. Decay Heat Per Assembly:

i. Array/Classes 14x14D, 14x14E, and 15x15G	≤ 710 Watts
ii All Other Array/Classes	As specified in Tables 2.1-5 or 2.1-7
e. Fuel Assembly Length:	≤ 176.8 inches (nominal design)
f. Fuel Assembly Width:	≤ 8.54 inches (nominal design)
g. Fuel Assembly Weight:	≤ 1,680 lbs (including NON-FUEL HARDWARE)

B. Quantity per MPC: Up to 24 fuel assemblies.

C. ~~Fuel assemblies shall not contain control components.~~ Deleted.

D. DAMAGED FUEL ASSEMBLIES and FUEL DEBRIS are not authorized for loading into the MPC-24.

Note 1: Fuel assemblies containing BPRAs or TPDs may be stored in any fuel cell location. Fuel assemblies containing CRAs or APSRs may only be loaded in fuel storage locations 9, 10, 15, and/or 16. These requirements are in addition to any other requirements specified for uniform or regionalized fuel loading.

Table 2.1-1 (page 3 of 33)
Fuel Assembly Limits

II. MPC MODEL: MPC-68

A. Allowable Contents

- Uranium oxide, BWR INTACT FUEL ASSEMBLIES listed in Table 2.1-3, with or without channels, and meeting the following specifications:

a. Cladding Type:	Zircaloy (Zr) or Stainless Steel (SS) as specified in Table 2.1-3 for the applicable fuel assembly array/class.
b. Maximum PLANAR-AVERAGE INITIAL ENRICHMENT:	As specified in Table 2.1-3 for the applicable fuel assembly array/class.
c. Initial Maximum Rod Enrichment:	As specified in Table 2.1-3 for the applicable fuel assembly array/class.
d. Post-irradiation Cooling Time and Average Burnup Per Assembly:	
i. Array/Classes 6x6A, 6x6C, 7x7A, and 8x8A:	Cooling time $\geq$ 18 years and an average burnup $\leq$ 30,000 MWD/MTU
ii. Array/Class 8x8F	Cooling time $\geq$ 10 years and an average burnup $\leq$ 27,500 MWD/MTU.
iii. Array/Classes 10x10D and 10x10E	Cooling time $\geq$ 10 years and an average burnup $\leq$ 22,500 MWD/MTU.
iv. All Other Array/Classes	As specified in Tables 2.1-4 or 2.1-6.

Table 2.1-1 (page 4 of 33)
Fuel Assembly Limits

II. MPC MODEL: MPC-68 (continued)

A. Allowable Contents (continued)

e. Decay Heat Per Assembly:

i. Array/Classes 6x6A, 6x6C, 7x7A, and 8x8A	≤ 115 Watts
ii. Array/Class 8x8F	≤ 183.5 Watts.
iii. Array/Classes 10x10D and 10x10E	≤ 95 Watts
iv. All Other Array/Classes	As specified in Tables 2.1-5 or 2.1-7.

f. Fuel Assembly Length:	≤ 176.5 inches (nominal design)
g. Fuel Assembly Width:	≤ 5.85 inches (nominal design)
h. Fuel Assembly Weight:	≤ 700 lbs, including channels

Table 2.1-1 (page 5 of 33)
Fuel Assembly Limits

II. MPC MODEL: MPC-68 (continued)

A. Allowable Contents (continued)

2. Uranium oxide, BWR DAMAGED FUEL ASSEMBLIES, with or without channels, placed in DAMAGED FUEL CONTAINERS. Uranium oxide BWR DAMAGED FUEL ASSEMBLIES shall meet the criteria specified in Table 2.1-3 and meet the following specifications:

a. Cladding Type:	Zircaloy (Zr) or Stainless Steel (SS) as specified in Table 2.1-3 for the applicable fuel assembly array/class.
b. Maximum PLANAR-AVERAGE INITIAL ENRICHMENT:	
i. Array/Classes 6x6A, 6x6C, 7x7A, and 8x8A	As specified in Table 2.1-3 for the applicable fuel assembly array/class.
ii. All Other Array/Classes specified in Table 2.1-3	4.0 wt% ^{235}U
c. Initial Maximum Rod Enrichment:	As specified in Table 2.1-3 for the applicable fuel assembly array/class.
d. Post-irradiation Cooling Time and Average Burnup Per Assembly:	
i. Array/Classes 6x6A, 6x6C, 7x7A,and 8x8A	Cooling time $\geq$ 18 years and an average burnup $\leq$ 30,000 MWD/MTU.
ii. Array/Class 8x8F	Cooling time $\geq$ 10 years and an average burnup $\leq$ 27,500 MWD/MTU.
iii. Array/Classes 10x10D and 10x10E	Cooling time $\geq$ 10 years and an average burnup $\leq$ 22,500 MWD/MTU.
iv. All Other Array Classes	As specified in Tables 2.1-4 or 2.1-6.

Table 2.1-1 (page 6 of 33)
Fuel Assembly Limits

II. MPC MODEL: MPC-68 (continued)

A. Allowable Contents (continued)

e. Decay Heat Per Assembly:

i.	Array/Class 6x6A, 6x6C, 7x7A, and 8x8A	≤ 115 Watts
ii.	Array/Class 8x8F	≤ 183.5 Watts
iii.	Array/Classes 10x10D and 10x10E	≤ 95 Watts
iv.	All Other Array/Classes	As specified in Tables 2.1-5 or 2.1-7

f. Fuel Assembly Length:

i.	Array/Class 6x6A, 6x6C, 7x7A, or 8x8A	≤ 135.0 inches (nominal design)
ii.	All Other Array/Classes	≤ 176.5 inches (nominal design)

g. Fuel Assembly Width:

i.	Array/Class 6x6A, 6x6C, 7x7A, or 8x8A	≤ 4.70 inches (nominal design)
ii.	All Other Array/Classes	≤ 5.85 inches (nominal design)

h. Fuel Assembly Weight:

i.	Array/Class 6x6A, 6x6C, 7x7A, or 8x8A	≤ 550 lbs, including channels and DFC
ii.	All Other Array/Classes	≤ 700 lbs, including channels and DFC

Table 2.1-1 (page 7 of 33)
Fuel Assembly Limits

II. MPC MODEL: MPC-68 (continued)

A. Allowable Contents (continued)

3. Mixed oxide (MOX), BWR INTACT FUEL ASSEMBLIES, with or without channels. MOX BWR INTACT FUEL ASSEMBLIES shall meet the criteria specified in Table 2.1-3 for fuel assembly array/class 6x6B, and meet the following specifications:

a. Cladding Type:	Zircaloy (Zr)
b. Maximum PLANAR-AVERAGE INITIAL ENRICHMENT:	As specified in Table 2.1-3 for fuel assembly array/class 6x6B.
c. Initial Maximum Rod Enrichment:	As specified in Table 2.1-3 for fuel assembly array/class 6x6B.
d. Post-irradiation Cooling Time and Average Burnup Per Assembly:	Cooling time $\geq$ 18 years and an average burnup $\leq$ 30,000 MWD/MTIHM.
e. Decay Heat Per Assembly:	$\leq$ 115 Watts
f. Fuel Assembly Length:	$\leq$ 135.0 inches (nominal design)
g. Fuel Assembly Width:	$\leq$ 4.70 inches (nominal design)
h. Fuel Assembly Weight:	$\leq$ 400 lbs, including channels

Table 2.1-1 (page 8 of 33)
Fuel Assembly Limits

II. MPC MODEL: MPC-68 (continued)

A. Allowable Contents (continued)

4. Mixed oxide (MOX), BWR DAMAGED FUEL ASSEMBLIES, with or without channels, placed in DAMAGED FUEL CONTAINERS. MOX BWR DAMAGED FUEL ASSEMBLIES shall meet the criteria specified in Table 2.1-3 for fuel assembly array/class 6x6B, and meet the following specifications:

a. Cladding Type:	Zircaloy (Zr)
b. Maximum PLANAR-AVERAGE INITIAL ENRICHMENT:	As specified in Table 2.1-3 for array/class 6x6B.
c. Initial Maximum Rod Enrichment:	As specified in Table 2.1-3 for array/class 6x6B.
d. Post-irradiation Cooling Time and Average Burnup Per Assembly:	Cooling time $\geq$ 18 years and an average burnup $\leq$ 30,000 MWD/MTIHM.
e. Decay Heat Per Assembly:	$\leq$ 115 Watts
f. Fuel Assembly Length:	$\leq$ 135.0 inches (nominal design)
g. Fuel Assembly Width:	$\leq$ 4.70 inches (nominal design)
h. Fuel Assembly Weight:	$\leq$ 550 lbs, including channels and DFC

Table 2.1-1 (page 9 of 33)
Fuel Assembly Limits

II. MPC MODEL: MPC-68 (continued)

A. Allowable Contents (continued)

5. Thoria rods (ThO_2 and UO_2) placed in Dresden Unit 1 Thoria Rod Canisters and meeting the following specifications:

a. Cladding Type:	Zircaloy (Zr)
b. Composition:	98.2 wt.% ThO_2, 1.8 wt. % UO_2 with an enrichment of 93.5 wt. % ^{235}U.
c. Number of Rods Per Thoria Rod Canister:	$\leq$ 18
d. Decay Heat Per Thoria Rod Canister:	$\leq$ 115 Watts
e. Post-irradiation Fuel Cooling Time and Average Burnup Per Thoria Rod Canister:	A fuel post-irradiation cooling time $\geq$ 18 years and an average burnup $\leq$ 16,000 MWD/MTIHM.
f. Initial Heavy Metal Weight:	$\leq$ 27 kg/canister
g. Fuel Cladding O.D.:	$\geq$ 0.412 inches
h. Fuel Cladding I.D.:	$\leq$ 0.362 inches
i. Fuel Pellet O.D.:	$\leq$ 0.358 inches
j. Active Fuel Length:	$\leq$ 111 inches
k. Canister Weight:	$\leq$ 550 lbs, including fuel

Table 2.1-1 (page 10 of 33)
Fuel Assembly Limits

II. MPC MODEL: MPC-68 (continued)

B. Quantity per MPC:

1. Up to one (1) Dresden Unit 1 Thoria Rod Canister;

2. Up to 68 array/class 6x6A, 6x6B, 6x6C, 7x7A, or 8x8A DAMAGED FUEL ASSEMBLIES in DAMAGE FUEL CONTAINERS;

3. Up to sixteen (16) other BWR DAMAGED FUEL ASSEMBLIES in DAMAGED FUEL CONTAINERS in fuel storage locations 1, 2, 3, 8, 9, 16, 25, 34, 35, 44, 53, 60, 61, 66, 67, and/or 68; and/or

4. Any number of BWR INTACT FUEL ASSEMBLIES up to a total of 68.

C. Array/Class 10x10D and 10x10E fuel assemblies in stainless steel channels must be stored in fuel storage locations 19 - 22, 28 - 31, 38 -41, and/or 47 - 50.

D. Dresden Unit 1 fuel assemblies with one Antimony-Beryllium neutron source are authorized for loading in the MPC-68. The Antimony-Beryllium source material shall be in a water rod location.

Table 2.1-1 (page 11 of 33)
Fuel Assembly Limits

III. MPC MODEL: MPC-68F

A. Allowable Contents

1. Uranium oxide, BWR INTACT FUEL ASSEMBLIES, with or without Zircaloy channels. Uranium oxide BWR INTACT FUEL ASSEMBLIES shall meet the criteria specified in Table 2.1-3 for fuel assembly array class 6x6A, 6x6C, 7x7A or 8x8A, and meet the following specifications:

a. Cladding Type:	Zircaloy (Zr)
b Maximum PLANAR-AVERAGE INITIAL ENRICHMENT:	As specified in Table 2.1-3 for the applicable fuel assembly array/class.
c. Initial Maximum Rod Enrichment:	As specified in Table 2.1-3 for the applicable fuel assembly array/class.
d. Post-irradiation Cooling Time and Average Burnup Per Assembly:	Cooling time $\geq$ 18 years and an average burnup $\leq$ 30,000 MWD/MTU.
e. Decay Heat Per Assembly	$\leq$ 115 Watts
f. Fuel Assembly Length:	$\leq$ 135.0 inches (nominal design)
g. Fuel Assembly Width:	$\leq$ 4.70 inches (nominal design)
h. Fuel Assembly Weight:	$\leq$ 400 lbs, including channels

Table 2.1-1 (page 12 of 33)
Fuel Assembly Limits

III. MPC MODEL: MPC-68F (continued)

A. Allowable Contents (continued)

2. Uranium oxide, BWR DAMAGED FUEL ASSEMBLIES, with or without Zircaloy channels, placed in DAMAGED FUEL CONTAINERS. Uranium oxide BWR DAMAGED FUEL ASSEMBLIES shall meet the criteria specified in Table 2.1-3 for fuel assembly array/class 6x6A, 6x6C, 7x7A, or 8x8A, and meet the following specifications:

a. Cladding Type:	Zircaloy (Zr)
b. Maximum PLANAR-AVERAGE INITIAL ENRICHMENT:	As specified in Table 2.1-3 for the applicable fuel assembly array/class.
c. Initial Maximum Rod Enrichment:	As specified in Table 2.1-3 for the applicable fuel assembly array/class.
d. Post-irradiation Cooling Time and Average Burnup Per Assembly:	Cooling time $\geq$ 18 years and an average burnup $\leq$ 30,000 MWD/MTU.
e. Decay Heat Per Assembly:	$\leq$ 115 Watts
f. Fuel Assembly Length:	$\leq$ 135.0 inches (nominal design)
g. Fuel Assembly Width:	$\leq$ 4.70 inches (nominal design)
h. Fuel Assembly Weight:	$\leq$ 550 lbs, including channels and DFC

Table 2.1-1 (page 13 of 33)
Fuel Assembly Limits

III. MPC MODEL: MPC-68F (continued)

A. Allowable Contents (continued)

3. Uranium oxide, BWR FUEL DEBRIS, with or without Zircaloy channels, placed in DAMAGED FUEL CONTAINERS. The original fuel assemblies for the uranium oxide BWR FUEL DEBRIS shall meet the criteria specified in Table 2.1-3 for fuel assembly array/class 6x6A, 6x6C, 7x7A, or 8x8A, and meet the following specifications:

a. Cladding Type:	Zircaloy (Zr)
b. Maximum PLANAR-AVERAGE INITIAL ENRICHMENT:	As specified in Table 2.1-3 for the applicable original fuel assembly array/class.
c Initial Maximum Rod Enrichment:	As specified in Table 2.1-3 for the applicable original fuel assembly array/class.
d. Post-irradiation Cooling Time and Average Burnup Per Assembly	Cooling time $\geq$ 18 years and an average burnup $\leq$ 30,000 MWD/MTU for the original fuel assembly.
e. Decay Heat Per Assembly	$\leq$ 115 Watts
f. Original Fuel Assembly Length	$\leq$ 135.0 inches (nominal design)
g. Original Fuel Assembly Width	$\leq$ 4.70 inches (nominal design)
h. Fuel Debris Weight	$\leq$ 550 lbs, including channels and DFC

Table 2.1-1 (page 14 of 33)
Fuel Assembly Limits

III. MPC MODEL: MPC-68F (continued)

A. Allowable Contents (continued)

4. Mixed oxide (MOX), BWR INTACT FUEL ASSEMBLIES, with or without Zircaloy channels. MOX BWR INTACT FUEL ASSEMBLIES shall meet the criteria specified in Table 2.1-3 for fuel assembly array/class 6x6B, and meet the following specifications:

a. Cladding Type:	Zircaloy (Zr)
b. Maximum PLANAR-AVERAGE INITIAL ENRICHMENT:	As specified in Table 2.1-3 for fuel assembly array/class 6x6B.
c. Initial Maximum Rod Enrichment:	As specified in Table 2.1-3 for fuel assembly array/class 6x6B.
d. Post-irradiation Cooling Time and Average Burnup Per Assembly:	Cooling time ≥ 18 years and an average burnup ≤ 30,000 MWD/MTIHM.
e. Decay Heat Per Assembly	≤ 115 Watts
f. Fuel Assembly Length:	≤ 135.0 inches (nominal design)
g. Fuel Assembly Width:	≤ 4.70 inches (nominal design)
h. Fuel Assembly Weight:	≤ 400 lbs, including channels

Table 2.1-1 (page 15 of 33)
Fuel Assembly Limits

III. MPC MODEL: MPC-68F (continued)

A. Allowable Contents (continued)

5. Mixed oxide (MOX), BWR DAMAGED FUEL ASSEMBLIES, with or without Zircaloy channels, placed in DAMAGED FUEL CONTAINERS. MOX BWR DAMAGED FUEL ASSEMBLIES shall meet the criteria specified in Table 2.1-3 for fuel assembly array/class 6x6B, and meet the following specifications:

a. Cladding Type:	Zircaloy (Zr)
b. Maximum PLANAR-AVERAGE INITIAL ENRICHMENT:	As specified in Table 2.1-3 for fuel assembly array/class 6x6B.
c. Initial Maximum Rod Enrichment:	As specified in Table 2.1-3 for fuel assembly array/class 6x6B.
d. Post-irradiation Cooling Time and Average Burnup Per Assembly:	Cooling time $\geq$ 18 years and an average burnup $\leq$ 30,000 MWD/MTIHM.
e. Decay Heat Per Assembly	$\leq$ 115 Watts
f. Fuel Assembly Length:	$\leq$ 135.0 inches (nominal design)
g. Fuel Assembly Width:	$\leq$ 4.70 inches (nominal design)
h. Fuel Assembly Weight:	$\leq$ 550 lbs, including channels and DFC

Table 2.1-1 (page 16 of 33)
Fuel Assembly Limits

III. MPC MODEL: MPC-68F (continued)

A. Allowable Contents (continued)

6. Mixed Oxide (MOX), BWR FUEL DEBRIS, with or without Zircaloy channels, placed in DAMAGED FUEL CONTAINERS. The original fuel assemblies for the MOX BWR FUEL DEBRIS shall meet the criteria specified in Table 2.1-3 for fuel assembly array/class 6x6B, and meet the following specifications:

a. Cladding Type:	Zircaloy (Zr)
b. Maximum PLANAR-AVERAGE INITIAL ENRICHMENT:	As specified in Table 2.1-3 for original fuel assembly array/class 6x6B.
c. Initial Maximum Rod Enrichment:	As specified in Table 2.1-3 for original fuel assembly array/class 6x6B.
d. Post-irradiation Cooling Time and Average Burnup Per Assembly:	Cooling time $\geq$ 18 years and an average burnup $\leq$ 30,000 MWD/MTIHM for the original fuel assembly.
e. Decay Heat Per Assembly	$\leq$ 115 Watts
f. Original Fuel Assembly Length:	$\leq$ 135.0 inches (nominal design)
g. Original Fuel Assembly Width:	$\leq$ 4.70 inches (nominal design)
h. Fuel Debris Weight:	$\leq$ 550 lbs, including channels and DFC

Table 2.1-1 (page 17 of 33)
Fuel Assembly Limits

III. MPC MODEL: MPC-68F (continued)

A. Allowable Contents (continued)

7. Thoria rods (ThO_2 and UO_2) placed in Dresden Unit 1 Thoria Rod Canisters and meeting the following specifications:

a. Cladding Type:	Zircaloy (Zr)
b. Composition:	98.2 wt.% ThO_2, 1.8 wt. % UO_2 with an enrichment of 93.5 wt. % ^{235}U.
c. Number of Rods Per Thoria Rod Canister:	$\leq$ 18
d. Decay Heat Per Thoria Rod Canister:	$\leq$ 115 Watts
e. Post-irradiation Fuel Cooling Time and Average Burnup Per Thoria Rod Canister:	A fuel post-irradiation cooling time $\geq$ 18 years and an average burnup $\leq$ 16,000 MWD/MTIHM.
f. Initial Heavy Metal Weight:	$\leq$ 27 kg/canister
g. Fuel Cladding O.D.:	$\geq$ 0.412 inches
h. Fuel Cladding I.D.:	$\leq$ 0.362 inches
i. Fuel Pellet O.D.:	$\leq$ 0.358 inches
j. Active Fuel Length:	$\leq$ 111 inches
k. Canister Weight:	$\leq$ 550 lbs, including fuel

Table 2.1-1 (page 18 of 33)
Fuel Assembly Limits

III. MPC MODEL: MPC-68F (continued)

B. Quantity per MPC (up to a total of 68 assemblies):
(All fuel assemblies must be array/class 6x6A, 6x6B, 6x6C, 7x7A, or 8x8A):

Up to four (4) DFCs containing uranium oxide BWR FUEL DEBRIS or MOX BWR FUEL DEBRIS. The remaining MPC-68F fuel storage locations may be filled with fuel assemblies of the following type, as applicable:

1. Uranium oxide BWR INTACT FUEL ASSEMBLIES;

2. MOX BWR INTACT FUEL ASSEMBLIES;

3. Uranium oxide BWR DAMAGED FUEL ASSEMBLIES placed in DFCs;

4. MOX BWR DAMAGED FUEL ASSEMBLIES placed in DFCs; or

5. Up to one (1) Dresden Unit 1 Thoria Rod Canister.

C. Fuel assemblies with stainless steel channels are not authorized for loading in the MPC-68F.

D. Dresden Unit 1 fuel assemblies with one Antimony-Beryllium neutron source are authorized for loading in the MPC-68F. The Antimony-Beryllium source material shall be in a water rod location.

Table 2.1-1 (page 19 of 33)
Fuel Assembly Limits

IV. MPC MODEL: MPC-24E

A. Allowable Contents

1. Uranium oxide, PWR INTACT FUEL ASSEMBLIES listed in Table 2.1-2, with or without NON-FUEL HARDWARE and meeting the following specifications (Note 1):

a. Cladding Type:	Zircaloy (Zr) or Stainless Steel (SS) as specified in Table 2.1-2 for the applicable fuel assembly array/class
b. Initial Enrichment:	As specified in Table 2.1-2 for the applicable fuel assembly array/class.
c. Post-irradiation Cooling Time and Average Burnup Per Assembly:	
i. Array/Classes 14x14D, 14x14E, and 15x15G	Cooling time $\geq$ 8 years and an average burnup $\leq$ 40,000 MWD/MTU.
ii. All Other Array/Classes	As specified in Tables 2.1-4 or 2.1-6.
iii. NON-FUEL HARDWARE	As specified in Table 2.1-8.

Table 2.1-1 (page 20 of 33)
Fuel Assembly Limits

IV. MPC MODEL: MPC-24E (continued)

A. Allowable Contents (continued)

d. Decay Heat Per Assembly:	
i. Array/Classes 14x14D, 14x14E, and 15x15G	≤ 710 Watts.
ii. All other Array/Classes	As specified in Tables 2.1-5 or 2.1-7.
e. Fuel Assembly Length:	≤ 176.8 inches (nominal design)
f. Fuel Assembly Width:	≤ 8.54 inches (nominal design)
g. Fuel Assembly Weight:	≤ 1,680 lbs (including NON-FUEL HARDWARE)

Table 2.1-1 (page 21 of 33)
Fuel Assembly Limits

IV. MPC MODEL: MPC-24E (continued)

A. Allowable Contents (continued)

- Uranium oxide, PWR DAMAGED FUEL ASSEMBLIES, with or without NON-FUEL HARDWARE, placed in DAMAGED FUEL CONTAINERS. Uranium oxide PWR DAMAGED FUEL ASSEMBLIES shall meet the criteria specified in Table 2.1-2 and meet the following specifications (Note 1):

a. Cladding Type:	Zircaloy (Zr) or Stainless Steel (SS) as specified in Table 2.1-2 for the applicable fuel assembly array/class
b. Initial Enrichment:	$\leq$ 4.0 wt% ^{235}U.
c. Post-irradiation Cooling Time and Average Burnup Per Assembly:	
i. Array/Classes 14x14D, 14x14E, and 15x15G	Cooling time $\geq$ 8 years and an average burnup $\leq$ 40,000 MWD/MTU.
ii. All Other Array/Classes	As specified in Tables 2.1-4 or 2.1-6.
iii. NON-FUEL HARDWARE	As specified in Table 2.1-8.

Table 2.1-1 (page 22 of 33)
Fuel Assembly Limits

IV. MPC MODEL: MPC-24E (continued)

A. Allowable Contents (continued)

d. Decay Heat Per Assembly

i. Array/Classes 14x14D, 14x14E, and 15x15G	≤ 710 Watts.
ii. All Other Array/Classes	As specified in Tables 2.1-5 or 2.1-7.
e. Fuel Assembly Length	≤ 176.8 inches (nominal design)
f. Fuel Assembly Width	≤ 8.54 inches (nominal design)
g. Fuel Assembly Weight	≤ 1,680 lbs (including NON-FUEL HARDWARE and DFC)

B. Quantity per MPC: Up to four (4) DAMAGED FUEL ASSEMBLIES in DAMAGED FUEL CONTAINERS, stored in fuel storage locations 3, 6, 19 and/or 22. The remaining MPC-24E fuel storage locations may be filled with PWR INTACT FUEL ASSEMBLIES meeting the applicable specifications.

C. FUEL DEBRIS is not authorized for loading in the MPC-24E.

Note 1: Fuel assemblies containing BPRAs or TPDs may be stored in any fuel storage location. Fuel assemblies containing CRAs or APSRs must be loaded in fuel storage locations 9,10,15 and/or 16. These requirements are in addition to any other requirements specified for uniform or regionalized fuel loading.

Table 2.1-1 (page 23 of 33)
Fuel Assembly Limits

V. MPC MODEL: MPC-32

A. Allowable Contents

1. Uranium oxide, PWR INTACT FUEL ASSEMBLIES listed in Table 2.1-2, with or without NON-FUEL HARDWARE and meeting the following specifications (Note 1):

a. Cladding Type:	Zircaloy (Zr) or Stainless Steel (SS) as specified in Table 2.1-2 for the applicable fuel assembly array/class
b. Initial Enrichment:	As specified in Table 2.1-2 for the applicable fuel assembly array/class.
c. Post-irradiation Cooling Time and Average Burnup Per Assembly	
i. Array/Classes 14x14D, 14x14E, and 15x15G	Cooling time $\geq$ 9 years and an average burnup $\leq$ 30,000 MWD/MTU or cooling time $\geq$ 20 years and an average burnup $\leq$ 40,000 MWD/MTU.
ii. All Other Array/Classes	As specified in Tables 2.1-4 or 2.1-6.
iii. NON-FUEL HARDWARE	As specified in Table 2.1-8.

Table 2.1-1 (page 24 of 33)
Fuel Assembly Limits

V. MPC MODEL: MPC-32 (continued)

A. Allowable Contents (continued)

d. Decay Heat Per Assembly	
i. Array/Classes 14x14D, 14x14E, and 15x15G	≤ 500 Watts
ii. All Other Array/Classes	As specified in Tables 2.1-5 or 2.1-7.
e. Fuel Assembly Length	≤ 176.8 inches (nominal design)
f. Fuel Assembly Width	≤ 8.54 inches (nominal design)
g. Fuel Assembly Weight	≤ 1,680 lbs (including NON-FUEL HARDWARE)

B. Quantity per MPC: Up to 32 PWR INTACT FUEL ASSEMBLIES.

C. DAMAGED FUEL ASSEMBLIES and FUEL DEBRIS are not authorized for loading in the MPC-32.

Note 1: Fuel assemblies containing BPRAs or TPDs may be stored in any fuel storage location. Fuel assemblies containing CRAs or APSRs must be loaded in fuel storage locations 13, 14, 19, and/or 20. These requirements are in addition to any other requirements specified for uniform or regionalized fuel loading.

Table 2.1-1 (page 25 of 33)
Fuel Assembly Limits

VI. MPC MODEL: MPC-68FF

A. Allowable Contents

1. Uranium oxide or MOX BWR INTACT FUEL ASSEMBLIES listed in Table 2.1-2, with or without channels and meeting the following specifications:

a. Cladding Type:	Zircaloy (Zr) or Stainless Steel (SS) as specified in Table 2.1-3 for the applicable fuel assembly array/class
. Maximum PLANAR-AVERAGE INITIAL ENRICHMENT:	As specified in Table 2.1-3 for the applicable fuel assembly array/class.
. Initial Maximum Rod Enrichment	As specified in Table 2.1-3 for the applicable fuel assembly array/class.
e. Post-irradiation Cooling Time and Average Burnup Per Assembly	
i. Array/Classes 6x6A, 6x6C, 7x7A, and 8x8A	Cooling time $\geq$ 18 years and an average burnup $\leq$ 30,000 MWD/MTU.
ii. Array/Class 8x8F	Cooling time $\geq$ 10 years and an average burnup $\leq$ 27,500 MWD/MTU.
iii. Array/Classes 10x10D and 10x10E	Cooling time $\geq$ 10 years and an average burnup $\leq$ 22,500 MWD/MTU.
iv. All Other Array/Classes	As specified in Tables 2.1-4 or 2.1-6.

Table 2.1-1 (page 26 of 33)
Fuel Assembly Limits

VI. MPC MODEL: MPC-68FF (continued)

A. Allowable Contents (continued)

e. Decay Heat Per Assembly

i. Array/Classes 6x6A, 6x6C, 7x7A, and 8x8A	≤ 115 Watts
ii. Array/Class 8x8F	≤ 183.5 Watts
iii. Array/Classes 10x10D and 10x10E	≤ 95 Watts
iv. All Other Array/Classes	As specified in Tables 2.1-5 or 2.1-7.

f. Fuel Assembly Length	≤ 176.5 inches (nominal design)
g. Fuel Assembly Width	≤ 5.85 inches (nominal design)
h. Fuel Assembly Weight	≤ 700 lbs, including channels

Table 2.1-1 (page 27 of 33)
Fuel Assembly Limits

VI. MPC MODEL: MPC-68FF (continued)

A. Allowable Contents (continued)

2. Uranium oxide or MOX BWR DAMAGED FUEL ASSEMBLIES or FUEL DEBRIS, with or without channels, placed in DAMAGED FUEL CONTAINERS. Uranium oxide and MOX BWR DAMAGED FUEL ASSEMBLIES and FUEL DEBRIS shall meet the criteria specified in Table 2.1-3, and meet the following specifications:

a. Cladding Type:	Zircaloy (Zr) or Stainless Steel (SS) in accordance with Table 2.1-3 for the applicable fuel assembly array/class.
. Maximum PLANAR-AVERAGE INITIAL ENRICHMENT:	
. Array/Classes 6x6A, 6x6B, 6x6C, 7x7A, and 8x8A.	As specified in Table 2.1-3 for the applicable fuel assembly array/class.
ii. All Other Array Classes	$\leq$ 4.0 wt.% ^{235}U.
. Initial Maximum Rod Enrichment	As specified in Table 2.1-3 for the applicable fuel assembly array/class.
d. Post-irradiation Cooling Time and Average Burnup Per Assembly:	
i. Array/Class 6x6A, 6x6B, 6x6C, 7x7A, or 8x8A	Cooling time $\geq$ 18 years and an average burnup $\leq$ 30,000 MWD/MTU (or MWD/MTIHM).
ii. Array/Class 8x8F	Cooling time $\geq$ 10 years and an average burnup $\leq$ 27,500 MWD/MTU.
iii. Array/Class 10x10D and 10x10E	Cooling time $\geq$ 10 years and an average burnup $\leq$ 22,500 MWD/MTU.
iv. All Other Array/Classes	As specified in Tables 2.1-4 or 2.1-6.

Table 2.1-1 (page 28 of 33)
Fuel Assembly Limits

VI. MPC MODEL: MPC-68FF (continued)

A. Allowable Contents (continued)

e. Decay Heat Per Assembly

i.	Array/Class 6x6A, 6x6B, 6x6C, 7x7A, or 8x8A	≤ 115 Watts
ii.	Array/Class 8x8F	≤ 183.5 Watts
iii.	Array/Classes 10x10D and 10x10E	≤ 95 Watts
iv.	All Other Array/Classes	As specified in Tables 2.1-5 or 2.1-7

f. Fuel Assembly Length

i.	Array/Class 6x6A, 6x6B, 6x6C, 7x7A, or 8x8A	≤ 135.0 inches (nominal design)
ii.	All Other Array/Classes	≤ 176.5 inches (nominal design)

g. Fuel Assembly Width

i.	Array/Class 6x6A, 6x6B, 6x6C, 7x7A, or 8x8A	≤ 4.70 inches (nominal design)
ii.	All Other Array/Classes	≤ 5.85 inches (nominal design)

h. Fuel Assembly Weight

i.	Array/Class 6x6A, 6x6B, 6x6C, 7x7A, or 8x8A	≤ 550 lbs, including channels and DFC
ii.	All Other Array/Classes	≤ 700 lbs, including channels and DFC

Table 2.1-1 (page 33 of 33)
Fuel Assembly limits

VI. MPC MODEL: MPC-68FF (continued)

B. Quantity per MPC (up to a total of 68 assemblies)

Up to sixteen (16) DFCs containing BWR DAMAGED FUEL ASSEMBLIES and/or up to eight (8) DFCs containing FUEL DEBRIS. DFCs shall be located only in fuel storage locations 1, 2, 3, 8, 9, 16, 25, 34, 35, 44, 53, 60, 61, 66, 67, and/or 68. The remaining MPC-68FF fuel storage locations may be filled with fuel assemblies of the following type:

3. Uranium Oxide BWR INTACT FUEL ASSEMBLIES; or

4. MOX BWR INTACT FUEL ASSEMBLIES;

C. Dresden Unit 1 fuel assemblies with one Antimony-Beryllium neutron source are authorized for loading in the MPC-68FF. The Antimony-Beryllium source material shall be in a water rod location.

D. Array/Class 10x10D and 10x10E fuel assemblies in stainless steel channels must be stored in fuel storage locations 19 - 22, 28 - 31, 38 -41, and/or 47 - 50.

Table 2.1-1 (page 30 of 33)
Fuel Assembly Limits

VII. MPC MODEL: MPC-24EF

A. Allowable Contents

1. Uranium oxide, PWR INTACT FUEL ASSEMBLIES listed in Table 2.1-2, with or without NON-FUEL HARDWARE and meeting the following specifications (Note 1):

a. Cladding Type:	Zircaloy (Zr) or Stainless Steel (SS) as specified in Table 2.1-2 for the applicable fuel assembly array/class
b. Initial Enrichment:	As specified in Table 2.1-2 for the applicable fuel assembly array/class.
c. Post-irradiation Cooling Time and Average Burnup Per Assembly:	
i. Array/Classes 14x14D, 14x14E, and 15x15G	Cooling time ≥ 8 years and an average burnup ≤ 40,000 MWD/MTU.
ii. All Other Array/Classes	As specified in Tables 2.1-4 or 2.1-6.
iii. NON-FUEL HARDWARE	As specified in Table 2.1-8.

Table 2.1-1 (page 31 of 33)
Fuel Assembly Limits

VII. MPC MODEL: MPC-24EF (continued)

A. Allowable Contents (continued)

d. Decay Heat Per Assembly:	
i. Array/Classes 14x14D, 14x14E, and 15x15G	$\leq$ 710 Watts.
ii. All other Array/Classes	As specified in Tables 2.1-5 or 2.1-7.
e. Fuel Assembly Length:	$\leq$ 176.8 inches (nominal design)
f. Fuel Assembly Width:	$\leq$ 8.54 inches (nominal design)
g. Fuel Assembly Weight:	$\leq$ 1,680 lbs (including NON-FUEL HARDWARE)

Table 2.1-1 (page 32 of 33)
Fuel Assembly Limits

VII. MPC MODEL: MPC-24EF (continued)

A. Allowable Contents (continued)

- Uranium oxide, PWR DAMAGED FUEL ASSEMBLIES and FUEL DEBRIS, with or without NON-FUEL HARDWARE, placed in DAMAGED FUEL CONTAINERS. Uranium oxide PWR DAMAGED FUEL ASSEMBLIES and FUEL DEBRIS shall meet the criteria specified in Table 2.1-2 and meet the following specifications (Note 1):

a. Cladding Type:	Zircaloy (Zr) or Stainless Steel (SS) as specified in Table 2.1-2 for the applicable fuel assembly array/class
b. Initial Enrichment:	$\leq$ 4.0 wt% ^{235}U.
c. Post-irradiation Cooling Time and Average Burnup Per Assembly:	
i. Array/Classes 14x14D, 14x14E, and 15x15G	Cooling time $\geq$ 8 years and an average burnup $\leq$ 40,000 MWD/MTU.
ii. All Other Array/Classes	As specified in Tables 2.1-4 or 2.1-6.
iii. NON-FUEL HARDWARE	As specified in Table 2.1-8.

Table 2.1-1 (page 33 of 33)
Fuel Assembly Limits

VII. MPC MODEL: MPC-24EF (continued)

A. Allowable Contents (continued)

d. Decay Heat Per Assembly	
i. Array/Classes 14x14D, 14x14E, and 15x15G	≤ 710 Watts.
ii. All Other Array/Classes	As specified in Tables 2.1-5 or 2.1-7.
e. Fuel Assembly Length	≤ 176.8 inches (nominal design)
f. Fuel Assembly Width	≤ 8.54 inches (nominal design)
g. Fuel Assembly Weight	≤ 1,680 lbs (including NON-FUEL HARDWARE and DFC)

B. Quantity per MPC: Up to four (4) DAMAGED FUEL ASSEMBLIES and/or FUEL DEBRIS in DAMAGED FUEL CONTAINERS, stored in fuel storage locations 3, 6, 19 and/or 22. The remaining MPC-24E fuel storage locations may be filled with PWR INTACT FUEL ASSEMBLIES meeting the applicable specifications.

Note 1: Fuel assemblies containing BPRAs or TPDs may be stored in any fuel storage location. Fuel assemblies containing CRAs or APSRs must be loaded in fuel storage locations 9,10,15 and/or 16. These requirements are in addition to any other requirements specified for uniform or regionalized fuel loading.

Table 2.1-2 (page 1 of 4)
PWR FUEL ASSEMBLY CHARACTERISTICS (Note 1)

Fuel Assembly Array/Class	**14x14A**	**14x14B**	**14x14C**	**14x14D**	**14x14E**
Clad Material (Note 2)	Zr	Zr	Zr	SS	SS
Design Initial U (kg/assy.) (Note 3)	≤ 407	≤ 407	≤ 425	≤ 400	≤ 206
Initial Enrichment (MPC-24, 24E and 24EF without soluble boron credit) (wt % ^{235}U) (Note 7)	≤ 4.6 (24) ≤ 5.0 (24E/24EF)	≤ 4.6 (24) ≤ 5.0 (24E/24EF)	≤ 4.6 (24) ≤ 5.0 (24E/24EF)	≤ 4.0 (24) ≤ 5.0 (24E/24EF)	≤ 5.0 (24) ≤ 5.0 (24E/24EF)
Initial Enrichment (MPC-24, 24E, 24EF, or 32 with soluble boron credit - see Notes 5 and 7) (wt % ^{235}U)	≤ 5.0	≤ 5.0	≤ 5.0	≤ 5.0	≤ 5.0
No. of Fuel Rod Locations	179	179	176	180	173
Fuel Rod Clad O.D. (in.)	≥ 0.400	≥ 0.417	≥ 0.440	≥ 0.422	≥ 0.3415
Fuel Rod Clad I.D. (in.)	≤ 0.3514	≤ 0.3734	≤ 0.3880	≤ 0.3890	≤ 0.3175
Fuel Pellet Dia. (in.)	≤ 0.3444	≤ 0.3659	≤ 0.3805	≤ 0.3835	≤ 0.3130
Fuel Rod Pitch (in.)	≤ 0.556	≤ 0.556	≤ 0.580	≤ 0.556	Note 6
Active Fuel Length (in.)	≤ 150	≤ 150	≤ 150	≤ 144	≤ 102
No. of Guide and/or Instrument Tubes	17	17	5 (Note 4)	16	0
Guide/Instrument Tube Thickness (in.)	≥ 0.017	≥ 0.017	≥ 0.038	≥ 0.0145	N/A

Table 2.1-2 (page 2 of 4)
PWR FUEL ASSEMBLY CHARACTERISTICS (Note 1)

Fuel Assembly Array/Class	**15x15A**	**15x15B**	**15x15C**	**15x15D**	**15x15E**	**15x15F**
Clad Material (Note 2)	Zr	Zr	Zr	Zr	Zr	Zr
Design Initial U (kg/assy.) (Note 3)	≤ 464	≤ 464	≤ 464	≤ 475	≤ 475	≤ 475
Initial Enrichment (MPC-24, 24E and 24EF without soluble boron credit) (wt % ^{235}U) (Note 7)	≤ 4.1 (24) ≤ 4.5 (24E/24EF)	≤ 4.1 (24) ≤ 4.5 (24E/24EF)	≤ 4.1 (24) ≤ 4.5 (24E/24EF)	≤ 4.1 (24) ≤ 4.5 (24E/24EF)	≤ 4.1 (24) ≤ 4.5 (24E/24EF)	≤ 4.1 (24) ≤ 4.5 (24E/24EF)
Initial Enrichment (MPC-24, 24E, 24EF, or 32 with soluble boron credit - see Notes 5 and 7) (wt % ^{235}U)	≤ 5.0	≤ 5.0	≤ 5.0	≤ 5.0	≤ 5.0	≤ 5.0
No. of Fuel Rod Locations	204	204	204	208	208	208
Fuel Rod Clad O.D. (in.)	≥ 0.418	≥ 0.420	≥ 0.417	≥ 0.430	≥ 0.428	≥ 0.428
Fuel Rod Clad I.D. (in.)	≤ 0.3660	≤ 0.3736	≤ 0.3640	≤ 0.3800	≤ 0.3790	≤ 0.3820
Fuel Pellet Dia. (in.)	≤ 0.3580	≤ 0.3671	≤ 0.3570	≤ 0.3735	≤ 0.3707	≤ 0.3742
Fuel Rod Pitch (in.)	≤ 0.550	≤ 0.563	≤ 0.563	≤ 0.568	≤ 0.568	≤ 0.568
Active Fuel Length (in.)	≤ 150	≤ 150	≤ 150	≤ 150	≤ 150	≤ 150
No. of Guide and/or Instrument Tubes	21	21	21	17	17	17
Guide/Instrument Tube Thickness (in.)	≥ 0.0165	≥ 0.015	≥ 0.0165	≥ 0.0150	≥ 0.0140	≥ 0.0140

Table 2.1-2 (page 3 of 4)
PWR FUEL ASSEMBLY CHARACTERISTICS (Note 1)

Fuel Assembly Array/ Class	**15x15G**	**15x15H**	**16x16A**	**17x17A**	**17x17B**	**17x17C**
Clad Material (Note 2)	SS	Zr	Zr	Zr	Zr	Zr
Design Initial U (kg/assy.) (Note 3)	≤ 420	≤ 475	≤ 443	≤ 467	≤ 467	≤ 474
Initial Enrichment (MPC-24 24E, and 24EF without soluble boron credit) (wt % ^{235}U) (Note 7)	≤ 4.0 (24) ≤ 4.5 (24E/24EF)	≤ 3.8 (24) ≤ 4.2 (24E/24EF)	≤ 4.6 (24) ≤ 5.0 (24E/24EF)	≤ 4.0 (24) ≤ 4.4 (24E/24EF)	≤ 4.0 (24) ≤ 4.4 (24E/24EF)	≤ 4.0 (24) ≤ 4.4 (24E/24EF)
Initial Enrichment (MPC-24, 24E, 24EF, or 32 with soluble boron credit - see Notes 5 and 7) (wt % ^{235}U)	≤ 5.0	≤ 5.0	≤ 5.0	≤ 5.0	≤ 5.0	≤ 5.0
No. of Fuel Rod Locations	204	208	236	264	264	264
Fuel Rod Clad O.D. (in.)	≥ 0.422	≥ 0.414	≥ 0.382	≥ 0.360	≥ 0.372	≥ 0.377
Fuel Rod Clad I.D. (in.)	≤ 0.3890	≤ 0.3700	≤ 0.3320	≤ 0.3150	≤ 0.3310	≤ 0.3330
Fuel Pellet Dia. (in.)	≤ 0.3825	≤ 0.3622	≤ 0.3255	≤ 0.3088	≤ 0.3232	≤ 0.3252
Fuel Rod Pitch (in.)	≤ 0.563	≤ 0.568	≤ 0.506	≤ 0.496	≤ 0.496	≤ 0.502
Active Fuel Length (in.)	≤ 144	≤ 150	≤ 150	≤ 150	≤ 150	≤ 150
No. of Guide and/or Instrument Tubes	21	17	5 (Note 4)	25	25	25
Guide/Instrument Tube Thickness (in.)	≥ 0.0145	≥ 0.0140	≥ 0.0400	≥ 0.016	≥ 0.014	≥ 0.020

Table 2.1-2 (page 4 of 4)
PWR FUEL ASSEMBLY CHARACTERISTICS

Notes:

. All dimensions are design nominal values. Maximum and minimum dimensions are specified to bound variations in design nominal values among fuel assemblies within a given array/class.

2. Zr designates cladding material made of zirconium or zirconium alloys.

3. Design initial uranium weight is the nominal uranium weight specified for each assembly by the fuel manufacturer or reactor user. For each PWR fuel assembly, the total uranium weight limit specified in this table may be increased up to 2.0 percent for comparison with users' fuel records to account for manufacturer's tolerances.

4. Each guide tube replaces four fuel rods.

5. Soluble boron concentration per LCO 3.3.1.

6. This fuel assembly array/class includes only the Indian Point Unit 1 fuel assembly. This fuel assembly has two pitches in different sectors of the assembly.

7. For those MPCs loaded with both INTACT FUEL ASSEMBLIES and DAMAGED FUEL ASSEMBLIES or FUEL DEBRIS, the maximum initial enrichment of the INTACT FUEL ASSEMBLIES is limited to the maximum initial enrichment of the DAMAGED FUEL ASSEMBLIES and FUEL DEBRIS (i.e., 4.0 wt.% ^{235}U).

Table 2.1-3 (page 1 of 5)
BWR FUEL ASSEMBLY CHARACTERISTICS (Note 1)

Fuel Assembly Array/Class	**6x6A**	**6x6B**	**6x6C**	**7x7A**	**7x7B**	**8x8A**
Clad Material (Note 2)	Zr	Zr	Zr	Zr	Zr	Zr
Design Initial U (kg/assy.) (Note 3)	110	110	110	≤ 100	≤ 195	≤ 120
Maximum PLANAR-AVERAGE INITIAL ENRICHMENT (wt.% ^{235}U) (Note 14)	≤ 2.7	≤ 2.7 for the UO_2 rods. See Note 4 for MOX rods	≤ 2.7	≤ 2.7	≤ 4.2	≤ 2.7
Initial Maximum Rod Enrichment (wt.% ^{235}U)	≤ 4.0	≤ 4.0	≤ 4.0	≤ 5.5	≤ 5.0	≤ 4.0
No. of Fuel Rod Locations	35 or 36	35 or 36 (up to 9 MOX rods)	36	49	49	63 or 64
Fuel Rod Clad O.D. (in.)	≥ 0.5550	≥ 0.5625	≥ 0.5630	≥ 0.4860	≥ 0.5630	≥ 0.4120
Fuel Rod Clad I.D. (in.)	≤ 0.5105	≤ 0.4945	≤ 0.4990	≤ 0.4204	≤ 0.4990	≤ 0.3620
Fuel Pellet Dia. (in.)	≤ 0.4980	≤ 0.4820	≤ 0.4880	≤ 0.4110	≤ 0.4910	≤ 0.3580
Fuel Rod Pitch (in.)	≤ 0.710	≤ 0.710	≤ 0.740	≤ 0.631	≤ 0.738	≤ 0.523
Active Fuel Length (in.)	≤ 120	≤ 120	≤ 77.5	≤ 80	≤ 150	≤ 120
No. of Water Rods (Note 11)	1 or 0	1 or 0	0	0	0	1 or 0
Water Rod Thickness (in.)	> 0	> 0	N/A	N/A	N/A	≥ 0
Channel Thickness (in.)	≤ 0.060	≤ 0.060	≤ 0.060	≤ 0.060	≤ 0.120	≤ 0.100

Table 2.1-3 (2 of 5)
BWR FUEL ASSEMBLY CHARACTERISTICS (Note 1)

Fuel Assembly Array/Class	**8x8B**	**8x8C**	**8x8D**	**8x8E**	**8x8F**	**9x9A**
Clad Material (Note 2)	Zr	Zr	Zr	Zr	Zr	Zr
Design Initial U (kg/assy.) (Note 3)	≤ 191	≤ 191	≤ 191	< 191	≤ 191	≤ 179
Maximum PLANAR-AVERAGE INITIAL ENRICHMENT (wt.% ^{235}U) (Note 14)	≤ 4.2	≤ 4.2	≤ 4.2	≤ 4.2	≤ 4.0	≤ 4.2
Initial Maximum Rod Enrichment (wt.% ^{235}U)	≤ 5.0	≤ 5.0	≤ 5.0	≤ 5.0	≤ 5.0	≤ 5.0
No. of Fuel Rod Locations	63 or 64	62	60 or 61	59	64	74/66 (Note 5)
Fuel Rod Clad O.D. (in.)	≥ 0.4840	≥ 0.4830	≥ 0.4830	≥ 0.4930	≥ 0.4576	≥ 0.4400
Fuel Rod Clad I.D. (in.)	≤ 0.4295	≤ 0.4250	≤ 0.4230	≤ 0.4250	≤ 0.3996	≤ 0.3840
Fuel Pellet Dia. (in.)	≤ 0.4195	≤ 0.4160	≤ 0.4140	≤ 0.4160	≤ 0.3913	≤ 0.3760
Fuel Rod Pitch (in.)	≤ 0.642	≤ 0.641	≤ 0.640	≤ 0.640	≤ 0.609	≤ 0.566
Design Active Fuel Length (in.)	≤ 150	≤ 150	≤ 150	≤ 150	≤ 150	≤ 150
No. of Water Rods (Note 11)	1 or 0	2	1 - 4 (Note 7)	5	N/A (Note 12)	2
Water Rod Thickness (in.)	≥ 0.034	> 0.00	> 0.00	≥ 0.034	≥ 0.0315	> 0.00
Channel Thickness (in.)	≤ 0.120	≤ 0.120	≤ 0.120	≤ 0.100	≤ 0.055	≤ 0.120

Table 2.1-3 (page 3 of 5)
BWR FUEL ASSEMBLY CHARACTERISTICS (Note 1)

Fuel Assembly Array/Class	**9x9B**	**9x9C**	**9x9D**	**9x9E (Note 13)**	**9x9F (Note 13)**	**9x9G**
Clad Material (Note 2)	Zr	Zr	Zr	Zr	Zr	Zr
Design Initial U (kg/assy.) (Note 3)	≤ 179	≤ 179	≤ 179	≤ 179	≤ 179	≤ 179
Maximum PLANAR-AVERAGE INITIAL ENRICHMENT (wt.% ^{235}U) (Note 14)	≤ 4.2	≤ 4.2	≤ 4.2	≤ 4.0	≤ 4.0	≤ 4.2
Initial Maximum Rod Enrichment (wt.% ^{235}U)	≤ 5.0	≤ 5.0	≤ 5.0	≤ 5.0	≤ 5.0	≤ 5.0
No. of Fuel Rod Locations	72	80	79	76	76	72
Fuel Rod Clad O.D. (in.)	≥ 0.4330	≥ 0.4230	≥ 0.4240	≥ 0.4170	≥ 0.4430	≥ 0.4240
Fuel Rod Clad I.D. (in.)	≤ 0.3810	≤ 0.3640	≤ 0.3640	≤ 0.3640	≤ 0.3860	≤ 0.3640
Fuel Pellet Dia. (in.)	≤ 0.3740	≤ 0.3565	≤ 0.3565	≤ 0.3530	≤ 0.3745	≤ 0.3565
Fuel Rod Pitch (in.)	≤ 0.572	≤ 0.572	≤ 0.572	≤ 0.572	≤ 0.572	≤ 0.572
Design Active Fuel Length (in.)	≤ 150	≤ 150	≤ 150	≤ 150	≤ 150	≤ 150
No. of Water Rods (Note 11)	1 (Note 6)	1	2	5	5	1 (Note 6)
Water Rod Thickness (in.)	> 0.00	≥ 0.020	≥ 0.0300	≥ 0.0120	≥ 0.0120	≥ 0.0320
Channel Thickness (in.)	≤ 0.120	≤ 0.100	≤ 0.100	≤ 0.120	≤ 0.120	≤ 0.120

Table 2.1-3 (page 4 of 5)
BWR FUEL ASSEMBLY CHARACTERISTICS (Note 1)

Fuel Assembly Array/Class	**10x10A**	**10x10B**	**10x10C**	**10x10D**	**10x10E**
Clad Material (Note 2)	Zr	Zr	Zr	SS	SS
Design Initial U (kg/assy.) (Note 3)	≤ 188	≤ 188	≤ 188	≤ 125	≤ 125
Maximum PLANAR-AVERAGE INITIAL ENRICHMENT (wt.% ^{235}U) (Note 14)	≤ 4.2	≤ 4.2	≤ 4.2	≤ 4.0	≤ 4.0
Initial Maximum Rod Enrichment (wt.% ^{235}U)	≤ 5.0	≤ 5.0	≤ 5.0	≤ 5.0	≤ 5.0
No. of Fuel Rod Locations	92/78 (Note 8)	91/83 (Note 9)	96	100	96
Fuel Rod Clad O.D. (in.)	≥ 0.4040	≥ 0.3957	≥ 0.3780	≥ 0.3960	≥ 0.3940
Fuel Rod Clad I.D. (in.)	≤ 0.3520	≤ 0.3480	≤ 0.3294	≤ 0.3560	≤ 0.3500
Fuel Pellet Dia. (in.)	≤ 0.3455	≤ 0.3420	≤ 0.3224	≤ 0.3500	≤ 0.3430
Fuel Rod Pitch (in.)	≤ 0.510	≤ 0.510	≤ 0.488	≤ 0.565	≤ 0.557
Design Active Fuel Length (in.)	≤ 150	≤ 150	≤ 150	≤ 83	≤ 83
No. of Water Rods (Note 11)	2	1 (Note 6)	5 (Note 10)	0	4
Water Rod Thickness (in.)	≥ 0.0300	> 0.00	≥ 0.031	N/A	≥ 0.022
Channel Thickness (in.)	≤ 0.120	≤ 0.120	≤ 0.055	≤ 0.080	≤ 0.080

Table 2.1-3 (page 5 of 5)
BWR FUEL ASSEMBLY CHARACTERISTICS

Notes:

1. All dimensions are design nominal values. Maximum and minimum dimensions are specified to bound variations in design nominal values among fuel assemblies within a given array/class.

2. Zr designates cladding material made of zirconium or zirconium alloys.

3. Design initial uranium weight is the nominal uranium weight specified for each assembly by the fuel manufacturer or reactor user. For each BWR fuel assembly, the total uranium weight limit specified in this table may be increased up to 1.5 percent for comparison with users' fuel records to account for manufacturer tolerances.

4. $\leq$ 0.635 wt. % ^{235}U and $\leq$ 1.578 wt. % total fissile plutonium (^{239}Pu and ^{241}Pu), (wt. % of total fuel weight, i.e., UO_2 plus PuO_2).

5. This assembly class contains 74 total rods; 66 full length rods and 8 partial length rods.

6. Square, replacing nine fuel rods.

7. Variable.

8. This assembly contains 92 total fuel rods; 78 full length rods and 14 partial length rods.

9. This assembly class contains 91 total fuel rods; 83 full length rods and 8 partial length rods.

10. One diamond-shaped water rod replacing the four center fuel rods and four rectangular water rods dividing the assembly into four quadrants.

11. These rods may also be sealed at both ends and contain Zr material in lieu of water.

12. This assembly is known as "QUAD+." It has four rectangular water cross segments dividing the assembly into four quadrants.

13. For the SPC 9x9-5 fuel assembly, each fuel rod must meet either the 9x9E or the 9x9F set of limits for clad O.D., clad I.D., and pellet diameter.

14. For those MPCs loaded with both INTACT FUEL ASSEMBLIES and DAMAGED FUEL ASSEMBLIES or FUEL DEBRIS, the maximum PLANAR AVERAGE INITIAL ENRICHMENT for the INTACT FUEL ASSEMBLIES is limited to 3.7 wt.% ^{235}U, as applicable.

Table 2.1-4

FUEL ASSEMBLY COOLING AND MAXIMUM AVERAGE BURNUP
(UNIFORM FUEL LOADING)

Post-Irradiation Cooling Time (years)	MPC-24 PWR Assembly Burnup (INTACT FUEL ASSEMBLIES) (MWD/MTU)	MPC-24E/24EF PWR Assembly Burnup (INTACT FUEL ASSEMBLIES) (MWD/MTU)	MPC-24E/24EF PWR Assembly Burnup (DAMAGED FUEL ASSEMBLIES AND FUEL DEBRIS) (MWD/MTU)	MPC-32 PWR Assembly Burnup (INTACT FUEL ASSEMBLIES (MWD/MTU)	MPC-68/68FF BWR Assembly Burnup (INTACT FUEL ASSEMBLIES) (MWD/MTU)	MPC-68/68FF BWR Assembly Burnup (DAMAGED FUEL ASSEMBLIES AND FUEL DEBRIS) (MWD/MTU)
≥ 5	40,600	41,100	39,200	32,200	38,300	33,400
≥ 6	45,000	45,000	43,700	36,500	41,600	36,600
≥ 7	45,900	46,300	45,200	37,500	42,300	37,000
≥ 8	48,300	48,900	47,300	39,900	44,800	39,100
≥ 9	50,300	50,700	49,000	41,500	46,600	40,700
≥ 10	51,600	52,100	50,100	42,900	48,000	41,900
≥ 11	53,100	53,700	51,500	44,100	49,600	43,000
≥ 12	54,500	55,100	52,600	45,000	50,800	44,100
≥ 13	55,600	56,100	53,800	45,700	51,800	45,000
≥ 14	56,500	57,100	54,900	46,500	52,700	45,800
≥ 15	57,400	58,000	55,800	47,200	53,900	46,500

Note: 1. Linear interpolation between points is permitted.

Table 2.1-5

FUEL ASSEMBLY COOLING AND MAXIMUM DECAY HEAT
(UNIFORM FUEL LOADING)

Post-Irradiation Cooling Time (years)	MPC-24 PWR Assembly Decay Heat (INTACT FUEL ASSEMBLIES) (Watts)	MPC-24E/24EF PWR Assembly Decay Heat (INTACT FUEL ASSEMBLIES) (Watts)	MPC-24E/24EF PWR Assembly Decay Heat (DAMAGED FUEL ASSEMBLIES AND FUEL DEBRIS) (Watts)	MPC-32 PWR Assembly Decay Heat (INTACT FUEL ASSEMBLIES (Watts)	MPC-68/68FF BWR Assembly Decay Heat (INTACT FUEL ASSEMBLIES) (Watts)	MPC-68/68FF BWR Assembly Decay Heat (DAMAGED FUEL ASSEMBLIES AND FUEL DEBRIS) (Watts)
≥ 5	1157	1173	1115	898	414	356
≥ 6	1123	1138	1081	873	394	337
≥ 7	1030	1043	1009	805	363	308
≥ 8	1020	1033	993	800	360	305
≥ 9	1010	1023	977	794	358	303
≥ 10	1000	1012	962	789	355	300
≥ 11	996	1008	958	785	353	299
≥ 12	992	1004	954	782	352	297
≥ 13	987	999	949	773	350	296
≥ 14	983	995	945	769	348	294
≥ 15	979	991	941	766	347	293

Notes:
1. Linear interpolation between points is permitted.
2. Includes all sources of heat (i.e., fuel and NON-FUEL HARDWARE).

Table 2.1-6 (page 1 of 2)

FUEL ASSEMBLY COOLING AND MAXIMUM AVERAGE BURNUP
(REGIONALIZED FUEL LOADING)

Post-irradiation Cooling Time (years)	MPC-24 PWR Assembly Burnup for Region 1 (MWD/MTU)	MPC-24 PWR Assembly Burnup for Region 2 (MWD/MTU)	MPC-24E/24EF PWR Assembly Burnup for Region 1 (MWD/MTU)	MPC-24E/24EF PWR Assembly Burnup for Region 2 (MWD/MTU)
≥ 5	49,800	32,200	51,600	32,200
≥ 6	56,100	37,400	58,400	37,400
≥ 7	56,400	41,100	58,500	41,100
≥ 8	58,800	43,800	60,900	43,800
≥ 9	60,400	45,800	62,300	45,800
≥ 10	61,200	47,500	63,300	47,500
≥ 11	62,400	49,000	64,900	49,000
≥ 12	63,700	50,400	65,900	50,400
≥ 13	64,800	51,500	66,800	51,500
≥ 14	65,500	52,500	67,500	52,500
≥ 15	66,200	53,700	68,200	53,700
≥ 16	-	55,000	-	55,000
≥ 17	-	55,900	-	55,900
≥ 18	-	56,800	-	56,800
≥ 19	-	57,800	-	57,800
≥ 20	-	58,800	-	58,800

Note: 1. Linear interpolation between points is permitted.

2. These limits apply to INTACT FUEL ASSEMBLIES, DAMAGED FUEL ASSEMBLIES, and FUEL DEBRIS.

Table 2.1-6 (page 2 of 2)

FUEL ASSEMBLY COOLING AND MAXIMUM AVERAGE BURNUP
(REGIONALIZED FUEL LOADING)

Post-irradiation Cooling Time (years)	MPC-32 PWR Assembly Burnup for Region 1 (MWD/MTU)	MPC-32 PWR Assembly Burnup for Region 2 (MWD/MTU)	MPC-68/68FF BWR Assembly Burnup for Region 1 (MWD/MTU)	MPC-68/68FF BWR Assembly Burnup for Region 2 (MWD/MTU)
≥ 5	39,800	22,100	45,100	26,200
≥ 6	43,400	26,200	47,400	30,500
≥ 7	44,500	29,100	47,400	33,600
≥ 8	46,700	31,200	50,400	35,900
≥ 9	48,400	32,700	52,100	37,600
≥ 10	49,600	34,100	53,900	39,000
≥ 11	50,900	35,200	55,500	40,200
≥ 12	51,900	36,200	56,500	41,200
≥ 13	52,900	37,000	57,500	42,300
≥ 14	53,800	37,800	58,800	43,300
≥ 15	54,700	38,600	59,900	44,200
≥ 16	-	39,400	-	45,000
≥ 17	-	40,200	-	45,900
≥ 18	-	40,800	-	46,700
≥ 19	-	41,500	-	47,500
≥ 20	-	42,200	-	48,500

Note 1. Linear interpolation between points is permitted.

2. These limits apply to INTACT FUEL ASSEMBLIES, DAMAGED FUEL ASSEMBLIES, and FUEL DEBRIS.

Table 2.1-7 (page 1 of 2)

FUEL ASSEMBLY COOLING AND MAXIMUM DECAY HEAT
(REGIONALIZED FUEL LOADING)

Post-irradiation Cooling Time (years)	MPC-24 PWR Assembly Decay Heat for Region 1 (Watts)	MPC-24 PWR Assembly Decay Heat for Region 2 (Watts)	MPC-24E/24EF PWR Assembly Decay Heat for Region 1 (Watts)	MPC-24E/24EF PWR Assembly Decay Heat for Region 2 (Watts)
≥ 5	1470	900	1540	900
≥ 6	1470	900	1540	900
≥ 7	1335	900	1395	900
≥ 8	1,301	900	1360	900
≥ 9	1268	900	1325	900
≥ 10	1235	900	1290	900
≥ 11	1221	900	1275	900
≥ 12	1207	900	1260	900
≥ 13	1193	900	1245	900
≥ 14	1179	900	1230	900
≥ 15	1165	900	1215	900
≥ 16	-	900	-	900
≥ 17	-	900	-	900
≥ 18	-	900	-	900
≥ 19	-	900	-	900
≥ 20	-	900	-	900

Notes:
1. Linear interpolation between points is permitted.
2. Includes all sources of decay heat (i.e., fuel and NON-FUEL HARDWARE).
3. These limits apply to INTACT FUEL ASSEMBLIES, DAMAGED FUEL ASSEMBLIES, and FUEL DEBRIS.

Table 2.1-7 (page 2 of 2)

FUEL ASSEMBLY COOLING AND MAXIMUM DECAY HEAT
(REGIONALIZED FUEL LOADING)

Post-irradiation Cooling Time (years)	MPC-32 PWR Assembly Decay Heat for Region 1 (Watts)	MPC-32 PWR Assembly Decay Heat for Region 2 (Watts)	MPC-68/68FF BWR Assembly Decay Heat for Region 1 (Watts)	MPC-68/68FF BWR Assembly Decay Heat for Region 2 (Watts)
≥ 5	1131	600	500	275
≥ 6	1072	600	468	275
≥ 7	993	600	418	275
≥ 8	978	600	414	275
≥ 9	964	600	410	275
≥ 10	950	600	405	275
≥ 11	943	600	403	275
≥ 12	937	600	400	275
≥ 13	931	600	397	275
≥ 14	924	600	394	275
≥ 15	918	600	391	275
≥ 16	-	600	-	275
≥ 17	-	600	-	275
≥ 18	-	600	-	275
≥ 19	-	600	-	275
≥ 20	-	600	-	275

Notes: 1. Linear interpolation between points is permitted.

2. Includes all sources of decay heat (i.e., fuel and NON-FUEL HARDWARE).

3. These limits apply to INTACT FUEL ASSEMBLIES, DAMAGED FUEL ASSEMBLIES, and FUEL DEBRIS.

Table 2.1-8
NON-FUEL HARDWARE COOLING AND AVERAGE BURNUP

Post-irradiation Cooling Time (years)	BPRA BURNUP (MWD/MTU)	TPD BURNUP (MWD/MTU)	CRA BURNUP (MWD/MTU)	APSR BURNUP (MWD/MTU)
≥ 3	≤ 20,000	NA (Note 3)	NA	NA
≥ 4	≤ 25000	≤ 20,000	NA	NA
≥ 5	≤ 30,000	≤ 25,000	≤ 630,000	≤ 45,000
≥ 6	≤ 40,000	≤ 30,000	-	≤ 54,500
≥ 7	≤ 45,000	≤ 40,000	-	≤ 68,000
≥ 8	≤ 50,000	≤ 45,000	-	≤ 83,000
≥ 9	≤ 60,000	≤ 50,000	-	≤ 111,000
≥ 10	-	≤ 60,000	-	≤ 180,000
≥ 11	-	≤ 75,000	-	≤ 630,000
≥ 12	-	≤ 90,000	-	-
≥ 13	-	≤ 180,000	-	-
≥ 14	-	≤ 630,000	-	-

Notes: 1. Linear interpolation between points is permitted, except that TPD and APSR burnups > 180,000 MWD/MTU and ≤ 630,000 MWD/MTU must be cooled ≥ 14 years and ≥ 11 years, respectively.

2. Applicable to uniform loading and regionalized loading.

3. NA means not authorized for loading.

3.0 DESIGN FEATURES

3.1 Site

3.1.1 Site Location

The HI-STORM 100 Cask System is authorized for general use by 10 CFR Part 50 license holders at various site locations under the provisions of 10 CFR 72, Subpart K.

3.2 Design Features Important for Criticality Control

3.2.1 MPC-24

1. Flux trap size: ≥ 1.09 in.
2. ^{10}B loading in the Boral neutron absorbers: ≥ 0.0267 g/cm^2

3.2.2 MPC-68 and MPC-68FF

1. Fuel cell pitch: ≥ 6.43 in.
2. ^{10}B loading in the Boral neutron absorbers: ≥ 0.0372 g/cm^2

3.2.3 MPC-68F

1. Fuel cell pitch: ≥ 6.43 in.
2. ^{10}B loading in the Boral neutron absorbers: ≥ 0.01 g/cm^2

3.2.4 MPC-24E and MPC-24EF

1. Flux trap size:

   i. Cells 3, 6, 19, and 22: ≥ 0.776 inch

   ii. All Other Cells: ≥ 1.076 inches

2. ^{10}B loading in the Boral neutron absorbers: ≥ 0.0372 g/cm^2

3.2.5 MPC-32

1. Fuel cell pitch: ≥ 9.158 inches
2. ^{10}B loading in the Boral neutron absorbers: ≥ 0.0372 g/cm^2

DESIGN FEATURES

Table 3-1 (page 1 of 5)

LIST OF ASME CODE EXCEPTIONS FOR HI-STORM 100 CASK SYSTEM

Component	Reference ASME Code Section/Article	Code Requirement	Exception, Justification & Compensatory Measures
MPC	NB-1100	Statement of requirements for Code stamping of components.	MPC enclosure vessel is designed and will be fabricated in accordance with ASME Code, Section III, Subsection NB to the maximum practical extent, but Code stamping is not required.
MPC	NB-2000	Requires materials to be supplied by ASME-approved material supplier.	Materials will be supplied by Holtec-approved suppliers with Certified Material Test Reports (CMTRs) in accordance with NB-2000 requirements.
MPC Lid and Closure Ring Welds	NB-4243	Full penetration welds required for Category C Joints (flat head to main shell per NB-3352.3).	MPC lid and closure ring are not full penetration welds. They are welded independently to provide a redundant seal. Additionally, a weld efficiency factor of 0.45 has been applied to the analyses of these welds.
MPC Lid to Shell Weld	NB-5230	Radiographic (RT) or ultrasonic (UT) examination required	Only UT or multi-layer liquid penetrant (PT) examination is permitted. If PT alone is used, at a minimum, it will include the root and final weld layers and each approximately 3/8 inch of weld depth.
MPC Closure Ring, Vent and Drain Cover Plate Welds	NB-5230	Radiographic (RT) or ultrasonic (UT) examination required	Root (if more than one weld pass is required) and final liquid penetrant examination to be performed in accordance with NB-5245. The MPC vent and drain cover plate welds are leak tested. The closure ring provides independent redundant closure for vent and drain cover plates.

(continued)

DESIGN FEATURES (continued)

3.4 Site-Specific Parameters and Analyses

Site-specific parameters and analyses that will require verification by the system user are, as a minimum, as follows:

1. The temperature of 80° F is the maximum average yearly temperature.

2. The allowed temperature extremes, averaged over a 3-day period, shall be greater than -40° F and less than 125° F.

3. a. For free-standing casks, the resultant horizontal acceleration (vectorial sum of two horizontal Zero Period Accelerations (ZPAs) at a three-dimensional seismic site), G_H, and vertical ZPA, G_V, expressed as fractions of 'g', shall satisfy the following inequality:

$$G_H + \mu G_V \leq \mu$$

where μ is the Coulomb friction coefficient for the HI-STORM 100/ISFSI pad interface. Unless demonstrated by appropriate testing that a higher value of μ is appropriate for a specific ISFSI, the value of μ used shall be 0.53. Representative values of G_H and G_V combinations for μ = 0.53 are provided in Table 3-2.

Table 3-2

Representative DBE Acceleration Values to Prevent HI-STORM 100 Sliding (μ = 0.53)

Equivalent Vectorial Sum of Two Horizontal ZPA's (G_H in g's)	**Corresponding Vertical ZPA (G_V in g's)**
0.445	0.160
0.424	0.200
0.397	0.250

(continued)

DESIGN FEATURES

3.4 Site-Specific Parameters and Analyses (continued)

[redacted]

(continued)

DESIGN FEATURES

3.4 Site-Specific Parameters and Analyses (continued)

4. The analyzed flood condition of 15 fps water velocity and a height of 125 feet of water (full submergence of the loaded cask) are not exceeded.

5. The potential for fire and explosion shall be addressed, based on site-specific considerations. This includes the condition that the on-site transporter fuel tank will contain no more than 50 gallons of diesel fuel while handling a loaded OVERPACK or TRANSFER CASK.

6. ISFSI Pad Design

   a. For free-standing casks, the ISFSI pad shall be verified by analysis to limit cask deceleration during a design basis drop and/or non-mechanistic tip-over event to $\leq$ 45 g's at the top of the MPC fuel basket. Analyses shall be performed using methodologies consistent with those described in the HI-STORM FSAR. A lift height above the ISFSI pad is not required to be established if the cask is lifted with a device designed in accordance with ANSI N14.6 and having redundant drop protection features.

[redacted]

(continued)

DESIGN FEATURES

3.4 Site-Specific Parameters and Analyses (continued)

7. In cases where engineered features (i.e., berms and shield walls) are used to ensure that the requirements of 10CFR72.104(a) are met, such features are to be considered important to safety and must be evaluated to determine the applicable Quality Assurance Category.

8. LOADING OPERATIONS, TRANSPORT OPERATIONS, and UNLOADING OPERATIONS shall only be conducted with working area ambient temperatures $\geq 0^{\circ}$ F.

(continued)

INSERT TO ATTACHMENT 4, NON-PROPRIETARY VERSION OF LAR 1014-1, REVISION1

The following Holtec International drawingsand Bills-of-Material submitted with License Amendment Request 1014-1 are Holtec proprietary information:

Drawings 2889 through 2899, all Revision 0
Drawings 3067 through 3075, all Revision 0
Drawing 3187, Revision 1
BM-3065 and 3066, both Revision 0
BM-3189, Revision2

BILL OF MATERIALS FOR 32-ASSEMBLY HI-STAR 100 PWR MPC.(BM-1477)

REF. DWG. 1392 & 1393.

SHEET 1 OF 2

REV. NO.	PREP. BY & DATE	CHECKED BY & DATE	PROJ. MANAGER & DATE	QA. MANAGER & DATE
8	T.C. 6-4-98 INCORPORATED DCR 5014-116	6/6/98	B.G. 6/5/98	V. G. 6-7-98

ITEM NO.	QTY.	MATERIAL	DESCRIPTION	NOMENCLATURE
1A	5	ALLOY "X" SEE NOTE 1.	PLATE 9/32" THK.X 55.59" W. X 176 1/2" LG	BASKET CELL PLATE
1B	2	ALLOY "X" SEE NOTE 1.	PLATE 9/32" THK X 37.15"W. X 176 1/2"LG.	BASKET CELL PLATE
1C	38	ALLOY "X" SEE NOTE 1.	PLATE 9/32" THK X 8.337" (REF) W. X 176 1/2"LG.	BASKET CELL PLATE
3A (3B)	52 (24)	BORAL	.101"THK. X 7.5"W.(7") X 156" LG.PER DET.DWG.1392. SEE NOTE 2.	NEUTRON ABSORBER
4A (4B)	52 (24)	ALLOY "X" SEE NOTE 1.	.075" THK. SHEATHING PER DET. DWG. 1392.	SHEATHING
5A	32		PLATE 3/8"THK X 8.5" SQ.	LOWER FUEL SPACER END PLATE
5B	32		PLATE 3/8"THK X 8.5" SQ.	LOWER FUEL SPACER END PLATE
6	1		1/2" THK X 68 3/8" O.D. X 187 5/8" LG. CYLINDER.	SHELL
7	1		BASEPLATE 2 1/2" THK X 68 3/8" O.D.	BASEPLATE
8A	4		PLATE 5/16"THK.X 14" APPROX. W X 168" LG. PER DET. DWG.1392.	BASKET SUPPORT
8B	8		PLATE 5/16"THK. X 14" APPROX.W X 168" LG. PER DET.DWG. 1392.	BASKET SUPPORT
8C	--		DELETED	----
9A	12		1/2" WIDE X 168" LG. THICKESS AS REQD.	BASKET SUPPORT SHIM
9B	8		2 1/2" WIDE X 168"LG THICKNESS AS REQD.ROLL TO SHELL ID.	BASKET SUPPORT
9C	---		DELETED	-------
9D	----		DELETED	----
9E	AS REQD.		SHIM 1/2" WIDE X 168" LG. THICKNESS AS REQD.	BASKET SUPPORT SHIM
10	4		PLATE 3/4" THK. X 3 1/2" WIDE X 8 3/4" LG.	LIFT LUG
11	4		PLATE 3/4" THK. X 3" WIDE X 4" LG.	LIFT LUG BASEPLATE
12	1		BAR 3 3/4" OD. x 5 7/8" LG.	DRAIN SHIELD BLOCK
13A	2	304 S/S - SURFACE HARDENED	BAR Ø 2 11/16 X 6 3/4" LG. DIMENSION SHOWN ON DWG 1393 SHT4	VENT AND DRAIN TUBE
13B	2	304 S/S - SURFACE HARDENED	BAR Ø 2 1/4 X 2 1/4" LG. DIMENSION SHOWN ON DWG 1393 SHT4	VENT AND DRAIN TUBE CAP
14	1	ALLOY "X" SEE NOTE 1.	PLATE 9 1/2" THK. X 67 1/4" O.D.	MPC LID
15	1	ALLOY "X" SEE NOTE 1.	RING 3/8" THK.X 53" ID. X 67 7/8" O.D.	MPC CLOSURE RING
16	AS REQD.	ALLOY "X" SEE NOTE 1.	SHELL SHIMS SIZES AS REQD.	SHELL SHIM

9

8

BILL OF MATERIALS FOR 32-ASSEMBLY HI-STAR 100 PWR MPC.(BM-1477)

REF. DWG. 1392 & 1393.

REV. NO.	PREP. BY & DATE	CHECKED BY & DATE	PROJ. MANAGER & DATE	QA. MANAGER & DATE
9	S.GEE 7-8-98 INCORPORATED DCR 5014-118	S.G. 7/15/98	7/16/98	U.G. 7-16-98

ITEM NO.	QTY.	MATERIAL	DESCRIPTION	NOMENCLATURE
17	---	---	DELETED	---
18	---	---	DELETED	---
19	2	ALLOY "X" SEE NOTE 1.	PLATE 3/8" THK X 3 7/8" OD.	PORT COVER PLATE
20	32	A-193-B8 OR SIMILAR	3/4"-10UNC X 1 3/8" LG. HEX. BOLT FULL THRD	UPPER FUEL SPACER BOLT
21	---	---	DELETED	---
22	---	---	DELETED	---
23	4	A-193-B8 OR SIMILAR	1 3/4"-5UNC X 2 3/4" LG SOCKET SET SCREW	LID LIFT HOLE PLUG
24	32	ALLOY "X" SEE NOTE 1.	3"-SCH 80 PIPE LGTH AS REQD.	UPPER FUEL SPACER PIPE
25	1 SET	ALLOY "X" SEE NOTE1.	LENGTH, WIDTH AND THICKNESS AS REQ'D.	LID SHIM
26	1	S/S	2" FEMALE X 1 1/4" MALE SCH. 40. S/S REDUCER	REDUCER
27	--	-------	DELETED	-----
28	1	ALLOY "X" SEE NOTE1.	BAR 3 3/4" OD X 5.5" LG.	VENT SHIELD BLOCK
29	4	ALLOY "X" SEE NOTE1.	BAR 3/4" OD. X 1/2" LG.	VENT SHIELD BLOCK SPACER
30	1	ALLOY "X" SEE NOTE1.	2"-SCH 10 PIPE X 173 1/2" APPROX. LG	DRAIN LINE
31	---	---	DELETED	----
32	32	ALLOY "X" SEE NOTE1.	PLATE 3/8" THK X 3 3/4" OD.	UPPER FUEL SPACER END PLATE
33	32	ALLOY "X" SEE NOTE1.	6" SQ. X 1/4" WALL TUBE LGTH AS REQD.	LOWER FUEL SPACER CLUMN
34	AS REQD.	ALUM. 1100	1/8" THK. ALUM SHEET AS REQD.	HEAT CONDUCTION ELEMENTS
35	2	ALUMINUM	0.065" THK X 1.484 OD, .250 HOLE	SEAL WASHER
36	2	S/S	1/4" DIA X 3/8" LG	SEAL WASHER BOLT
37	2	ALLOY "X" SEE NOTE1.	1/8" THK. X 6" X 8" APPROX	DRAIN LINE
38	2	ALLOY "X" SEE NOTE1.	1/8" THK X 4" OD. X 2 3/8" OD. X 4" HIGT CONE	DRAIN LINE

NOTES. (FOR SHEET 1 & 2)

1. ALLOY X IS ANY OF THE FOLLOWING ACCEPTABLE STAINLESS STEEL ALLOYS: ASME TYPE 316, 316LN, 304, 304LN. THE ALLOY TO BE USED SHALL BE SPECIFIED BY THE LICENSEE.
2. MINIMUM BORAL B-10 LOADING IS 0.0372 g/cm^2 BORAL TO BE PASSIVATED PRIOR TO INSTALLATION.
3. ALL DIMENSIONS ARE APPROXIMATE DIMENSIONS.
4. ITEMS 8A, 8B, 9A, 9B, 9E, 16 AND 34 MAY BE MADE FROM MORE THAN ONE PIECE. THE ENDS OF THE PIECES DO NOT NEED TO BE WELDED TOGETHER BUT THEY MUST BE FLUSH WITH EACH OTHER WHEN INSTALLED.

BILL OF MATERIALS FOR 24-ASSEMBLY HI-STAR 100 PWR MPC.(BM-1478)

REF. DWG. 1395 & 1396.

REV.NO.	PREP. BY & DATE	CHECKED BY & DATE	PROJ. MANAGER & DATE	QA. MANAGER & DATE
10	S.GEE 11-3-99 REVISED AS INDICATED	11/22/99	B.G. 11/23/99	MC 11/22/99

ITEM NO.	QTY.	MATERIAL	DESCRIPTION	NOMENCLATURE
1A	2	ALLOY "X" SEE NOTE 1.	PLATE 5/16" THK.X 63.20" REF W. X 176 1/2" LG	BASKET CELL PLATE
1B	1		PLATE 5/16" THK X 60.57" REF W. X 176 1/2"LG.	BASKET CELL PLATE
1C	2		PLATE 5/16" THK X 43.42" REF W. X 176 1/2"LG.	BASKET CELL PLATE
1D	1		PLATE 5/16" THK X 20.402 " REF W. X 176 1/2"LG.	BASKET CELL PLATE
1E	1		PLATE 5/16" THK X 7.7175" REF W. X 176 1/2"LG.	BASKET CELL PLATE
1F	22		PLATE 5/16" THK X10.4625 " REF W. X 176 1/2"LG.	BASKET CELL PLATE
1G	1		PLATE 5/16" THK X 3.7445 " REF W. X 176 1/2"LG.	BASKET CELL PLATE
1H	2		PLATE 5/16" THK X 9.03 " REF W. X 176 1/2"LG.	BASKET CELL PLATE
2	24		PIPE 3"-SCH 80 LGTH AS REQD.	UPPER FUEL SPACER PIPE
3A (3B)	84(12)	BORAL	.075"THK. X 7.5"W.X6 1/4" X 156" LG.PER DET.DWG.1395. SEE NOTE 2.	NEUTRON ABSORBER
4A (4B)	84(12)	ALLOY "X" SEE NOTE 1.	.06" THK. SHEATHING PER DET. DWG. 1395.	SHEATHING
5A	4		PLATE 5/16"THK X 3" W. X 176 1/2"LG.	BASKET CELL PLATE
5B	4		PLATE 5/16"THK X 3 3/4" APPROX. W. X 176 1/2"LG.	BASKET CELL PLATE
5C	4		PLATE 1.5" APP THK. X 3" W. X 168"LG.	BASKET SUPPORT
5D	4		2 1/2" W X 168" LG	BASKET SUPPORT
5E	4		2 "WIDE X 168"LG. THICKNESS AS REQD.	BASKET SUPPORT
5F	-		DELETED	---
5G	4		1 1/4" W X 1" THK X 168" LG.	BASKET SUPPORT SHIM
5H	---		DELETED	-----
6	1		1/2" THK X 68 3/8" O.D. X 167 5/8" LG. CYLINDER.	SHELL
7	1		BASEPLATE 2 1/2" THK X 68 3/8" O.D.	BASEPLATE
8A	22		9/32"THK. ANGLE X 176 1/2" LG. FROM PLATE PER DET. DWG.1395.	BASKET CELL ANGLE
8B	2		9/32"THK. CHANNEL X 176 1/2" LG. FROM PLATE PER DET.DWG. 1395.	BASKET CELL CHANNEL
9A	1		5/16" THK.X 10" W. APP. X 168" LG. PER DET.	BASKET SUPPORT
9B	2		5/16"THK. X 7 1/2" APP. W. X 168" LG. PER DET.	BASKET SUPPORT
9C	1		5/16"THK. X 5" APP. W. X 168" LG. PER DET.	BASKET SUPPORT
9D	AS REQD		AS REQUIRED	BASKET SUPPORT
9E	----		DELETED	---
9F	---		DELETED	---
9G	---		DELETED	---
9H	---		DELETED	---
10	4		PLATE 3/4" THK. X 3 1/2" WIDE X 6 3/4" LG.	LIFT LUG

BILL OF MATERIALS FOR 24-ASSEMBLY HI-STAR 100 PWR MPC.(BM-1478)

EID #3098

REF. DWG. 1395 & 1396.

REV.NO.	PREP. BY & DATE	CHECKED BY & DATE	PROJ. MANAGER & DATE	QA. MANAGER & DATE
13	J.A. 4-11-00 ECO 1021-7 & 8 ECO 1022-3	B.G. 4/14/00	4/14/00	MPh 4/14/00

ITEM NO.	QTY.	MATERIAL	DESCRIPTION	NOMENCLATURE
11	4	ALLOY "X" SEE NOTE 1.	PLATE 3/4" THK. X 4" WIDE X 3" LG.	LIFT LUG BASEPLATE
12	1	ALLOY "X" SEE NOTE 1.	BAR 3.75" OD. X 5 7/8" LG.	DRAIN SHIELD BLOCK
13A	2	304 S/S	BAR 2 11/16" OD X 6.75" LG, DIMENSIONS AS SHOWN ON DWG 1396 SH 4	VENT AND DRAIN TUBE
13B	2	304 S/S	BAR 2 1/4 OD X 2 1/4 LG, DIMENSIONS AS SHOWN ON DWG 1396 SH 4	VENT AND DRAIN TUBE CAP
14	1	ALLOY "X" SEE NOTE 1.	9 1/2" THK. X 67 1/4" O.D.	MPC LID
15	1	↓	RING 3/8" THK.X 53 1/4" ID. X 67 5/8" O.D.	MPC CLOSURE RING
16	1	S/S	2-1/2" SCH. 10 PIPE, 158" LG WITH FUNNEL	DRAIN GUIDE TUBE
17	—	—	DELETED	—
18	AS REQD	ALLOY "X" SEE NOTE 1.	AS REQUIRED	BASKET SUPPORT
19	2	ALLOY "X" SEE NOTE 1.	PLATE 3/8" THK. X 3 7/8"OD.	PORT COVER PLATE
20	24	A-193-B8 OR SIMILAR	3/4"-10UNC X 1 1/4"LG. HEX BOLT WITH FULL THRD.	UPPER FUEL SPACER BOLT
21	AS REQD	ALLOY "X" SEE NOTE 1.	3/4" X 2" X THICKNESS AS REQUIRED	LIFT LUG SHIM
22	---	---	DELETED	---
23	4	A-193-B8 OR SIMILAR	1 3/4"-5UNC X 2 3/4"LG SOCKET SET SCREW	LID LIFT HOLE PLUG
24	24	ALLOY "X" SEE NOTE 1.	PLATE 3/8" THK X 4" OD.	UPPER FUEL SPACER END PLATE
25	1 SET	ALLOY "X" SEE NOTE 1.	LENGTH, WIDTH AND THICKNESS OF SHIMS AS REQUIRED.	LID SHIM
26	1	S/S	COUPLING	COUPLING
27	AS REQD	ALLOY "X" SEE NOTE 1.	3/4" X 5" DIAM.	UPPER FUEL SPACER END PLATE
28	1	ALLOY "X" SEE NOTE 1.	BAR 3.75" OD. X 5.5"LG.	VENT QUICK DISCONN. CPLG.
29	4	ALLOY "X" SEE NOTE 1.	BAR 3/4" OD. X 1/2" LG.	VENT SHIELD BLOCK SPACER
30	1	ALLOY "X" SEE NOTE 1.	2"-SCH 10 PIPE X 173 1/2"APPROX. LG.	DRAIN LINE
31	4	S/S	SOCKET SET SCREW 1/4-20 1/4" LG	COVER PLATE PLUG
32	24	S/S SEE NOTE 5	6" SQ. TUBING X 1/4" WALL LENGTH AS REQ'D.	LOWER FUEL SPACER COLUMN
33A	24	ALLOY "X" SEE NOTE 1.	PLATE 3/8" THK X 8.5" SQ.	LOWER FUEL SPACER END PLATE
33B	24	———	PLATE 3/8" THK X 8.5" SQ.	LOWER FUEL SPACER END PLATE
34	—	———	DELETED	———
35	AS REQ'D.	ALUM. ALLOY 1100 & S/S	1/8" THICK X 176 1/2" LG. ALUM. SHEET (153" LG (APP.) AT DRAIN PIPE LOCATION) WITH S/S SPRINGS	HEAT CONDUCTION ELEMENTS
36	2	ALUMINUM	0.065" THK 1.494 OD, 0.250" HOLE	SEAL WASHER
37	2	S/S	1/4" DIA X 3/8" LG	SEAL WASHER BOLT
38	2	ALLOY "X" SEE NOTE 1.	1/8" X 10 1/2" X 9 1/2" SHEET	DRAINLINE
39	—	---	DELETED	---

NOTES: (FOR SHEET 1 & 2)

1. ALLOY X IS ANY OF THE FOLLOWING ACCEPTABLE STAINLESS STEEL ALLOYS: ASME TYPE 316, 316LN, 304, 304LN. THE ALLOY TO BE USED SHALL BE SPECIFIED BY THE LICENSEE.
2. MINIMUM BORAL B-10 LOADING IS 0.0267 g/cm^2. BORAL TO BE PASSIVATED PRIOR TO INSTALLATION.
3. ALL DIMENSIONS ARE APPROXIMATE DIMENSIONS.
4. ITEMS 5C,5D,5E,5G 9A,9B,9C,9D,16,18, AND 35 MAY BE MADE FROM MORE THAN ONE PIECE. THE ENDS OF PIECES DO NOT NEED TO BE WELDED TOGETHER BUT THEY MUST BE FLUSH WITH EACH OTHER WHEN INSTALLED.
5. MUST BE TYPE 304, 304LN, 316, OR 316LN WITH TENSILE STRENGTH>75ksi, YIELD STRENGTH>30ksi, AND CHEMICALS PER ASTM A554.

BILL OF MATERIALS FOR 68-ASSEMBLY HI-STAR 100 BWR MPC.(BM-1479)

E.I.D. #3099
SHEET 1 OF 2

REF. DWGS. 1401 & 1402.

REV. NO.	PREP. BY & DATE	CHECKED BY DATE	PROJ. MANAGER & DATE	QA. MANAGER & DATE
12	J.A. 4-11-00 ECO 1021-4	B.G. 4/14/00	4/14/00	4/14/00

ITEM NO.	QTY.	MATERIAL	DESCRIPTION	NOMENCLATURE
1A	3	ALLOY "X" SEE NOTE 1.	PLATE 1/4" THK. X 65.65"W. X 176" LG PER DET. DWG.1401.	BASKET CELL PLATE
1B	4		PLATE 1/4" THK. X 52.67"W. X 176" LG PER DET. DWG.1401.	BASKET CELL PLATE
1C	2		PLATE 1/4" THK. X 39.69W. X 176" LG PER DET. DWG.1401.	BASKET CELL PLATE
1D	2		PLATE 1/4" THK. X 13.73"W. X 176" LG PER DET. DWG.1401.	BASKET CELL PLATE
1E	78		PLATE 1/4" THK. X 6.24"W. X 176" LG PER DET. DWG.1401.	BASKET CELL PLATE
2	68		3"- SCH 80 PIPE LGTH AS REQD.	UPPER FUEL SPCER CLUMN
3A	116	BORAL	.101"THK. X 4 3/4"W. X 156" LG.PER DET.DWG.1401. SEE NOTE 2.	NEUTRON ABSORBER
4A	116	ALLOY "X" SEE NOTE 1.	.075" THK. SHEATHING PER DET. DWG.1401.	SHEATHING
5	8		BAR 1" WIDE X 168" LG X THICKNESS AS REQUIRED	BASKET SUPPORT SHIM
6	1		1/2" THK X 68 3/8" O.D. X 187 5/8" LG. CYLINDER.	SHELL
7	1		BASEPLATE 2 1/2" THK X 68 3/8" O.D.	BASEPLATE
8	8		PLATE 5/16"THK. X 10" APPROX.W.X 168 1/2" LG. PER DET. DWG.1401.	BASKET SUPPORT
9A	4		BAR 1" W. X .8" APPROX.THK. X 168 1/2" LG.	BASKET SUPPORT
9B	---		DELETED	---
9C	8		2 1/2" WIDE X 168 1/2" LG. THICKNESS AS REQD. ROLL TO SHELL I.D.	BASKET SUPPORT
9D	AS REQD		AS REQUIRED	BASKET SUPPORT
10	4		PLATE 3/4" THK. X 3 1/2" WIDE X 8 3/4" LG.	LIFT LUG
11	4		PLATE 3/4" THK. X 2 1/2" WIDE X 4" LG.	LIFT LUG BASEPLATE
12	1		BAR 3.75"OD. X 5 7/8" LG.	DRAIN SHIELD BLOCK
13A	2	304 S/S	BAR 2 11/16" OD X 6 3/4" REF LG, DIMENSION ON DWG 1402 SHT 4	VENT AND DRAIN TUBE
13B	2	304 S/S	BAR 2 1/4" OD X 2 1/4" REF LG, DIMENSION ON DWG 1402 SHT 4	VENT AND DRAIN TUBE CAP
14	1	ALLOY "X" SEE NOTE 1.	10" THK. X 67 1/4" O.D. [MPC-68] 10" THK. X 66 1/4" O.D. [MPC-68F]	MPC LID
15	1	ALLOY "X" SEE NOTE 1.	RING 3/8" THK. X 53 1/4" ID. X 67 5/8" O.D. [MPC-68] RING 3/8" THK. X 53 1/4" ID. X 67 1/8" O.D. [MPC-68F]	MPC CLOSURE RING
16	1	S/S	2-1/2"-SCH 10 PIPE 158" LG WITH FUNNEL	DRAIN GUIDE TUBE

12

\DRAWINGS\5014\5014\MPC\BM1479-1.R12

BILL OF MATERIALS FOR 68-ASSEMBLY HI-STAR 100 BWR MPC.(BM-1479) (E.I.D. 3083)

REF. DWGS. 1401 & 1402.

REV. NO.	PREP. BY & DATE	CHECKED BY DATE	PROJ. MANAGER & DATE	QA. MANAGER & DATE
15	J.A. 4-11-00 INCORPORATED ECO-1021-3, 7, & 8	B.G. 4/13/00	4/14/00	4/14/00

ITEM NO.	QTY.	MATERIAL	DESCRIPTION	NOMENCLATURE
17	1	ALLOY "X" SEE NOTE 1.	1" THK X 68 3/8" OD X 11 5/8" LG. CYLINDER [MPC-68F]	SHELL
18	8	ALLOY "X" SEE NOTE 1.	3/8" THK FEMALE SUPPORT SHIM PER DETAIL, DWG. 1401, SHT 4.	BASKET SUPPORT SHIM
19	2	ALLOY "X" SEE NOTE 1.	PLATE 3/8" THK X 3 7/8" OD.	PORT COVER PLATE
20	68	A-193-B8 OR SIMILAR	3/4"-10UNC X 1.375"LG. FULL THRD. HEX. BOLT	UPPER FUEL SPACER BOLT
21	AS REQD	ALLOY "X" SEE NOTE 1.	3/4" W X 2' LG X THICKNESS AS REQUIRED	LIFT LEG SHIM
22	---	---	DELETED	---
23	4	A-193-B8 OR SIMILAR	1 3/4"-5UNC X 2 3/4" LG. SOCKET SET SCREW.	LIFT HOLE PLUG
24	68	ALLOY "X" SEE NOTE 1.	PLATE 3/8" THK X 4" OD.	UPPER FUEL SPACER END PLATE
25	1 SET	ALLOY "X" SEE NOTE 1	LENGTH, WIDTH , THICKNESS AND QUANTITY AS REQD.	LID SHIM
26	1	S/S	2" FEMALE X 1 1/4" MALE SCH. 40, S/S COUPLING	COUPLING
27			DELETED	
28	1	ALLOY "X" SEE NOTE 1.	BAR 3.75" OD. X 5.5" LG.	VENT SHIELD BLOCK
29	4	ALLOY "X" SEE NOTE 1.	BAR .75"OD X .5"LG.	VENT SHIELD BLOCK SPACER
30	1	ALLOY "X" SEE NOTE 1.	2"-SCH 10 PIPE X 173" APPROX.LG.	DRAIN LINE
15 31	4	S/S	SOCKET SET SCREW 1/4-20 1/4" LG	COVER PLATE PLUG
32	--		DELETED	---
15 33	68	S/S SEE NOTE 5	4" SQ. TUBE X 1/4" WALL LENGTH AS REQD. (FOR SHORT FUEL ONLY)	LOWER FUEL SPACER COLUMN
34A	68	ALLOY "X" SEE NOTE 1.	3/8" THK. X 5 3/4" SQ. PLATE (FOR SHORT FUEL ONLY)	LOWER FUEL SPACER END PLATE
34B	68	ALLOY "X" SEE NOTE 1.	3/8" THK. X 5 3/4" SQ. PLATE (FOR SHORT FUEL ONLY)	LOWER FUEL SPACER END PLATE
35			DELETED	
36	----	------	DELETED	
37	AS REQD	ALUM. ALLOY 1100	1/8" THK. X 176" LG. ALUM. SHEET.(153" LG APP. AT DRAIN PIPE LOCATION.) W/S/S SPRINGS.	HEAT CONDUCTION ELEMENTS
38	2	ALUMINUM	.065" THK X 1.494 OD, .250 HOLE	SEAL WASHER
39	2	S/S	1/4" DIA X 3/8 LG	SEAL WASHER BOLT
40	2	ALLOY "X" SEE NOTE 1	1/8" THK. 6" X 6" APPROX. SHEET	DRAIN LINE
41	----	-----------	DELETED	-----------

NOTES: (FOR SHEET 1 & 2)

1. ALLOY X IS ANY OF THE FOLLOWING ACCEPTABLE STAINLESS STEEL ALLOYS: ASME TYPE 316, 316LN, 304, 304LN. THE ALLOY TO BE USED SHALL BE SPECIFIED BY THE LICENSEE.
2. FOR MPC-68 AND MPC-68FF, MINIMUM BORAL B-10 LOADING IS 0.0372 g/cm_2. FOR MPC-68F, MINIMUM BORAL B-10 LOADING IS 0.01 g/cm_2. BORAL TO BE PASSIVATED PRIOR TO INSTALLATION. 15
3. ALL DIMENSIONS ARE APPROXIMATE DIMENSIONS.
4. ITEMS 5,8,9A,9B,9C,16,18,36 AND 37 MAY BE MADE FROM MORE THAN ONE PIECE. THE ENDS OF PIECES DO NOT NEED TO BE WELDED TOGETHER BUT THEY MUST BE FLUSH WITH EACH OTHER WHEN INSTALLED.
5. 15 MUST BE TYPE 304, 304LN, 316, OR 316LN WITH TENSILE STRENGTH>75ksi, YIELD STRENGTH>30ksi, AND CHEMICALS PER ASTM A554.

\DRAWINGS\5014\5014\MPC\BM1479-2.R15

(BM-2898) BILL OF MATERIALS FOR 24-ASSEMBLY HI-STAR 100 PWR MPC-24E (SHEET 1).

REF. DWG. 2889 TO 2892, 1395 SHT 3 & 1396 SHT 1 TO SHT 5

REV.NO.	PREP. BY & DATE	CHECKED BY & DATE	PROJ. MANAGER & DATE	QA. MANAGER & DATE
1	J.D.A. 8-16-2000 INCORPORATED CHANGES	E Rosenbaum E Rosenbaum 8/24/2000	Brian Gutherman 8/24/00	MPhipps 8/24/00

ITEM NO.	QTY.	MATERIAL	DESCRIPTION	NOMENCLATURE
1A	2	ALLOY "X" SEE NOTE 1.	PLATE 5/16" THK.X 64.543" REF W. X 176 1/2" LG	BASKET CELL PLATE
1B	4		PLATE 5/16" THK X 23.165" REF W. X 176 1/2"LG.	BASKET CELL PLATE
1C	2		PLATE 5/16" THK X 45.985" REF W. X 176 1/2"LG.	BASKET CELL PLATE
1D	4		5/16 ANGLE X 10.847 X 10.847 X 176 1/2" LG	BASKET CELL ANGLE
1E	4		PLATE 5/16" THK X 9.5" REF W. X 176 1/2"LG.	BASKET CELL PLATE
1F	16		PLATE 5/16" THK X 10.535 " REF W. X 176 1/2"LG.	BASKET CELL PLATE
2	24		PIPE 3"-SCH 80 LGTH AS REQD.	UPPER FUEL SPACER PIPE
3A (3B)	72(24)	BORAL	.101"THK. X 7.5"W.(6 1/4") X 156" LG.PER DET.DWG.1395 SHT 3. SEE NOTE 2	NEUTRON ABSORBER
4A (4B)	72(24)	ALLOY "X" SEE NOTE 1.	.06" THK. SHEATHING PER DET. DWG. 1395.	SHEATHING
5A	4		PLATE 5/16" THK X 3" W. X 176 1/2" LG.	BASKET CELL PLATE
5B	8		PLATE 2" REF WIDE X 168" LG X THICKNESS AS REQD	BASKET SUPPORT
5C	4		PLATE 3" REF W. X 168"LG. X THICKNESS AS REQD	BASKET SUPPORT
5D	4		PLATE 5/16"THK X 1.472" APPROX W. X 176 1/2"LG.	BASKET CELL PLATE
6	1		1/2" THK X 68 3/8" O.D. X 187 5/8" LG. CYLINDER.(MPC-24;MPC-24E) 1/2" THK X 68 3/8" O.D. X 176" LG. CYLINDER.(MPC-24EF)	SHELL
7	1		BASEPLATE 2 1/2" THK X 68 3/8" O.D.	BASEPLATE
8A	12		5/16"THK. ANGLE X 176 1/2" LG. FROM PLATE PER DET. DWG.1395 SHT 3	BASKET CELL ANGLE
8B	8		5/16"THK. CHANNEL X 176 1/2" LG. FROM PLATE PER DET.DWG. 1395 SHT 3.	BASKET CELL CHANNEL
8C	4		5/16"THK. ANGLE X 176 1/2" LG. FROM PLATE PER DET.DWG. 1395 SHT 3	BASKET CELL CHANNEL
9	4		PLATE 1.25" APP. THK. X 2" W. X 168"LG.	BASKET SUPPORT
10	4		PLATE 3/4" THK. X 3 1/2" WIDE X 8 3/4" LG.	LIFT LUG

\DRAWINGS\5014\5014\MPC\BM2898R0

(BM2899) BILL OF MATERIALS FOR 24-ASSEMBLY HI-STAR 100 PWR MPC-24E.(SHEET 2)

REF. DWG. 2889 TO 2892,1395 SHT 3 & 1396 SHT 1 TO SHT 5.

REV.NO.	PREP. BY & DATE	CHECKED BY & DATE	PROJ. MANAGER & DATE	QA. MANAGER & DATE
1	J.D.A. 8-16-00 INCORPORATED CHANGES	E Rosenbaum E Rosenbaum 8/24/2000	Brian Gutherman 8/24/00	M Phillips MP 8/24/00

ITEM NO.	QTY.	MATERIAL	DESCRIPTION	NOMENCLATURE
11	4	ALLOY "X" SEE NOTE 1.	PLATE 3/4" THK. X 4" WIDE X 3" LG.	LIFT LUG BASEPLATE
12	1	ALLOY "X" SEE NOTE 1.	BAR 3.75" OD. X 5 7/8" LG.	DRAIN SHIELD BLOCK
13A	2	304 S/S	BAR 2 11/16" OD X 6.75" LG, DIMENSIONS AS SHOWN ON DWG 1396 SH 4	VENT AND DRAIN TUBE
13B	2	304 S/S	BAR 2 1/4 OD X 2 1/4 LG, DIMENSIONS AS SHOWN ON DWG 1396 SH 4	VENT AND DRAIN TUBE CAP
14	1	ALLOY "X" SEE NOTE 1.	9 1/2" THK. X 67 1/4" O.D.	MPC LID
15	1	↓	RING 3/8" THK.X 53 1/4" ID. X 67 5/8" O.D.	MPC CLOSURE RING
16	1	S/S	2 1/2" SCH 10S PIPE, 158" LG WITH FUNNEL.	DRAIN GUIDE TUBE
17	1	ALLOY "X" SEE NOTE 1.	1" THK. x 68-3/8 O.D. x 11 5/8" LG. CYLINDER (MPC-24EF)	SHELL
18	AS REQD	ALLOY "X" SEE NOTE 1.	AS REQUIRED	BASKET SUPPORT
19	2	ALLOY "X" SEE NOTE 1.	PLATE 3/8" THK. X 3 7/8"OD.	PORT COVER PLATE
20	AS REQD	A-193-B8 OR SIMILAR	3/4"-10UNC X 1 1/4"LG. HEX BOLT WITH FULL THRD.	UPPER FUEL SPACER BOLT
21	AS REQD	ALLOY "X" SEE NOTE 1.	3/4" X 2" X THICKNESS AS REQUIRED	LIFT LUG SHIM
22	---	---	DELETED	---
23	4	A-193-B8 OR SIMILAR	1 3/4"-5UNC X 2 3/4"LG SOCKET SET SCREW	LID LIFT HOLE PLUG
24	AS REQD	ALLOY "X" SEE NOTE 1.	PLATE 3/4" THK X 5" OD.	UPPER FUEL SPACER END PLATE
25	1 SET	ALLOY "X" SEE NOTE 1.	LENGTH, WIDTH AND THICKNESS OF SHIMS AS REQUIRED.	LID SHIM
26	1	S/S	COUPLING	COUPLING
27	AS REQD	ALLOY "X" SEE NOTE 1.	3/4" X 5" DIAM.	UPPER FUEL SPACER END PLATE
28	1	ALLOY "X" SEE NOTE 1.	BAR 3.75" OD. X 5.5"LG.	VENT QUICK DISCONN. CPLG.
29	4	ALLOY "X" SEE NOTE 1.	BAR 3/4" OD. X 1/2" LG.	VENT SHIELD BLOCK SPACER
30	1	ALLOY "X" SEE NOTE 1.	2"-SCH 10 PIPE X 173 1/2"APPROX. LG.	DRAIN LINE
31	--	------	DELETED	----
32	AS REQD	│	6" SQ. TUBING X 1/4" WALL LENGTH AS REQ'D.	LOWER FUEL SPACER COLUMN
33A	AS REQD	│	PLATE 3/8" THK X 8.5" SQ.	LOWER FUEL SPCAER END PLATE
33B	AS REQD	↓	PLATE 3/8" THK X 8.5" SQ.	LOWER FUEL SPCAER END PLATE
34	---	------	DELETED	------
35	AS REQ'D.	ALUM. ALLOY 1100 & S/S	1/8" THICK X 176 1/2" LG. ALUM. SHEET (153" LG (APP.) AT DRAIN PIPE LOCATION) WITH S/S SPRINGS	HEAT CONDUCTION ELEMENTS
36	2	ALUMINUM	0.065" THK 1.494 OD, 0.250" HOLE	SEAL WASHER
37	2	S/S	1/4" DIA X 3/8" LG	SEAL WASHER BOLT
38	2	ALLOY "X" SEE NOTE 1.	1/8" X 10 1/2" X 9 1/2" SHEET	DRAINLINE
39	8	ALLOY "X" SEE NOTE 1.	1/8" X 4" X 4 1/2" APPROX SHEET	DRAINLINE
40	1	ALLOY "X" SEE NOTE 1.	2.722" SQ APPROX X 176 1/2" LG	CENTER COLUMN

NOTES: (FOR SHEET 1 & 2) THE ALLOY TO BE USED SHALL BE SPECIFIED BY THE LICENSEE.

1. ALLOY X IS ANY OF THE FOLLOWING ACCEPTABLE STAINLESS STEEL ALLOYS: ASME TYPE 316, 316LN, 304, 304LN.
2. MINIMUM BORAL B-10 LOADING IS 0.0267 g/cm^2. BORAL TO BE PASSIVATED PRIOR TO INSTALLATION.
3. ALL DIMENSIONS ARE APPROXIMATE DIMENSIONS.
4. ITEMS 5C,5D, 9,18, AND 35 MAY BE MADE FROM MORE THAN ONE PIECE. THE ENDS OF PIECES DO NOT NEED TO BE WELDED TOGETHER BUT THEY MUST BE FLUSH WITH EACH OTHER WHEN INSTALLED.

\DRAWINGS\5014\5014\MPC\BM2899R0

FIGURE WITHHELD AS SENSITIVE UNCLASSIFIED INFORMATION

REVISION	
HOLTEC INTERNATIONAL	EQUIPMENT DESIGN ANALYSIS CONSULTING
DESCRIPTION HI-STAR 100 MPC-32 CONSTRUCTION	
CLIENT N/A	
COMPANION DRAWINGS 1393	REV. 10
PROJECT No. 5014	DRAWING No. 1392 SHT 1 OF 4 (E.I.D. 2840)
P.O. No. N/A	

A

FIGURE WITHHELD AS SENSITIVE UNCLASSIFIED INFORMATION

REVISION	
HOLTEC INTERNATIONAL	EQUIPMENT DESIGN ANALYSIS CONSULTING
DESCRIPTION HI-STAR 100 MPC-32 CONSTRUCTION	
CLIENT N/A	
COMPANION DRAWINGS 1393	REV. 10
PROJECT No. 5014	DRAWING No. 1392 SHT 2 OF 4 (E.I.D. 2841)
P.O. No. N/A	

A

FIGURE WITHHELD AS SENSITIVE UNCLASSIFIED INFORMATION

REVISION	
HOLTEC INTERNATIONAL	EQUIPMENT DESIGN ANALYSIS CONSULTING
DESCRIPTION HI-STAR 100 MPC-32 CONSTRUCTION	
CLIENT N/A	
COMPANION DRAWINGS 1393	REV. 10
PROJECT No. 5014 P.O. No. N/A	DRAWING No. 1392 SHT 3 OF 4 (E.I.D. 2842)

A

FIGURE WITHHELD AS SENSITIVE UNCLASSIFIED INFORMATION

REVISION	
HOLTEC INTERNATIONAL	EQUIPMENT DESIGN ANALYSIS CONSULTING
DESCRIPTION HI-STAR 100 MPC-32 CONSTRUCTION	
CLIENT N/A	
COMPANION DRAWINGS 1393	REV. 9
PROJECT No. 5014	DRAWING No. 1392 SHT 4 OF 4 (E.I.D. 2843)
P.O. No. N/A	

FIGURE WITHHELD AS SENSITIVE UNCLASSIFIED INFORMATION

BY:	BY:	
REVISION		
HOLTEC INTERNATIONAL	EQUIPMENT DESIGN ANALYSIS CONSULTING	
DESCRIPTION HI-STAR 100 MPC-32 CONSTRUCTION		
CLIENT N/A		
COMPANION DRAWINGS 1392		REV. 11
PROJECT No. 5014	DRAWING No. 1393 SHT 1 OF 6 (E.I.D. 2844)	
P.O. No. N/A		

FIGURE WITHHELD AS SENSITIVE UNCLASSIFIED INFORMATION

REVISION	
HOLTEC INTERNATIONAL	EQUIPMENT DESIGN ANALYSIS CONSULTING
DESCRIPTION HI-STAR 100 MPC-32 CONSTRUCTION	
CLIENT N/A	
COMPANION DRAWINGS 1392	REV. 10
PROJECT No. 5014	DRAWING No. 1393 SHT 2 OF 6 (E.I.D. 2845)
P.O. No. N/A	

A

FIGURE WITHHELD AS SENSITIVE UNCLASSIFIED INFORMATION

REVISION	
HOLTEC INTERNATIONAL	EQUIPMENT DESIGN ANALYSIS CONSULTING
DESCRIPTION HI-STAR 100 MPC-32 CONSTRUCTION	
CLIENT N/A	
COMPANION DRAWINGS 1392	REV. 10
PROJECT No. 5014	DRAWING No. 1393 SHT 3 OF 6
P.O. No. N/A	(E.I.D. 2846)

FIGURE WITHHELD AS SENSITIVE UNCLASSIFIED INFORMATION

REVISION	
HOLTEC INTERNATIONAL	EQUIPMENT DESIGN ANALYSIS CONSULTING
DESCRIPTION HI-STAR 100 MPC-32 CONSTRUCTION	
CLIENT N/A	
COMPANION DRAWINGS 1392	REV. 9
PROJECT No. 5014 P.O. No. N/A	DRAWING No. 1393 SHT 4 OF 6 (E.I.D. 2847)

A

FIGURE WITHHELD AS SENSITIVE UNCLASSIFIED INFORMATION

REVISION	
HOLTEC INTERNATIONAL	EQUIPMENT DESIGN ANALYSIS CONSULTING
DESCRIPTION HI-STAR 100 MPC-32 CONSTRUCTION	
CLIENT N/A	
COMPANION DRAWINGS 1392	REV. 8
PROJECT No. 5014	DRAWING No.
P.O. No. N/A	1393 SHT 5 OF 6 EID #3095

A

FIGURE WITHHELD AS SENSITIVE UNCLASSIFIED INFORMATION

REVISION

HOLTEC INTERNATIONAL

EQUIPMENT DESIGN
ANALYSIS
CONSULTING

DESCRIPTION
HI-STAR 100 MPC-32 CONSTRUCTION

CLIENT N/A

COMPANION DRAWINGS 1392	REV. 8
PROJECT No. 5014	DRAWING No. 1393 SHT 6 OF 6 (E.I.D. 2848)
P.O. No. N/A	

FIGURE WITHHELD AS SENSITIVE UNCLASSIFIED INFORMATION

REVISION		
HOLTEC INTERNATIONAL	EQUIPMENT DESIGN ANALYSIS CONSULTING	
DESCRIPTION HI-STAR 100 MPC-24 CONSTRUCTION		
CLIENT N/A		
COMPANION DRAWINGS 1396		REV. 12
PROJECT No. 5014	DRAWING No. 1395 SHT 1 OF 4 (E.I.D. 2853)	
P.O. No. N/A		

A

FIGURE WITHHELD AS SENSITIVE UNCLASSIFIED INFORMATION

REVISION	
HOLTEC INTERNATIONAL	EQUIPMENT DESIGN ANALYSIS CONSULTING
DESCRIPTION HI-STAR 100 MPC-24 CONSTRUCTION	
CLIENT N/A	
COMPANION DRAWINGS 1396	REV. 10
PROJECT No. 5014 P.O. No. N/A	DRAWING No. 1395 SHT 2 OF 4

A

FIGURE WITHHELD AS SENSITIVE UNCLASSIFIED INFORMATION

REVISION		
HOLTEC INTERNATIONAL	EQUIPMENT DESIGN ANALYSIS CONSULTING	
DESCRIPTION HI-STAR 100 MPC-24 CONSTRUCTION		
CLIENT N/A		
COMPANION DRAWINGS 1396, 2889 TO 2892		REV. 11
PROJECT No. 5014	DRAWING No. 1395 SHT 3 OF 4 (E.I.D. 2893)	
P.O. No. N/A		

FIGURE WITHHELD AS SENSITIVE UNCLASSIFIED INFORMATION

BY:	BY:
REVISION	
HOLTEC INTERNATIONAL	EQUIPMENT DESIGN ANALYSIS CONSULTING
DESCRIPTION HI-STAR 100 MPC-24 CONSTRUCTION	
CLIENT N/A	
COMPANION DRAWINGS 1396	REV. 9
PROJECT No. 5014	DRAWING No.
P.O. No. N/A	1395 SHT 4 OF 4

FIGURE WITHHELD AS SENSITIVE UNCLASSIFIED INFORMATION

REVISION		
HOLTEC INTERNATIONAL	EQUIPMENT DESIGN ANALYSIS CONSULTING	
DESCRIPTION HI-STAR 100 MPC-24 CONSTRUCTION		
CLIENT N/A		
COMPANION DRAWINGS 1395, 2889 TO 2892		REV. 15
PROJECT No. 5014	DRAWING No. 1396 SHT 1 OF 6 (E.I.D. 2894)	
P.O. No. N/A		

FIGURE WITHHELD AS SENSITIVE UNCLASSIFIED INFORMATION

REVISION	
HOLTEC INTERNATIONAL	EQUIPMENT DESIGN ANALYSIS CONSULTING
DESCRIPTION HI-STAR 100 MPC-24 CONSTRUCTION	
CLIENT N/A	
COMPANION DRAWINGS 1395, 2889 TO 2892	REV. 13
PROJECT No. 5014	DRAWING No. 1396 SHT 2 OF 6 (E.I.D. 2881)
P.O. No. N/A	

A

FIGURE WITHHELD AS SENSITIVE UNCLASSIFIED INFORMATION

BY:	BY:
REVISION	
HOLTEC INTERNATIONAL	EQUIPMENT DESIGN ANALYSIS CONSULTING
DESCRIPTION HI-STAR 100 MPC-24 CONSTRUCTION	
CLIENT N/A	
COMPANION DRAWINGS 1395, 2889 TO 2892	REV. 12
PROJECT No. 5014	DRAWING No. 1396 SHT 3 OF 6 (E.I.D. 2895)
P.O. No. N/A	

A

FIGURE WITHHELD AS SENSITIVE UNCLASSIFIED INFORMATION

REVISION	
HOLTEC INTERNATIONAL	EQUIPMENT DESIGN ANALYSIS CONSULTING
DESCRIPTION HI-STAR 100 MPC-24 CONSTRUCTION	
CLIENT N/A	
COMPANION DRAWINGS 1395, 2889 TO 2892	REV. 10
PROJECT No. 5014	DRAWING No. 1396 SHT 4 OF 6 (E.I.D. 2896)
P.O. No. N/A	

FIGURE WITHHELD AS SENSITIVE UNCLASSIFIED INFORMATION

REVISION		
HOLTEC INTERNATIONAL	EQUIPMENT DESIGN ANALYSIS CONSULTING	
DESCRIPTION	HI-STAR 100 MPC-24 CONSTRUCTION	
CLIENT N/A		
COMPANION DRAWINGS 1395, 2889 TO 2892		REV. 9
PROJECT No. 5014	DRAWING No. 1396 SHT 5 OF 6 (E.I.D. 2897)	
P.O. No. N/A		

A

FIGURE WITHHELD AS SENSITIVE UNCLASSIFIED INFORMATION

REVISION	
HOLTEC INTERNATIONAL	EQUIPMENT DESIGN ANALYSIS CONSULTING
DESCRIPTION HI-STAR 100 MPC-24 CONSTRUCTION	
CLIENT N/A	
COMPANION DRAWINGS 1395	REV. 8
PROJECT No. 5014	DRAWING No.
P.O. No. N/A	1396 SHT 6 OF 6

A

FIGURE WITHHELD AS SENSITIVE UNCLASSIFIED INFORMATION

REVISION	
HOLTEC INTERNATIONAL	EQUIPMENT DESIGN ANALYSIS CONSULTING
DESCRIPTION HI-STAR 100 MPC-68 CONSTRUCTION	
CLIENT N/A	
COMPANION DRAWINGS 1402	REV. 13
PROJECT No. 5014 P.O. No. N/A	DRAWING No. 1401 SHT 1 OF 4 (E.I.D. 2859)

FIGURE WITHHELD AS SENSITIVE UNCLASSIFIED INFORMATION

REVISION

HOLTEC INTERNATIONAL

EQUIPMENT DESIGN
ANALYSIS
CONSULTING

DESCRIPTION
HI-STAR 100 MPC-68 CONSTRUCTION

CLIENT N/A

COMPANION DRAWINGS 1402

REV. 9

PROJECT No. 5014

P.O. No. N/A

DRAWING No. 1401 SHT 2 OF 4

A

FIGURE WITHHELD AS SENSITIVE UNCLASSIFIED INFORMATION

REVISION

HOLTEC
INTERNATIONAL

EQUIPMENT DESIGN
ANALYSIS
CONSULTING

DESCRIPTION
HI-STAR 100 MPC-68 CONSTRUCTION

CLIENT N/A

COMPANION DRAWINGS
1402

REV.
10

PROJECT No. 5014

P.O. No. N/A

DRAWING No.
1401

FIGURE WITHHELD AS SENSITIVE UNCLASSIFIED INFORMATION

REVISION	
HOLTEC INTERNATIONAL	EQUIPMENT DESIGN ANALYSIS CONSULTING
DESCRIPTION HI-STAR 100 MPC-68 CONSTRUCTION	
CLIENT N/A	
COMPANION DRAWINGS 1402	REV. 9
PROJECT No. 5014	DRAWING No.
P.O. No. N/A	1401 SHT 4 OF 4

A

FIGURE WITHHELD AS SENSITIVE UNCLASSIFIED INFORMATION

REVISION	
HOLTEC INTERNATIONAL	EQUIPMENT DESIGN ANALYSIS CONSULTING
DESCRIPTION HI-STAR 100 MPC-68 CONSTRUCTION	
CLIENT N/A	
COMPANION DRAWINGS 1401	REV. 15
PROJECT No. 5014	DRAWING No. 1402 SHT 1 OF 6
P.O. No. N/A	(E.I.D. 3084)

FIGURE WITHHELD AS SENSITIVE UNCLASSIFIED INFORMATION

REVISION	
HOLTEC INTERNATIONAL	EQUIPMENT DESIGN ANALYSIS CONSULTING
DESCRIPTION HI-STAR 100 MPC-68 CONSTRUCTION	
CLIENT N/A	
COMPANION DRAWINGS 1401	REV. 14
PROJECT No. 5014	DRAWING No. 1402 SHT 2 OF 6 (E.I.D. 2882)
P.O. No. N/A	

A

FIGURE WITHHELD AS SENSITIVE UNCLASSIFIED INFORMATION

BY: BY:

REVISION

HOLTEC INTERNATIONAL

EQUIPMENT DESIGN
ANALYSIS
CONSULTING

DESCRIPTION
HI-STAR 100 MPC-68 CONSTRUCTION

CLIENT N/A

CLIENT

COMPANION DRAWINGS
1401

REV.
13

PROJECT No. 5014

P.O. No. N/A

DRAWING No.
1402 SHT 3 OF 6
(E.I.D. 3085)

A

FIGURE WITHHELD AS SENSITIVE UNCLASSIFIED INFORMATION

REVISION	
HOLTEC INTERNATIONAL	EQUIPMENT DESIGN ANALYSIS CONSULTING
DESCRIPTION HI-STAR 100 MPC-68 CONSTRUCTION	
CLIENT N/A	
COMPANION DRAWINGS 1401	REV. 11
PROJECT No. 5014	DRAWING No. 1402 SHT 4 OF 6 EID #3096
P.O. No. N/A	

A

FIGURE WITHHELD AS SENSITIVE UNCLASSIFIED INFORMATION

REVISION	
HOLTEC INTERNATIONAL	EQUIPMENT DESIGN ANALYSIS CONSULTING
DESCRIPTION HI-STAR 100 MPC-68 CONSTRUCTION	
CLIENT N/A	
COMPANION DRAWINGS 1401	REV. 10
PROJECT No. 5014	DRAWING No. 1402 SHT 5 OF 6 EID #3097
P.O. No. N/A	

A

FIGURE WITHHELD AS SENSITIVE UNCLASSIFIED INFORMATION

REVISION	
HOLTEC INTERNATIONAL	EQUIPMENT DESIGN ANALYSIS CONSULTING
DESCRIPTION HI-STAR 100 MPC-68 CONSTRUCTION	
CLIENT N/A	
COMPANION DRAWINGS 1401	REV. 10
PROJECT No. 5014	DRAWING No. 1402 SHT 6 OF 6 (E.I.D. 2860)
P.O. No. N/A	

A

CHAPTER 1[†]: GENERAL DESCRIPTION

1.0 GENERAL INFORMATION

This Topical Safety Analysis Report (TSAR) for Holtec International's HI-STORM 100 System is a compilation of information and analyses to support a United States Nuclear Regulatory Commission (NRC) licensing review as a spent nuclear fuel (SNF) dry storage cask under requirements specified in 10CFR72 [1.0.1]. This application seeks NRC approval and issuance of a Certificate of Compliance (C of C) for storage under provisions of 10CFR72, Subpart L, for the HI-STORM 100 System to safely store spent nuclear fuel (SNF) at an Independent Spent Fuel Storage Installation (ISFSI). This report has been prepared in the format and content suggested in NRC Regulatory Guide 3.61 [1.0.2] and NUREG-1536 Standard Review Plan for Dry Cask Storage Systems [1.0.3] to facilitate the NRC review process.

The purpose of this chapter is to provide a general description of the design features and storage capabilities of the HI-STORM 100 System, drawings of the structures, systems, and components important to safety, and the qualifications of the applicant. This report is also suitable for incorporation into a site-specific Safety Analysis Report which may be submitted by an applicant for ~~the~~ *a* license to store SNF at an ISFSI or a facility similar in objective and scope. Table 1.0.1 contains a listing of the terminology and notation used in this TSAR.

To aid NRC review, additional tables and references have been added to facilitate the location of information requested by NUREG-1536. Table 1.0.2 provides a matrix of the topics in NUREG-1536 and Regulatory Guide 3.61, the corresponding 10CFR72 requirements, and a reference to the applicable TSAR section that addresses each topic.

The HI-STORM 100 TSAR is in full compliance with the intent of all regulatory requirements listed in Section III of each chapter of NUREG-1536. However, an exhaustive review of the provisions in NUREG-1536, particularly Section IV (Acceptance Criteria) and Section V (Review Procedures) has identified certain deviations from a verbatim compliance to all *guidance* ~~requirements~~. A list of all such items, along with a discussion of their intent and Holtec International's approach for compliance with the underlying intent is presented in Table 1.0.3 herein. Table 1.0.3 also contains the justification for the alternative method for compliance adopted in this TSAR. The justification may be in the form of a supporting analysis, established industry practice, or other NRC guidance

† 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).

documents. Each chapter in this TSAR provides a clear statement with respect to the extent of compliance to the NUREG-1536 provisions.

Chapter 1 is in full compliance with NUREG-1536; no exceptions are taken.

The generic design basis and the corresponding safety analysis of the HI-STORM 100 System contained in this TSAR are intended to bound the SNF characteristics, design, conditions, and interfaces that exist in the vast majority of domestic power reactor sites and potential away-from-reactor storage sites in the contiguous United States. This TSAR also provides the basis for component fabrication and acceptance, and the requirements for safe operation and maintenance of the components, consistent with the design basis and safety analysis documented herein. In accordance with 10CFR72, Subpart K, site-specific implementation of the generically certified HI-STORM 100 System requires that the licensee perform a site-specific safety evaluation, as defined in 10CFR72.212. The HI-STORM 100 System TSAR identifies a limited number of conditions that are necessarily site-specific and are to be addressed in the licensee's 10CFR72.212 evaluation. These include:

- Siting of the ISFSI and design of the storage pad and security system. Site-specific demonstration of compliance with regulatory dose limits. Implementation of a site-specific ALARA program.

- An evaluation of site-specific hazards and design conditions that may exist at the ISFSI site or the transfer route between the plant's cask receiving bay and the ISFSI. These include, but are not limited to, explosion and fire hazards, flooding conditions, land slides, and lightning protection.

- Determination that the physical and nucleonic characteristics and the condition of the SNF assemblies to be dry stored meet the fuel acceptance requirements of the Certificate of Compliance.

- An evaluation of interface and design conditions that exist within the plant's fuel building in which canister fuel loading, canister closure, and canister transfer operations are to be conducted in accordance with the applicable 10CFR50 requirements and technical specifications for the plant.

- Detailed site-specific operating, maintenance, and inspection procedures prepared in accordance with the generic procedures and requirements provided in Chapters 8 and 9, and the technical specifications provided in the Certificate of Compliance.

- Performance of pre-operational testing.

- Implementation of a safeguards and accountability program in accordance with 10CFR73. Preparation of a physical security plan in accordance with 10CFR73.55.

- Review of the reactor emergency plan, quality assurance (QA) program, training program, and radiation protection program.

The generic safety analyses contained in the HI-STORM 100 TSAR may be used as input and for guidance by the licensee in performing a 10CFR72.212 evaluation.

Within this report, all figures, tables and references cited are identified by the double decimal system m.n.i, where m is the chapter number, n is the section number, and i is the sequential number. Thus, for example, Figure 1.2.3 is the third figure in Section 1.2 of Chapter 1.

Revision of this document to Revision ~~8~~ *11* was made on a *page or* section level basis *depending upon the extent of the changes*. Therefore, *from chapter to chapter,* if any change occurred in a section, *anywhere from a single page to* the whole section was updated to Revision ~~8~~ *11*. The sole exception is the figures ~~and drawings~~, which were updated to Revision ~~8~~ *11 on a figure-specific basis* only if a change was made specifically to that figure. ~~or drawing~~. *Drawings are controlled separately within the Holtec QA program and have individual revision numbers. Bills-of-Material (BOMs) are considered separate drawings and are not necessarily at the same revision level as the drawing(s) to which they apply. If a drawing was revised in support of the current TSAR revision, that drawing is included in Section 1.5 at its latest revision level.*

Table 1.0.1

TERMINOLOGY AND NOTATION

ALARA is an acronym for As Low As Reasonably Achievable.

Boral is a generic term to denote an aluminum-boron carbide cermet manufactured in accordance with U.S. Patent No. 4027377. The individual material supplier may use another trade name to refer to the same product.

Boral™ means Boral manufactured by AAR Advanced Structures.

BWR is an acronym for boiling water reactor.

C.G. is an acronym for center of gravity.

Confinement Boundary means the outline formed by the sealed, cylindrical enclosure of the Multi-Purpose Canister (MPC) shell welded to a solid baseplate, a lid welded around the top circumference of the shell wall, the port cover plates welded to the lid, and the closure ring welded to the lid and MPC shell providing the redundant sealing.

Confinement System means the Multi-Purpose Canister (MPC) which encloses and confines the spent nuclear fuel during storage.

Controlled Area means that area immediately surrounding an ISFSI for which the owner/user exercises authority over its use and within which operations are performed.

***Cooling Time** for a spent fuel assembly is the time between its discharge from the reactor (reactor shutdown) and the time the spent fuel assembly is loaded into the multi-purpose canister.*

DBE means Design Basis Earthquake.

DCSS is an acronym for Dry Cask Storage System.

Damaged Fuel Assembly is a fuel assembly with known or suspected cladding defects, as determined by review of records, greater than pinhole leaks or hairline cracks, ~~missing~~ *empty* fuel rod *locations* that are not replaced with dummy fuel rods, or those that cannot be handled by normal means. ~~A damaged fuel assembly's inability to be handled by normal means may be due to mechanical damage and must not be due to fuel cladding damage.~~ *Fuel assemblies which*

Table 1.0.1

TERMINOLOGY AND NOTATION

cannot be handled by normal means due to fuel cladding damage are considered fuel debris.

Damaged Fuel Container *(or Canister)* means a specially designed enclosure for damaged fuel or fuel debris which permits gaseous and liquid media to escape while minimizing dispersal of gross particulates. The Damaged Fuel Container/*Canister* (DFC) features a lifting location which is suitable for remote handling of a loaded or unloaded DFC.

Design Life is the minimum duration for which the component is engineered to perform its intended function set forth in this TSAR, if operated and maintained in accordance with this TSAR.

Design Report is a document prepared, reviewed and QA validated in accordance with the provisions of 10CFR72 Subpart G. The Design Report shall demonstrate compliance with the requirements set forth in the Design Specification. A Design Report is mandatory for systems, structures, and components designated as Important to Safety.

Design Specification is a document prepared in accordance with the quality assurance requirements of 10CFR72 Subpart G to provide a complete set of design criteria and functional requirements for a system, structure, or component, designated as Important to Safety, intended to be used in the operation, implementation, or decommissioning of the HI-STORM 100 System.

Enclosure Vessel means the pressure vessel defined by the cylindrical shell, baseplate, port cover plates, lid, and closure ring which provides confinement for the helium gas contained within the MPC. The Enclosure Vessel (EV) and the fuel basket together constitute the multi-purpose canister.

Fracture Toughness is a property which is a measure of the ability of a material to limit crack propagation under a suddenly applied load.

Fuel Basket means a honeycombed structural weldment with square openings which can accept a fuel assembly of the type for which it is designed.

Fuel Debris refers to ruptured fuel rods, severed rods, ~~and~~ loose fuel pellets, *or fuel assemblies* with known or suspected defects which cannot be handled by normal means due to fuel cladding damage.

High Burnup Fuel *is spent fuel assemblies with burnups greater than 45,000 MWD/MTU.*

HI-TRAC transfer cask or HI-TRAC means the transfer cask used to house the MPC during MPC fuel loading, unloading, drying, sealing, and on-site transfer operations to a HI-STORM storage overpack or HI-STAR storage/transportation overpack. The HI-TRAC shields the loaded MPC allowing loading operations to be performed while limiting radiation exposure to personnel. The HI-TRAC is equipped with a pair of lifting trunnions and pocket trunnions to lift and downend/upend the HI-TRAC with a loaded MPC. HI-TRAC is an acronym for **H**oltec **I**nternational **Tra**nsfer **C**ask. In this submittal there are two HI-TRAC transfer casks, the 125 ton HI-TRAC (HI-TRAC-125) and the 100 ton HI-TRAC (HI-TRAC-100). The 100 ton HI-TRAC is provided for use at sites with a maximum crane capacity of ~~100~~ *less than 125* tons. The term HI-TRAC is used as a generic term to refer to both the 125 ton and 100 ton HI-TRAC.

HI-STORM 100 overpack or storage overpack means the cask ~~which~~ *that* receives and contains the sealed multi-purpose canisters containing spent nuclear fuel. It provides the gamma and neutron shielding, ventilation passages, missile protection, and protection against natural phenomena and accidents for the MPC. *The term "overpack" as used in this TSAR refers to both the standard design overpack (HI-STORM 100), the short design overpack (HI-STORM 100S), and either of these as an overpack designed for high seismic deployment (HI-STORM 100A or HI-STORM 100SA) unless otherwise clarified.*

HI-STORM 100 System consists of a loaded MPC placed within the HI-STORM 100 overpack.

~~**Holtite™** is a trade name denoting an approved neutron shield material for use in the HI-STORM 100 System. In this application, Holtite-A is the only approved neutron shield material.~~

Holtite™ is the trade name for all present and future neutron shielding materials formulated under Holtec International's R&D program dedicated to developing shielding materials for application in dry storage and transport systems. The Holtite development program is an ongoing experimentation effort to identify neutron shielding materials with enhanced shielding and temperature tolerance characteristics. Holtite-A™ is the first and only shielding material qualified under the Holtite R&D program. As such, the terms Holtite and Holtite-A may be used interchangeably throughout this TSAR.

Holtite™-A is a *trademarked Holtec International neutron shield material.* ~~commercially available neutron shield material developed by Bisco, Inc., and currently sold under the trade name NS-4-FR. The neutron shield material is specified with a minimum B_4C loading of 1 weight percent. An equivalent neutron shield material with equivalent neutron shielding properties and composition, but not sold under the trade name NS-4-FR, may be used.~~

Table 1.0.1 (continued)

TERMINOLOGY AND NOTATION

Important to Safety (ITS) means a function or condition required to store spent nuclear fuel safely; to prevent damage to spent nuclear fuel during handling and storage, and to provide reasonable assurance that spent nuclear fuel can be received, handled, packaged, stored, and retrieved without undue risk to the health and safety of the public.

Independent Spent Fuel Storage Installation (ISFSI) means a facility designed, constructed, and licensed for the interim storage of spent nuclear fuel and other radioactive materials associated with spent fuel storage in accordance with 10CFR72.

Intact Fuel Assembly is defined as a fuel assembly without known or suspected cladding defects greater than pinhole leaks and hairline cracks, and which can be handled by normal means. Partial fuel assemblies, that is fuel assemblies from which fuel rods are missing, shall not be classified as Intact Fuel Assemblies unless dummy fuel rods are used to displace an amount of water greater than or equal to that displaced by the original fuel rod(s).

License Life means the duration for which the system is authorized by virtue of its certification by the U.S. NRC.

Lowest Service Temperature (LST) is the minimum metal temperature of a part for the specified service condition.

Maximum Reactivity means the highest possible k-effective including bias, uncertainties, and calculational statistics evaluated for the worst-case combination of fuel basket manufacturing tolerances.

METCON™ is a trade name for the HI-STORM 100 overpack. The trademark is derived from the **metal-con**crete composition of the HI-STORM 100 overpack.

MGDS is an acronym for Mined Geological Disposal System.

***Moderate Burnup Fuel** is spent fuel assemblies with burnups less than or equal to 45,000 MWD/MTU.*

Multi-Purpose Canister (MPC) means the sealed canister which consists of a honeycombed fuel basket for spent nuclear fuel storage, contained in a cylindrical canister shell which is welded to a baseplate, lid with welded port cover plates, and closure ring. MPC is an acronym for multi-purpose canister. There are different MPCs with different fuel basket geometries for storing PWR or BWR fuel, but all MPCs have identical exterior dimensions. The MPC is the confinement boundary for storage conditions. ~~The MPCs used as part of the HI-STORM 100 System are identical to the HI-STAR 100 MPCs evaluated in the HI-STAR 100 storage (Docket No. 72-1008) and transport (Docket No. 71-9261) applications.~~

NDT is an acronym for Nil Ductility Transition Temperature, which is defined as the temperature at which the fracture stress in a material with a small flaw is equal to the yield stress in the same material if it had no flaws.

Neutron Shielding means a material used to thermalize and capture neutrons emanating from the radioactive spent nuclear fuel.

Non-Fuel Hardware *is defined as Burnable Poison Rod Assemblies (BPRAs), Thimble Plug Devices (TPDs), Control Rod Assemblies (CRAs), Axial Power Shaping Rods (APSRs) and other similarly designed devices with different names*

Planar-Average Initial Enrichment is the average of the distributed fuel rod initial enrichments within a given axial plane of the assembly lattice.

Preferential Fuel Loading *is a requirement in the CoC applicable to uniform fuel loading whenever fuel assemblies with significantly different post-irradiation cooling times (≥ 1 year) are to be loaded in the same MPC. Fuel assemblies with the longest post-irradiation cooling time are loaded into fuel storage locations at the periphery of the basket. Fuel assemblies with shorter post-irradiation cooling times are placed toward the center of the basket. Regionalized fuel loading meets the intent of preferential fuel loading. Preferential fuel loading is a requirement in addition to other restrictions in the CoC such as those for non-fuel hardware and damaged fuel containers.*

Post-Core Decay Time (PCDT) *is synonymous with cooling time.*

PWR is an acronym for pressurized water reactor.

Reactivity is used synonymously with effective neutron multiplication factor or k-effective.

Regionalized Fuel Loading *is a term used to describe an optional fuel loading strategy used in lieu of uniform fuel loading. Regionalized fuel loading allows high heat emitting fuel assemblies*

Table 1.0.1 (continued)

TERMINOLOGY AND NOTATION

to be stored in fuel storage locations in the center of the fuel basket provided lower heat emitting fuel assemblies are stored in the peripheral fuel storage locations. Users choosing regionalized fuel loading must also consider other restrictions in the CoC such as those for non-fuel hardware and damaged fuel containers. Regionalized fuel loading meets the intent of preferential fuel loading.

SAR is an acronym for Safety Analysis Report (10CFR71).

Service Life means the duration for which the component is reasonably expected to perform its intended function, if operated and maintained in accordance with the provisions of this TSAR. Service Life may be much longer than the Design Life because of the conservatism inherent in the codes, standards, and procedures used to design, fabricate, operate, and maintain the component.

Single Failure Proof means that the handling system is designed so that all directly loaded tension and compression members are engineered to satisfy the enhanced safety criteria of Paragraphs 5.1.6(1)(a) and (b) of NUREG-0612.

SNF is an acronym for spent nuclear fuel.

SSC is an acronym for Structures, Systems and Components.

STP is Standard Temperature and Pressure conditions.

Thermosiphon *is the term used to describe the buoyancy-driven natural convection circulation of helium within the MPC fuel basket.*

TSAR is an acronym for Topical Safety Analysis Report (10CFR72).

Uniform Fuel Loading *is a fuel loading strategy where any authorized fuel assembly may be stored in any fuel storage location, subject to other restrictions in the CoC, such as preferential fuel loading, and those applicable to non-fuel hardware, and damaged fuel containers.*

ZPA is an acronym for zero period acceleration.

1.1 INTRODUCTION

HI-STORM 100 (acronym for Holtec International Storage and Transfer Operation Reinforced Module) is a spent nuclear fuel storage system designed to be in full compliance with the requirements of 10CFR72. The annex "100" is a model number designation which denotes a system weighing over 100 tons. The HI-STORM 100 System consists of a sealed metallic canister, herein abbreviated as the "MPC", contained within an overpack. Its design features are intended to simplify and reduce on-site SNF loading, handling, and monitoring operations, and to provide for radiological protection and maintenance of structural and thermal safety margins.

The HI-STORM 100S overpack is a variant of the HI-STORM 100 overpack and has its own set of design drawings in Section 1.5. The "S" suffix indicates a shorter overpack with a re-designed top lid. The HI-STORM 100S accepts the same MPCs and fuel types as the HI-STORM 100 and the basic structural, shielding, and thermal-hydraulic characteristics remain unchanged. Hereafter in this TSAR reference to HI-STORM 100 System or the HI-STORM 100 overpack is construed to apply to both the HI-STORM 100 and the HI-STORM 100S. Where appropriate, the text distinguishes between the two overpack designs. See Figures 1.1.1A and 1.1.3A.

[redacted]

The HI-STORM 100 System is designed to accommodate a wide variety of spent nuclear fuel assemblies in a single overpack design by utilizing different MPCs. The external dimensions of all MPCs are identical to allow the use of a single overpack. Each of the MPCs has different internals (baskets) to accommodate distinct fuel characteristics. Each MPC is identified by the maximum quantity of fuel assemblies it is capable of receiving. The MPC-24, *MPC-24E, and MPC-24EF* contains a maximum of 24 PWR fuel assemblies*; the MPC-32 contains a maximum of 32 PWR fuel assemblies;* and the MPC-68, *MPC-68F, and MPC-68FF* contains a maximum of 68 BWR fuel assemblies.

The HI-STORM 100 overpack is constructed from a combination of steel and concrete, both of which are materials with long, proven histories of usage in nuclear applications. HI-STORM 100 incorporates and combines many desirable features of previously-approved concrete and metal module designs. In essence, the HI-STORM 100 overpack is a hybrid of metal and concrete systems, with the design objective of emulating the best features and dispensing with the drawbacks of both. The HI-STORM overpack is best referred to as a METCON™ (metal/concrete composite) system.

Figure 1.1.1 shows the HI-STORM 100 with two of its major constituents, the MPC and the storage overpack, in a cut-away view. The MPC, shown partially withdrawn from the storage overpack, is an integrally welded pressure vessel designed to meet the stress limits of the ASME Boiler and Pressure Vessel Code, Section III, Subsection NB [1.1.1]. The MPC defines the confinement boundary for the stored spent nuclear fuel assemblies with respect to 10CFR72 requirements and attendant review considerations. The HI-STORM 100 storage overpack provides mechanical protection, cooling, and radiological shielding for the contained MPC.

In essence, the HI-STORM 100 System is the storage-only counterpart of the HI-STAR 100 System (Docket Numbers 72-1008 (Ref. [1.1.2]) and 71-9261 (Ref. [1.1.3])). Both HI-STORM and HI-STAR are engineered to house identical MPCs. Since the MPC is designed to meet the requirements of both 10CFR71 and 10CFR72 for transportation and storage, respectively, the HI-STORM 100 System allows rapid decommissioning of the ISFSI by simply transferring the loaded MPC's directly into HI-STAR 100 overpacks for off-site transport. This alleviates the additional fuel handling steps required by storage-only casks to unload the cask and repackage the fuel into a suitable transportation cask.

In contrast to the HI-STAR 100 overpack, which provides a containment boundary for the SNF during transport, the HI-STORM 100 storage overpack does not constitute a containment or confinement enclosure. The HI-STORM 100 overpack is equipped with large penetrations near its lower and upper extremities to permit natural circulation of air to provide for the passive cooling of the MPC and the contained radioactive material. The HI-STORM is engineered to be an effective barrier against the radiation emitted by the stored materials, and an efficiently configured metal/concrete composite to attenuate the loads transmitted to the MPC during a natural phenomena or hypothetical accident event. Other auxiliary functions of the HI-STORM 100 overpack include isolation of the SNF from abnormal environmental or man-made events, such as impact of a tornado borne missile. As the subsequent chapters of this TSAR demonstrate, the HI-STORM overpack is engineered with large margins of safety with respect to cooling, shielding, and mechanical/structural functions.

The HI-STORM 100 System is autonomous inasmuch as it provides SNF and radioactive material confinement, radiation shielding, criticality control and passive heat removal independent of any other facility, structures, or components. The surveillance and maintenance required by the plant's staff is minimized by the HI-STORM 100 System since it is completely passive and is composed of materials with long proven histories in the nuclear industry. The HI-STORM 100 System can be used either singly or as the basic storage module in an ISFSI. The site for an ISFSI can be located either at a reactor or away from a reactor.

The information presented in this report is intended to demonstrate the acceptability of the HI-STORM 100 System for use under the general license provisions of Subpart K by meeting the criteria set forth in 10CFR72.236.

The modularity of the HI-STORM 100 System accrues several advantages. Different MPCs, identical in exterior dimensions, manufacturing requirements, and handling features, but different in their SNF arrangement details, are designed to fit a common overpack. Even though the different MPCs have fundamentally identical design and manufacturing attributes, qualification of HI-STORM 100 requires consideration of the variations in the characteristics of the MPCs. In most cases, however, it is possible to identify the most limiting MPC geometry and the specific loading condition for the safety evaluation, and the detailed analyses are then carried out for that bounding condition. In those cases where this is not possible, multiple parallel analyses are performed.

The HI-STORM overpack is not engineered for transport and, therefore, will not be submitted for 10CFR Part 71 certification. HI-STORM 100, however, is designed to possess certain key elements of flexibility.

For example:

- The HI-STORM 100 overpack is stored at the ISFSI pad in a vertical orientation which helps minimize the size of the ISFSI and leads to an effective natural convection cooling flow around the MPC.

- The HI-STORM 100 overpack can be loaded with a loaded MPC using the HI-TRAC transfer cask inside the 10CFR50 [1.1.4] facility, prepared for storage, transferred to the ISFSI, and stored in a vertical configuration, or directly loaded using the HI-TRAC transfer cask at *or nearby* the ISFSI storage pad.

[redacted]

The MPC is a multi-purpose SNF storage device both with respect to the type of fuel assemblies and its versatility of use. The MPC is engineered as a cylindrical prismatic structure with square cross section storage cavities. The number of storage locations depends on the type of fuel. Regardless of the storage cell count, the construction of the MPC is fundamentally the same; it is built as a honeycomb of cellular elements positioned within a circumscribing cylindrical canister shell. The manner of cell-to-cell weld-up and cell-to-canister shell interface employed in the MPC imparts extremely high structural stiffness to the assemblage, which is an important attribute for mechanical accident events. Figure 1.1.2 shows an elevation cross section of a MPC.

The MPC is identical to those presented in References [1.1.2] and [1.1.3], *except for MPC-24. 24E, 24EF, 32, and 68FF* . Referencing these ~~submittals~~ *documents, as applicable,* avoids repetition of information on the MPCs which is comprehensively set forth in the above-mentioned Holtec International ~~applications~~ *documents* docketed with the NRC. However, sufficient information and drawings are presented in this report to maintain clarity of exposition of technical data.

The HI-STORM 100 storage overpack is designed to provide the necessary neutron and gamma shielding to comply with the provisions of 10CFR72 for dry storage of SNF at an ISFSI. A cross sectional view of the HI-STORM 100 storage overpack is presented in Figure 1.1.3. A HI-TRAC transfer cask is required for loading of the MPC and movement of the loaded MPC from the cask loading area of a nuclear plant spent fuel pool to the storage overpack. The HI-TRAC is engineered to be emplaced with an empty MPC into the cask loading area of nuclear plant spent fuel pools for fuel loading (or unloading). The HI-TRAC/MPC assembly is designed to preclude intrusion of pool water into the narrow annular space between the HI-TRAC and the MPC while the assembly is submerged in the pool water. The HI-TRAC transfer cask also allows dry loading (or unloading) of SNF into the MPC.

To summarize, the HI-STORM 100 System has been engineered to:

- minimize handling of the SNF;
- provide shielding and physical protection for the MPC;
- permit rapid and unencumbered decommissioning of the ISFSI;
- require minimal ongoing surveillance and maintenance by plant staff;
- minimize dose to operators during loading and handling;
- allow transfer of the loaded MPC to a HI-STAR overpack for transportation.

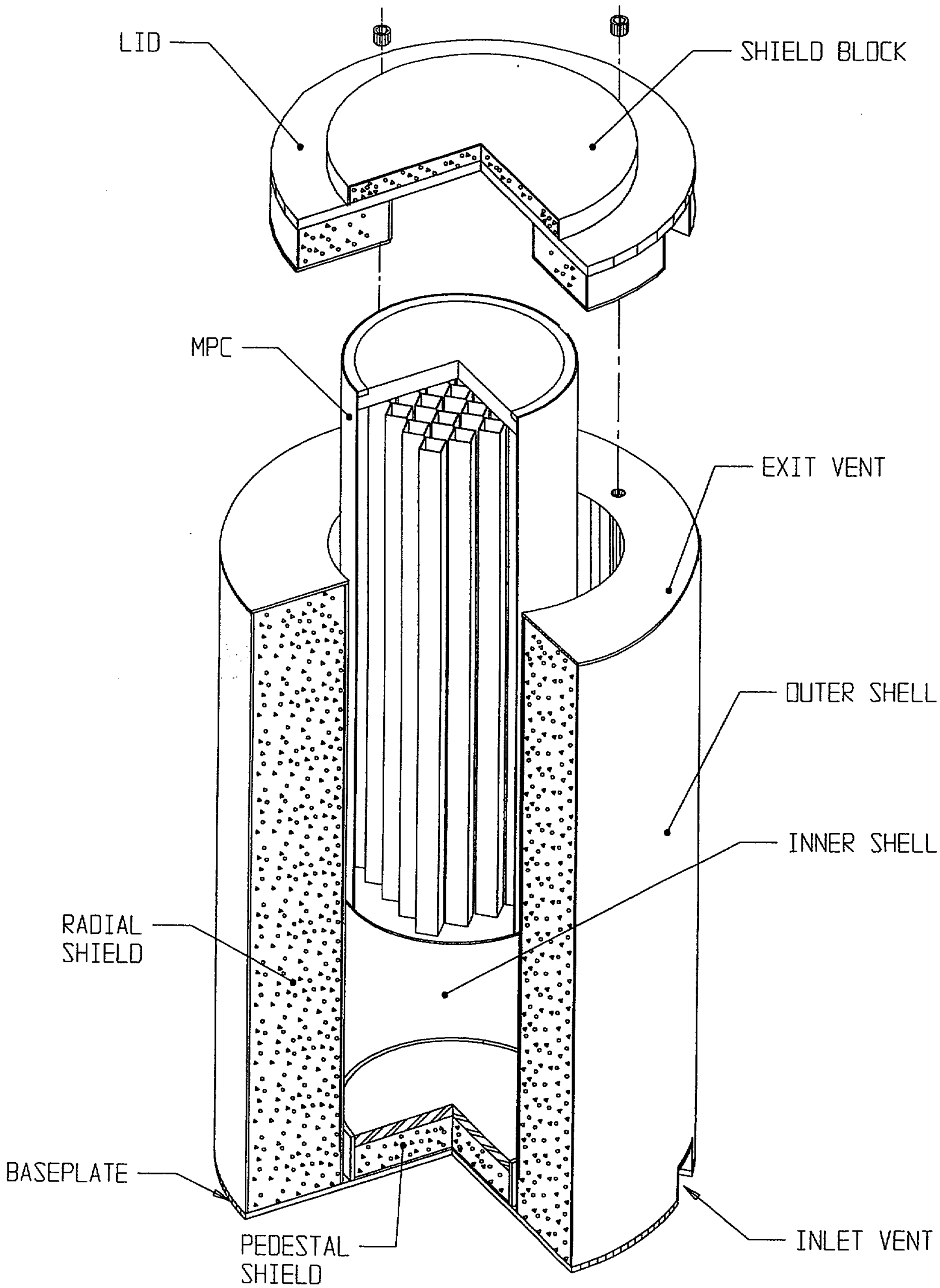

FIGURE 1.1.1A; HI-STORM 100S OVERPACK WITH MPC PARTIALLY INSERTED

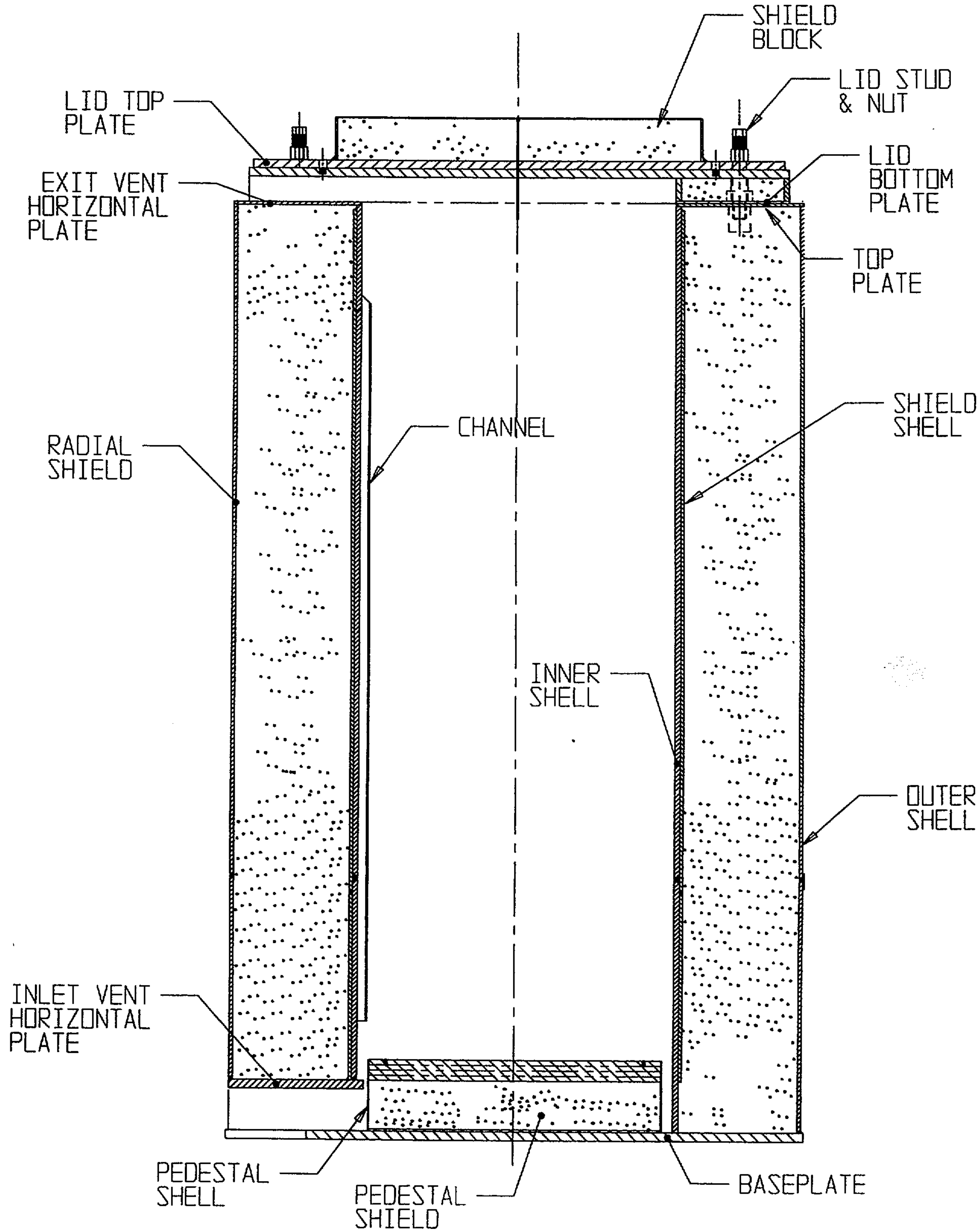

FIGURE 1.1.3A; HI-STORM 100S OVERPACK CROSS SECTIONAL ELEVATION VIEW

FIGURES 1.4.4 AND 1.1.4 ARE HOLTEC PROPRIETARY

1.2 GENERAL DESCRIPTION OF HI-STORM 100 System

1.2.1 System Characteristics

The basic HI-STORM 100 System consists of interchangeable MPCs providing a confinement boundary for BWR or PWR spent nuclear fuel, a storage overpack providing a structural and radiological boundary for long-term storage of the MPC placed inside it, and a transfer cask providing a structural and radiological boundary for transfer of a loaded MPC from a nuclear plant spent fuel storage pool to the storage overpack. Figure 1.2.1 provides a cross sectional view of the HI-STORM 100 System with an MPC inserted into a storage overpack. *Figure 1.2.1A provides a cross sectional view of the HI-STORM 100S System with an MPC inserted into a storage overpack.* Each of these components is described below, including information with respect to component fabrication techniques and designed safety features. All structures, systems, and components of the HI-STORM 100 System which are identified as Important to Safety are specified in Table 2.2.6. This discussion is supplemented with a full set of detailed design drawings in Section 1.5.

The HI-STORM 100 System is comprised of three discrete components:

i. multi-purpose canister (MPC)
ii. storage overpack (HI-STORM)
iii. transfer cask (HI-TRAC)

Necessary auxiliaries required to deploy the HI-STORM 100 System for storage are:

i. vacuum drying system
ii. helium (He) backfill system with leakage detector
iii. lifting and handling systems
iv welding equipment
v. transfer vehicles/trailer

All MPCs have identical exterior dimensions which render them interchangeable. The outer diameter of the MPC is 68-3/8 inches† and the overall length is 190-1/2 inches. See Section 1.5 for the detailed design drawings. Due to the differing storage contents of each MPC, the maximum loaded weight differs ~~between~~ *among* MPCs. See Table 3.2.1 for each MPC weight. However, the maximum weight of a loaded MPC is approximately 44-1/2 tons. Tables 1.2.1 and 1.2.2 contain the key parameters for the MPCs.

A single, *base* HI-STORM overpack design is provided which is capable of storing each type of MPC. The overpack inner cavity is sized to accommodate the MPCs. The inner diameter of the overpack inner shell is 73-1/2 inches and the height of the cavity is 191-1/2 inches. The overpack inner shell is provided with channels distributed around the inner cavity to present an inside diameter of 69-1/2 inches. The channels are intended to offer a flexible medium to absorb some of the impact during a non-mechanistic tip-over, while still allowing the cooling air flow through the ventilated

† *Dimensions discussed in this section are considered nominal values.*

overpack. The outer diameter of the overpack is 132-1/2 inches. ~~and~~ The overall height *of the HI-STORM 100 and the HI-STORM 100S* is 239-1/2 inches *and 232 inches, respectively*. See Section 1.5 for the detailed design drawings. The weight of the overpack without an MPC is approximately 135 tons. See Table 3.2.1 for the detailed weights.

Before proceeding to present detailed physical data on the HI-STORM 100 System, it is of contextual importance to summarize the design attributes which enhance the performance and safety of the system. Some of the principal features of the HI-STORM 100 System which enhance its effectiveness as an SNF storage device and a safe SNF confinement structure are:

- the honeycomb design of the MPC fuel basket;
- the effective distribution of neutron and gamma shielding materials within the system;
- the high heat dissipation capability;
- engineered features to promote convective heat transfer;
- the structural robustness of the steel-concrete-steel overpack construction.

The honeycomb design of the MPC fuel baskets renders the basket into a multi-flange plate weldment where all structural elements (i.e., box walls) are arrayed in two orthogonal sets of plates. Consequently, the walls of the cells are either completely co-planar (i.e., no offset) or orthogonal with each other. There is complete edge-to-edge continuity between the contiguous cells.

Among the many benefits of the honeycomb construction is the uniform distribution of the metal mass of the basket over the entire length of the basket. Physical reasoning suggests that a uniformly distributed mass provides a more effective shielding barrier than can be obtained from a nonuniform basket. In other words, the honeycomb basket is a most effective radiation attenuation device. The complete cell-to-cell connectivity inherent in the honeycomb basket structure provides an uninterrupted heat transmission path, making the MPC an effective heat rejection device.

The composite shell construction in the overpack, steel-concrete-steel, allows ease of fabrication and eliminates the need for the sole reliance on the strength of concrete.

A description of each of the components is provided in the following sections, along with information with respect to its fabrication and safety features. This discussion is supplemented with the full set of Design Drawings and Bills-of-Material in Section 1.5.

1.2.1.1 Multi-Purpose Canisters

The MPCs are welded cylindrical structures as shown in cross sectional views of Figures 1.2.2 ~~and 1.2.4~~ *through 1.2.4.A*. The outer diameter and cylindrical height of each MPC are fixed. Each spent fuel MPC is an assembly consisting of a honeycombed fuel basket, a baseplate, canister shell, a lid, and a closure ring, as depicted in the MPC cross section elevation view, Figure 1.2.5. The number of spent nuclear fuel storage locations in each of the MPCs depends on the fuel assembly characteristics.

There are ~~three~~ *seven* MPC models, distinguished by the type and number of fuel assemblies authorized for loading. The MPC-24 is designed to store up to 24 intact PWR fuel assemblies. *The MPC-24E is designed to store up to 24 total PWR fuel assemblies including up to four (4) damaged PWR fuel assemblies. The MPC-24EF is designed to store up to 24 total PWR fuel assemblies including up to four (4) damaged PWR fuel assemblies or fuel classified as fuel debris.* The MPC-68 is designed to store up to 68 ~~intact~~ *total BWR fuel assemblies including up to 68* damaged *Dresden Unit 1 or Humboldt Bay* BWR fuel assemblies. *Damaged BWR fuel assemblies other than Dresden Unit 1 and Humboldt Bay are limited to 16 fuel storage locations in the MPC-68 with the remainder being intact BWR fuel assemblies, up to a total of 68.* The MPC-68F is designed to store up to 68 intact or damaged *Dresden Unit 1 and Humboldt Bay* BWR fuel assemblies. ~~and~~ Up to four *of the 68 fuel storage locations in the MPC-68F may be Dresden Unit 1 and Humboldt Bay* BWR fuel assemblies classified as fuel debris. *The MPC-68FF is designed to store up to 68 total BWR fuel assemblies including up to 16 damaged BWR fuel assemblies. Up to eight (8) of the 16 BWR damaged fuel assembly storage locations may be filled with BWR fuel classified as fuel debris. In addition, all fuel loading combinations permitted in the MPC-68F are also permitted in the MPC-68FF.* Design Drawings for all of the MPCs are provided in Section 1.5.

The MPC provides the confinement boundary for the stored fuel. Figure 1.2.6 provides an elevation view of the MPC confinement boundary. The confinement boundary is defined by the MPC baseplate, shell, lid, port covers, and closure ring. The confinement boundary is a ~~seal~~-*strength*-welded enclosure of all stainless steel construction.

The ~~construction features of the~~ PWR MPC-24, *MPC-24E* and ~~the BWR MPC-68 are similar. However, the PWR MPC-24 canister in Figure 1.2.4, which is designed for high-enriched PWR fuel,~~ *MPC-24EF* differ~~s~~ in construction from the MPC-68 *(including the MPC-68F and MPC-68FF)* in one important aspect: the fuel storage cells are physically separated from one another by a "flux trap", for criticality control. *The PWR MPC-32 is designed similar to the MPC-68 (without flux traps) and its design includes credit for soluble boron in the MPC water during wet fuel loading and unloading operations for criticality control.* All MPC baskets are formed from an array of plates welded to each other, such that a honeycomb structure is created which resembles a multiflanged, closed-section beam in its structural characteristics.

The MPC fuel basket is positioned and supported within the MPC shell by a set of basket supports welded to the inside of the MPC shell. Between the periphery of the basket, the MPC shell, and the basket supports, heat conduction elements are installed. These heat conduction elements are fabricated from thin aluminum alloy 1100 in shapes *and a design* which ~~enable~~ *allows* a snug fit in the confined spaces and ease of installation. The heat conduction elements are installed along the full length of the MPC basket *except at the drain pipe location* to create a nonstructural thermal connection which facilitates heat transfer from the basket to shell. In their ~~installed~~ *operating* condition, the heat conduction elements contact the MPC shell and basket walls.

Lifting lugs attached to the inside surface of the MPC canister shell serve to permit placement of the empty MPC into the HI-TRAC transfer cask. The lifting lugs also serve to axially locate the MPC lid prior to welding. These internal lifting lugs are not used to handle a loaded MPC. Since the MPC lid is installed prior to any handling of a loaded MPC, there is no access to the lifting lugs once the MPC is loaded.

The top end of the MPC incorporates a redundant closure system. Figure 1.2.6 shows the MPC closure details. The MPC lid is a circular plate edge-welded to the MPC outer shell. This plate is equipped with vent and drain ports which are utilized to remove moisture and air from the MPC, and backfill the MPC with a specified ~~*mass*~~ *pressure* of inert gas (helium). The vent and drain ports are covered and seal welded before the closure ring is installed. The closure ring is a circular ring edge-welded to the MPC shell and lid. The MPC lid provides sufficient rigidity to allow the entire MPC loaded with SNF to be lifted by threaded holes in the MPC lid.

To maintain a constant exterior axial length between the *PWR MPCs* ~~*MPC-24*~~ and *the BWR MPCs* ~~*MPC-68*~~, the thickness of the *PWR MPCs'* ~~*MPC-24*~~ lid is ½ inch thinner than the MPC-68s' lid to accommodate the longest PWR fuel assembly which is approximately a ½ inch longer than the longest BWR fuel assembly. For fuel assemblies that are shorter than the design basis length, upper and lower fuel spacers (as appropriate) maintain the axial position of the fuel assembly within the MPC basket. The upper fuel spacers are threaded into the underside of the MPC lid as shown in Figure 1.2.5. The lower fuel spacers are placed in the bottom of each fuel basket cell. The upper and lower fuel spacers are designed to withstand normal, off-normal, and accident conditions of storage. An axial clearance of approximately 2 inches is provided to account for the irradiation and thermal growth of the fuel assemblies. The *suggested values for the* upper and lower fuel spacer lengths are listed in Tables 2.1.9 and 2.1.10 for each fuel assembly type. *The actual length of fuel spacers will be determined on a site-specific or fuel assembly-specific basis.*

The MPC is constructed entirely from stainless steel alloy materials (except for the neutron absorber and aluminum heat conduction elements). No carbon steel parts are permitted in the MPC. Concerns regarding interaction of coated carbon steel materials and various MPC operating environments [1.2.1] are not applicable to the MPC. All structural components in a MPC shall be made of Alloy X, a designation which warrants further explanation.

Alloy X is a material which is expected to be acceptable as a Mined Geological Disposal System (MGDS) waste package and which meets the thermophysical properties set forth in this document.

At this time, there is considerable uncertainty with respect to the material of construction for an MPC which would be acceptable as a waste package for the MGDS. Candidate materials being considered for acceptability by the DOE include:

- Type 316
- Type 316LN
- Type 304
- Type 304LN

The DOE material selection process is primarily driven by corrosion resistance in the potential environment of the MGDS. As the decision regarding a suitable material to meet disposal requirements is not imminent, this application requests approval for use of any one of the four Alloy X materials.

For the MPC design and analysis, Alloy X (as defined in this application) may be one of the following materials. ~~(o~~*O*nly a single alloy from the list of acceptable Alloy X materials may be used in the fabrication of a single MPC *basket or shell - the basket and shell may be of different alloys*.

- Type 316
- Type 316LN
- Type 304
- Type 304LN

The Alloy X approach is accomplished by qualifying the MPC for all mechanical, structural, neutronic, radiological, and thermal conditions using material thermophysical properties which are the least favorable for the entire group for the analysis in question. For example, when calculating the rate of heat rejection to the outside environment, the value of thermal conductivity used is the lowest for the candidate material group. Similarly, the stress analysis calculations use the lowest value of the ASME Code allowable stress intensity for the entire group. Stated differently, we have defined a material, which is referred to as Alloy X, whose thermophysical properties, from the MPC design perspective, are the least favorable of the candidate materials.

The evaluation of the Alloy X constituents to determine the least favorable properties is provided in Appendix 1.A.

Other alloy materials which are identified to be more suitable by the DOE for the MGDS in the future and which are also bounded by the Alloy X properties set forth in Appendix 1.A can be used in the MPC after an amendment to this TSAR is approved.

The Alloy X approach is conservative because no matter which material is ultimately utilized in the MPC construction, the Alloy X approach guarantees that the performance of the MPC will exceed the analytical predictions contained in this document.

1.2.1.2 Overpacks

1.2.1.2.1 HI-STORM 100 Overpack (Storage

The HI-STORM 100 *and 100S* overpacks ~~is a~~ *are* rugged, heavy-walled cylindrical vessels. Figures 1.2.7, ~~*and*~~ 1.2.8, *and 1.2.8.A* provide cross sectional views of the HI-STORM 100 System, *including both of the overpack designs.* [redacted] The main structural function of the storage overpack is provided by carbon steel, and the main shielding function is provided by plain concrete. The overpack plain concrete is enclosed by cylindrical steel shells, a thick steel baseplate, and a top plate. The overpack lid has appropriate concrete shielding ~~*attached to its underside and top*~~ to provide neutron and gamma attenuation in the vertical direction.

The storage overpack provides an internal cylindrical cavity of sufficient height and diameter for housing an MPC. The inner shell of the overpack has channels attached to its inner diameter. The channels provide guidance for MPC insertion and removal and a flexible medium to absorb impact loads during the non-mechanistic tip-over, while still allowing the cooling air flow to circulate through the overpack. Stainless steel shims are attached to channels to allow the proper inner diameter dimension to be obtained and to provide a guiding surface for MPC insertion and removal.

The storage overpack has air ducts to allow for passive natural convection cooling of the contained MPC. Four air inlets and four air outlets are located at the lower and upper extremities of the overpack, respectively. The air inlets and outlets are covered by a fine mesh screen to reduce the potential for blockage. Routine inspection of the screens (or, alternatively, temperature monitoring) ensures that blockage of the screens themselves will be detected and removed in a timely manner. Analysis, provided in this TSAR, evaluates the effects of partial and complete blockage of the air ducts.

The four air inlets and four air outlets are penetrations through the thick concrete shielding provided by the HI-STORM *100* overpack. *The outlet air ducts for the HI-STORM 100S overpack are integral to the lid.* Within the air inlets and outlets, an array of gamma shield cross plates are installed. These gamma shield cross plates are designed to scatter any particles traveling through the ducts. The result of scattering the particles in the ducts is a significant decrease in the local dose rates around the four air inlets and four air outlets. The configuration of the gamma shield cross plates is such that the increase in the resistance to flow in the air inlets and outlets is minimized.

Four threaded anchor blocks at the top of the overpack are provided for lifting. The anchor blocks are integrally welded to the radial plates which in turn are full-length welded to the overpack inner shell, outer shell, and baseplate (see Figure 1.2.7). The four anchor blocks are located on 90° centers. The overpack may also be lifted from the bottom using specially-designed lifting transport devices, including hydraulic jacks, air pads, and Hillman rollers. Slings or other suitable devices mate with lifting lugs which are inserted into threaded holes in the top surface of the overpack lid to allow lifting of the overpack lid. After the lid is bolted to the storage overpack main body, these lifting bolts shall be removed and replaced with flush plugs.

The plain concrete between the overpack inner and outer steel shells is specified to provide the necessary shielding properties and compressive strength. The concrete shall be in accordance with the requirements specified in Appendix 1.D.

The principal function of the concrete is to provide shielding against gamma and neutron radiation. However, in an implicit manner it helps enhance the performance of the HI-STORM overpack in other respects as well. For example, the massive bulk of concrete imparts a large thermal inertia to the HI-STORM overpack, allowing it to moderate the rise in temperature of the system under hypothetical conditions when all ventilation passages are assumed to be blocked. The case of a postulated fire accident at the ISFSI is another example where the high thermal inertia characteristics of the HI-STORM concrete control the temperature of the MPC. Although the annular concrete mass in the overpack shell is not a structural member, it does act as an elastic/plastic filler of the inter-shell space, such that, while its cracking and crushing under a tip-over accident is not of significant consequence, its deformation characteristics are germane to the analysis of the structural members.

Density and compressive strength are the key parameters which delineate the performance of concrete in the HI-STORM System. The density of concrete used in the inter-shell annulus, pedestal, and HI-STORM lid has been set as defined in Appendix 1.D. For evaluating the physical properties of concrete for completing the analytical models, conservative formulations of Reference [1.2.6] are used.

To ensure the stability of the concrete at temperature, the concrete composition has been specified in accordance with NUREG-1536, "Standard Review Plan for Dry Cask Storage Systems" [1.2.10]. Thermal analyses, presented in Chapter 4, show that the temperatures during normal storage conditions do not threaten the physical integrity of the HI-STORM overpack concrete.

[redacted]

[redacted]

[redacted]

[redacted]

[redacted]

[redacted]

[redacted]

[redacted]

[redacted]

1.2.1.2.2 HI-TRAC (Transfer Cask)

Like the storage overpack, the HI-TRAC transfer cask is a rugged, heavy-walled cylindrical vessel. The main structural function of the transfer cask is provided by carbon steel, and the main neutron and gamma shielding functions are provided by water and lead, respectively. The transfer cask is a steel, lead, steel layered cylinder with a water jacket attached to the exterior. Figure 1.2.9 provides a typical cross section of a HI-TRAC with the pool lid installed.

The transfer cask provides an internal cylindrical cavity of sufficient size for housing an MPC.

The top lid has additional neutron shielding to provide neutron attenuation in the vertical direction (from SNF in the MPC below). The MPC access hole through the HI-TRAC top lid is provided to allow the lowering/raising of the MPC between the HI-TRAC transfer cask, and the HI-STORM or HI-STAR overpacks. The HI-TRAC is provided with two bottom lids, each used separately. The pool lid is bolted to the bottom flange of the HI-TRAC and is utilized during MPC fuel loading and sealing operations. In addition to providing shielding in the axial direction, the pool lid incorporates a seal which is designed to hold clean demineralized water in the HI-TRAC inner cavity, thereby preventing contamination of the exterior of the MPC by the contaminated fuel pool water. After the MPC has been drained, dried, and sealed, the pool lid is removed and the HI-TRAC transfer lid is attached. The transfer lid incorporates two sliding doors which allow the opening of the HI-TRAC bottom for the MPC to be raised/lowered. Figure 1.2.10 provides a cross section of the HI-TRAC with the transfer lid installed.

Trunnions are provided for lifting and rotating the transfer cask body between vertical and horizontal positions. The lifting trunnions are located just below the top flange and the pocket trunnions are located above the bottom flange. The two lifting trunnions are provided to lift and vertically handle the HI-TRAC, and the pocket trunnions provide a pivot point for the rotation of the HI-TRAC for downending or upending.

Two HI-TRAC transfer casks of different weights are provided to house the MPCs. The 125 ton HI-TRAC weight does not exceed 125 tons during any loading or transfer operation. The 100 ton HI-TRAC weight does not exceed 100 tons during any loading or transfer operation. The internal cylindrical cavities of the two HI-TRACs are identical. However, the external dimensions are different. The 100ton HI-TRAC has a reduced thickness of lead and water shielding and consequently, the external dimensions are different. The structural steel thickness is identical in the two HI-TRACs. This allows most structural analyses of the 125 ton HI-TRAC to bound the 100 ton HI-TRAC design. Additionally, as the two HI-TRACs are identical except for a reduced thickness of lead and water, the 125 ton HI-TRAC has a larger thermal resistance than the smaller and lighter 100 ton HI-TRAC. Therefore, for normal conditions the 125 ton HI-TRAC thermal analysis bounds that of the 100 ton HI-TRAC. Separate shielding analyses are performed for each HI-TRAC since the shielding thicknesses are different between the two.

1.2.1.3 Shielding Materials

The HI-STORM 100 System is provided with shielding to ensure the radiation and exposure requirements in 10CFR72.104 and 10CFR72.106 are met. This shielding is an important factor in minimizing the personnel doses from the gamma and neutron sources in the SNF in the MPC for ALARA considerations during loading, handling, transfer, and storage. The fuel basket structure of edge-welded composite boxes and Boral™ neutron poison panels attached to the fuel storage cell vertical surfaces provide the initial attenuation of gamma and neutron radiation emitted by the radioactive spent fuel. The MPC shell, baseplate, lid and closure ring provide additional thicknesses of steel to further reduce the gamma flux at the outer canister surfaces.

In the HI-STORM 100 storage overpack, the primary shielding in the radial direction is provided by concrete and steel. In addition, the storage overpack has a thick circular concrete slab attached to the underside of the lid, and a thick circular concrete pedestal upon which the MPC rests. These slabs provide gamma and neutron attenuation in the axial direction. The thick overpack lid and concrete shield ring atop the lid provide additional gamma attenuation in the upward direction, reducing both direct radiation and skyshine. Several steel plate and shell elements provide additional gamma shielding as needed in specific areas, as well as incremental improvements in the overall shielding effectiveness.

In the HI-TRAC transfer cask radial direction, gamma and neutron shielding consists of steel-lead-steel and water, respectively. In the axial direction, shielding is provided by the top lid, and the pool or transfer lid. In the HI-TRAC pool lid, layers of steel-lead-steel provide an additional measure of gamma shielding to supplement the gamma shielding at the bottom of the MPC. In the transfer lid, layers of steel-lead-steel provide gamma attenuation. For the 125 ton HI-TRAC transfer lid, the neutron shield material, Holtite-A, is also provided. The 125 ton HI-TRAC top lid is composed of steel-neutron shield-steel, with the neutron shield material being Holtite-A. The 100 ton HI-TRAC top lid is composed of steel only providing gamma attenuation.

1.2.1.3.1 Boral Neutron Absorber

Boral is a thermal neutron poison material composed of boron carbide and aluminum (aluminum powder and plate). Boron carbide is a compound having a high boron content in a physically stable and chemically inert form. The boron carbide contained in Boral is a fine granulated powder that conforms to ASTM C-750-80 nuclear grade Type III. The Boral cladding is made of alloy aluminum, a lightweight metal with high tensile strength which is protected from corrosion by a highly resistant oxide film. The two materials, boron carbide and aluminum, are chemically compatible and ideally suited for long-term use in the radiation, thermal, and chemical environment of a nuclear reactor, spent fuel pool, or dry cask.

The documented historical applications of Boral, in environments comparable to those in spent fuel pools and fuel storage casks, dates to the early 1950s (the U.S. Atomic Energy Commission's AE-6 Water-Boiler Reactor [1.2.2]). Technical data on the material was first printed in 1949, when the report "Boral: A New Thermal Neutron Shield" was published [1.2.3]. In 1956, the first edition of the Reactor Shielding Design Manual [1.2.4] was published and it contained a section on Boral and its properties.

In the research and test reactors built during the 1950s and 1960s, Boral was frequently the material of choice for control blades, thermal-column shutters, and other items requiring very good thermal-neutron absorption properties. It is in these reactors that Boral has seen its longest service in environments comparable to today's applications.

Boral found other uses in the 1960s, one of which was a neutron poison material in baskets used in the shipment of irradiated, enriched fuel rods from Canada's Chalk River laboratories to Savannah River. Use of Boral in shipping containers continues, with Boral serving as the poison in current British Nuclear Fuels Limited casks and the recently licensed Storable Transport Cask by Nuclear Assurance Corporation [1.2.5].

As indicated in Tables 1.2.3-1.2.5, Boral has been licensed by the NRC for use in numerous BWR and PWR spent fuel storage racks and has been extensively used in international nuclear installations.

Boral has been exclusively used in fuel storage applications in recent years. Its use in spent fuel pools as the neutron absorbing material can be attributed to its proven performance and several unique characteristics, such as:

- The content and placement of boron carbide provides a very high removal cross section for thermal neutrons.
- Boron carbide, in the form of fine particles, is homogeneously dispersed throughout the central layer of the Boral panels.
- The boron carbide and aluminum materials in Boral do not degrade as a result of long-term exposure to radiation.
- The neutron absorbing central layer of Boral is clad with permanently bonded surfaces of aluminum.
- Boral is stable, strong, durable, and corrosion resistant.

Boral absorbs thermal neutrons without physical change or degradation of any sort from the anticipated exposure to gamma radiation and heat. The material does not suffer loss of neutron attenuation capability when exposed to high levels of radiation dose.

Holtec International's QA Program ensures that Boral is manufactured under the control and surveillance of a Quality Assurance/Quality Control Program that conforms to the requirements of 10CFR72, Subpart G. Holtec International has procured over 200,000 panels of Boral from AAR Advanced Structures in over 30 projects. Boral has always been purchased with a minimum ^{10}B loading requirement. Coupons extracted from production runs were tested using the wet chemistry procedure. The actual ^{10}B loading, out of thousands of coupons tested, has never been found to fall below the design specification. The size of this coupon database is sufficient to provide reasonable assurance that all future Boral procurements will continue to yield Boral with full compliance with the stipulated minimum loading. Furthermore, the surveillance, coupon testing, and material tracking processes which have so effectively controlled the quality of Boral are expected to continue to yield Boral of similar quality in the future. Nevertheless, to add another layer of insurance, only 75% ^{10}B credit of the fixed neutron absorber is assumed in the criticality analysis in compliance with Chapter 6.0, IV, 4.c of NUREG-1536, Standard Review Plan for Dry Cask Storage Systems.

1.2.1.3.2 Neutron Shielding

The specification of the HI-STORM overpack and HI-TRAC transfer cask neutron shield material is predicated on functional performance criteria. These criteria are:

- Attenuation of neutron radiation to appropriate levels;
- Durability of the shielding material under normal conditions, in terms of thermal, chemical, mechanical, and radiation environments;
- Stability of the homogeneous nature of the shielding material matrix;
- Stability of the shielding material in mechanical or thermal accident conditions to the desired performance levels; and
- Predictability of the manufacturing process under adequate procedural control to yield an in-place neutron shield of desired function and uniformity.

Other aspects of a shielding material, such as ease of handling and prior nuclear industry use, are also considered, within the limitations of the main criteria. Final specification of a shield material is a result of optimizing the material properties with respect to the main criteria, along with the design of the shield system, to achieve the desired shielding results.

Neutron attenuation in the HI-STORM overpack is provided by the thick walls of concrete contained in the steel vessel, lid, and pedestal. Concrete is a shielding material with a long proven history in the nuclear industry. The concrete composition has been specified to ensure its continued integrity at the long term temperatures required for SNF storage.

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) will be added to reduce the freezing point for low temperature operations (e.g., below 32°F) [1.2.7].

Neutron shielding in the 125 ton HI-TRAC transfer cask in the axial direction is provided by Holtite-A within the top lid and transfer lid. Holtite-A is a poured-in-place solid borated synthetic neutron-absorbing polymer. ~~commercially available under the trade name NS-4-FR (or equivalent) and will be~~ *Holtite-A is* specified with a ~~*minimum*~~ *nominal* B_4C loading of 1 weight percent for the HI-STORM 100 System. Appendix 1.B provides the Holtite-A material properties *germane to its function as a neutron shield.* Holtec has performed confirmatory qualification tests on Holtite-A under the company's QA program.

In the following, a brief summary of the performance characteristics and properties of Holtite-A is provided.

Density

The specific gravity of Holtite-A is 1.68 g/cm^3 as specified in Appendix 1.B. To conservatively bound any potential weight loss at the design temperature and any inability to reach the theoretical density, the density is reduced by 4% to 1.61 g/cm^3. The density used for the shielding analysis is conservatively assumed to be 1.61 g/cm^3 to underestimate the shielding capabilities of the neutron shield.

Hydrogen

The weight concentration of hydrogen is 6.0%. However, all shielding analyses conservatively assume 5.9% hydrogen by weight in the calculations.

Boron Carbide

Boron carbide dispersed within Holtite-A in finely dispersed powder form is present in 1% (~~*minimum*~~ *nominal*) weight concentration. Holtite-A may be specified with a B_4C content of up to 6.5 weight percent. For the HI-STORM 100 System, Holtite-A is specified with a ~~*minimum*~~ *nominal* B_4C weight percent of 1%.

Design Temperature

The design temperature of Holtite-A is set at 300°F. The maximum spatial temperature of Holtite-A under all normal operating conditions must be demonstrated to be below this design temperature.

Thermal Conductivity

~~Table 1.B.1 lists the thermal conductivity of Holtite-A specified by the manufacturer.~~

The Holtite-A neutron shielding material is stable below the design temperature for the long term and provides excellent shielding properties for neutrons. *A conservative, lower bound conductivity is stipulated for use in the thermal analyses of Chapter 4 (Section 4.2) based on* *information in the technical literature.* ~~Technical papers provided in Appendix 1.B validate the neutron shield material's long-term stability within the design temperature and the material's ability to resist the effects of a fire accident. Holtite-A has been utilized in similar applications and has been licensed for use in a transportation cask under Docket No. 71-9235 and for storage in the HI-STAR 100 overpack under Docket No. 72-1008.~~

1.2.1.3.3 Gamma Shielding Material

For gamma shielding, the HI-STORM 100 storage overpack primarily relies on massive concrete sections contained in a robust steel vessel. A carbon steel plate, the shield shell, is located adjacent to the overpack inner shell to provide additional gamma shielding (Figure 1.2.7). Carbon steel supplements the concrete gamma shielding in most portions of the storage overpack, most notably the baseplate and the lid. To reduce the radiation streaming through the overpack air inlets and outlets, gamma shield cross plates are installed in the ducts (Figure 1.2.8) to scatter the radiation. This scattering acts to significantly reduce the local dose rates adjacent to the overpack air inlets and outlets.

In the HI-TRAC transfer cask, the primary gamma shielding is provided by lead. As in the storage overpack, carbon steel supplements the lead gamma shielding of the HI-TRAC transfer cask.

1.2.1.4 Lifting Devices

Lifting of the HI-STORM 100 System may be accomplished either by attachment at the top of the storage overpack ("top lift"), as would typically be done with a crane, or by attachment at the bottom ("bottom lift"), as would be effected by a number of lifting/handling devices.

For a top lift, the storage overpack is equipped with four threaded anchor blocks arranged circumferentially around the overpack. These anchor blocks are used for overpack lifting as well as securing the overpack lid to the overpack body. The anchor blocks are integrally welded to the overpack radial plates which in turn are full-length welded to the overpack inner shell, outer shell, and baseplate. Studs are threaded into the anchor blocks to secure the lid and provide for lifting. These four studs provide for direct attachment of lifting devices which, along with a specially-designed lift rig to ensure a vertical lift, allow lifting by a crane or similar equipment. The lift rig shall be designed to lift a fully-loaded storage overpack with margins of safety specified in ANSI N14.6 [1.2.9].

A bottom lift of the HI-STORM 100 storage overpack is effected by the insertion of four hydraulic jacks underneath the inlet vent horizontal plates (Figure 1.2.1). A slot in the overpack baseplate allows the hydraulic jacks to be placed underneath the inlet vent horizontal plate. The hydraulic jacks lift the loaded overpack to a sufficient height to allow air pads to be placed or removed from under the overpack baseplate.

The HI-TRAC transfer cask is equipped with two lifting trunnions and two pocket trunnions.

The lifting trunnions are positioned just below the top forging. The two pocket trunnions are located above the bottom forging and attached to the outer shell. The pocket trunnions are designed to allow rotation of the HI-TRAC. All trunnions are built from a high strength alloy with proven corrosion and non-galling characteristics. The lifting trunnions are designed in accordance with NUREG-0612 and ANSI N14.6. The lifting trunnions are installed by threading into tapped holes just below the top forging. ~~The lifting trunnions feature a locking plate, which is placed onto the trunnion shaft and bolted to the HI-TRAC external surface to prevent the lifting trunnion from backing out.~~

The top of the MPC lid is equipped with four threaded holes that allow lifting of the loaded MPC. These holes allow the loaded MPC to be raised/lowered through the HI-TRAC transfer cask using lifting cleats. The threaded holes in the MPC lid are designed in accordance with NUREG-0612 and ANSI N14.6.

1.2.1.5 Design Life

The design life of the HI-STORM 100 System is 40 years. This is accomplished by using material of construction with a long proven history in the nuclear industry and specifying materials known to withstand their operating environments with little to no degradation. A maintenance program, as specified in Chapter 9, is also implemented to ensure the HI-STORM 100 System will exceed its design life of 40 years. The design considerations that assure the HI-STORM 100 System performs as designed throughout the service life include the following:

HI-STORM Overpack and HI-TRAC Transfer Cask

- Exposure to Environmental Effects
- Material Degradation
- Maintenance and Inspection Provisions

MPC

- Corrosion
- Structural Fatigue Effects
- Maintenance of Helium Atmosphere
- Allowable Fuel Cladding Temperatures
- Neutron Absorber Boron Depletion

The adequacy of the HI-STORM 100 System for its design life is discussed in Sections 3.4.11 and 3.4.12.

1.2.2 Operational Characteristics

1.2.2.1 Design Features

The HI-STORM 100 System incorporates some unique design improvements. These design innovations have been developed to facilitate the safe long term storage of SNF. Some of the design originality is discussed in Subsection 1.2.1 and below.

The free volume of the MPCs is inerted with 99.995% pure helium gas during the spent nuclear fuel loading operations. Table 1.2.2 specifies the helium fill ~~*mass*~~ *pressure* to be placed in the MPC internal cavity. ~~*as a function of the free space. As the fill pressure is highly dependent on the MPC internal temperature, which increases because of the decay heat and the vacuum drying process, it is more accurate to measure the mass placed in the MPC internal cavity rather than pressure.*~~

The HI-STORM overpack has been designed to synergistically combine the benefits of steel and concrete. The steel-concrete-steel construction of the HI-STORM overpack provides ease of fabrication, increased strength, and an optimal radiation shielding arrangement. The concrete is primarily provided for radiation shielding and the steel is primarily provided for structural functions.

The strength of concrete in tension and shear is conservatively neglected. Only the compressive strength of the concrete is accounted for in the analyses.

The criticality control features of the HI-STORM 100 are designed to maintain the neutron multiplication factor k-effective (including uncertainties and calculational bias) at less than 0.95 under all normal, off-normal, and accident conditions of storage as analyzed in Chapter 6. This level of conservatism and safety margins is maintained, while providing the highest storage capacity.

1.2.2.2 Sequence of Operations

Table 1.2.6 provides the basic sequence of operations necessary to defuel a spent fuel pool using the HI-STORM 100 System. The detailed sequence of steps for storage-related loading and handling operations is provided in Chapter 8 and is supported by the Design Drawings in Section 1.5. A summary of the *general actions needed for the* loading and unloading operations is provided below. Figures 1.2.16 and 1.2.17 provide a pictorial view of typical loading and unloading operations, respectively.

Loading Operations

At the start of loading operations, the HI-TRAC transfer cask is configured with the pool lid installed†. The HI-TRAC water jacket is filled with demineralized water or a 25% ethylene glycol solution depending on the ambient temperature conditions. The lift yoke is used to position HI-TRAC in the designated preparation area or setdown area for HI-TRAC inspection and MPC insertion. The annulus is filled with plant demineralized water, and an inflatable annulus seal is installed. The inflatable seal prevents contact between spent fuel pool water and the MPC shell reducing the possibility of contaminating the outer surfaces of the MPC. The MPC is then filled with ~~spent fuel pool~~ water. *Based on the MPC model and fuel enrichment (as required by the CoC), this may be borated water* or plant demineralized water. HI-TRAC and the MPC are lowered into the spent fuel pool for fuel loading using the lift yoke. Pre-selected assemblies are loaded into the MPC and a visual verification of the assembly identification is performed.

While still underwater, a thick shielding lid (the MPC lid) is installed. The lift yoke is remotely engaged to the HI-TRAC lifting trunnions and is used to lift the HI-TRAC close to the spent fuel pool surface. As an ALARA measure, dose rates are measured on the top of the HI-TRAC and MPC prior to removal from the pool to check for activated debris on the top surface. The MPC lift bolts (securing the MPC lid to the lift yoke) are removed. As HI-TRAC is removed from the spent fuel pool, the lift yoke and HI-TRAC are sprayed with demineralized water to help remove contamination.

HI-TRAC is removed from the pool and placed in the designated preparation area. The top surfaces of the MPC lid and the upper flange of HI-TRAC are decontaminated. The inflatable annulus seal is removed, and an annulus shield is installed. The annulus shield provides additional personnel shielding at the top of the annulus and also prevents small items from being dropped into the annulus. Dose rates are measured at the MPC lid and around the mid-height circumference of HI-TRAC to ensure that the dose rates are within expected values. The Automated Welding System baseplate shield *(if used)* is installed to reduce dose rates around the top of the cask. The MPC water level is lowered slightly and the MPC lid is seal-welded using the Automated Welding System (AWS) *or other approved welding process*. Liquid penetrant examinations are performed on the root and final passes. A *multi-layer liquid penetrant or* volumetric examination is also performed on the MPC lid-to-shell weld. The water level is raised to the top of the MPC and the weld is hydrostatically tested. Then a small volume of the water is displaced with helium gas. The helium gas is used for leakage testing. A helium leakage rate test is performed on the MPC lid confinement weld (lid-to-shell) to verify weld integrity and to ensure that ~~required~~ leakage rates are within acceptance criteria. ~~The water level is raised to the top of the MPC again and then the~~ *The* MPC water is displaced from the MPC by blowing pressurized helium or nitrogen gas into the vent port of the MPC, thus displacing the water

† *This text discusses the use of separate pool and transfer lids. Users may, with appropriate design features and crane capacity, use only one lid. See Chapter 8 for additional details on applicable requirements for the single-lid option.*

through the drain line. ~~*The volume of water displaced from the MPC is measured to determine the free volume inside the MPC. This information is used to determine the helium backfill requirements for the MPC.*~~

The Vacuum Drying System (VDS) is connected to the MPC and is used to remove all liquid water from the MPC in a stepped evacuation process. The stepped evacuation process is used to preclude the formation of ice in the MPC and Vacuum Drying System lines. The internal pressure is reduced and held for a duration to ensure that all liquid water has evaporated.

Following this dryness test, the VDS is disconnected and the Helium Backfill System (HBS) is attached and the MPC is backfilled with a predetermined ~~*amount*~~ *pressure* of helium gas. The helium backfill ensures adequate heat transfer during storage, provides an inert atmosphere for long-term fuel integrity, and provides the means of 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. 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, aligned, tacked in place, and seal welded, providing redundant closure of the MPC lid and cover plates confinement 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 is removed and the remaining water in the annulus is drained. The AWS Baseplate shield is removed. The MPC lid and accessible areas of the top of the MPC shell are smeared for removable contamination and HI-TRAC dose rates are measured. The HI-TRAC top lid is installed and the bolts are torqued. The MPC lift cleats are installed on the MPC lid. The MPC lift cleats are the primary lifting point of the MPC. Two cleats provide redundant support of the MPC when it is lifted or supported.

Two or four stays (depending on the site crane hook configuration) are installed between the MPC lift cleats and the lift yoke main pins. The stays secure the MPC within HI-TRAC while the pool lid is replaced with the transfer lid. The HI-TRAC is manipulated to replace the pool lid with the transfer lid. The MPC lift cleats and stays support the MPC during the transfer operations.

MPC transfer from the HI-TRAC transfer cask into the overpack may be performed inside or outside the fuel building. Similarly, HI-TRAC and HI-STORM may be transferred to the ISFSI in several different ways. The loaded HI-TRAC may be handled in the vertical or horizontal orientation. The loaded HI-STORM can only be handled vertically.

For MPC transfers inside the fuel building, the empty HI-STORM overpack is inspected and positioned in the truck bay with the lid removed and, *for the HI-STORM 100 overpack,* the vent duct shield inserts installed. The loaded HI-TRAC is placed using the fuel building crane on top of HI-STORM. Alignment pins help guide HI-TRAC during this operation.

After the HI-TRAC is positioned atop the HI-STORM, the MPC is raised slightly. The transfer lid door locking pins are removed and the doors are opened. The MPC is lowered into HI-STORM. Following verification that the MPC is fully lowered, slings are disconnected and lowered onto the MPC lid. *For the HI-STORM 100*, the doors are closed and the locking pins are installed. HI-TRAC is removed from on top of HI-STORM along with the vent shield inserts. *For the HI-STORM 100S, the HI-TRAC is lifted above the overpack to a height sufficient to allow closure of the transfer lid doors without interfering with the MPC lift cleats. The HI-TRAC is then removed and placed in its designated storage location.* The MPC lift cleats and slings are removed from atop the MPC.

For the HI-STORM *100, the overpack* lid is installed, and the upper vent screens and gamma shield cross plates are installed. The HI-STORM lid studs *and nuts* are installed and torqued. *For the HI-STORM 100S, if the temporary or permanent lid is used, the lid is installed and the appropriate studs and nuts are installed and torqued. After the overpack has left the Part 50 facility, the permanent overpack lid is installed and the permanent studs and nuts are installed and torqued. Upper vent screens and gamma shield cross plates are installed. As plant-specific needs dictate, the loaded HI-STORM 100 or 100S overpack may be moved into or out of the Part 50 facility without the temporary or permanent lid installed. When moving the overpack to the ISFSI, the permanent lid should be installed as soon as practicable after the loaded overpack has left the Part 50 facility.*

For MPC transfers outside of the fuel building, the empty HI-STORM overpack is inspected and positioned in the cask transfer facility with the lid removed and, *for the HI-STORM 100*, the vent duct shield inserts installed. The loaded HI-TRAC is transported to the cask transfer facility in the vertical or horizontal orientation. A number of methods may be utilized as long as the handling limitations prescribed in the technical specifications are not exceeded.

To place the loaded HI-TRAC in a horizontal orientation, a transport frame or "cradle" is utilized. The cradle is equipped with rotation trunnions which engage the HI-TRAC pocket trunnions. While the loaded HI-TRAC is lifted by the lifting trunnions, the HI-TRAC is lowered onto the cradle rotation trunnions. Then, the crane lowers and the HI-TRAC pivots around the pocket trunnions and is placed in the horizontal position in the cradle.

If the loaded HI-TRAC is transferred to the cask transfer facility in the horizontal orientation, the HI-TRAC and cradle are placed on a transport vehicle. The transport vehicle may be an air pad, railcar, heavy-haul trailer, dolly, etc. If the loaded HI-TRAC is transferred to the cask transfer facility in the vertical orientation, the HI-TRAC may be lifted by the lifting trunnions or seated on the transport vehicle. During the transport of the loaded HI-TRAC, standard plant heavy load handling practices shall be applied including administrative controls for the travel path and tie-down mechanisms.

After the loaded HI-TRAC arrives at the cask transfer facility, the HI-TRAC is upended by a crane if the HI-TRAC is in a horizontal orientation. The loaded HI-TRAC is then placed, using the crane located in the transfer area, on top of HI-STORM. Alignment pins help guide HI-TRAC during this operation.

After the HI-TRAC is positioned atop the HI-STORM, the MPC is raised slightly. The transfer lid door locking pins are removed and the doors are opened. The MPC is lowered into HI-STORM. Following verification that the MPC is fully lowered, slings are disconnected and lowered onto the MPC lid. *For the HI-STORM 100,* the doors are closed and the locking pins are installed. HI-TRAC is removed from on top of HI-STORM along with the vent duct shield inserts. *For the HI-STORM 100S, the HI-TRAC is lifted above the overpack to a height sufficient to allow closure of the transfer lid doors without interfering with the MPC lift cleats. The HI-TRAC is then removed and placed in its designated storage location.* The MPC lift cleats and slings are removed from atop the MPC. The HI-STORM lid is installed, and the upper vent screens and gamma shield cross plates are installed. The HI-STORM lid studs *and nuts* are installed and torqued.

After the HI-STORM has been loaded either within the fuel building or at a dedicated cask transfer facility, the HI-STORM is then moved to its designated position on the ISFSI pad. The HI-STORM overpack may be moved using a number of methods as long as the handling limitations listed in the technical specifications are not exceeded. The loaded HI-STORM must be handled in the vertical orientation. However, the loaded overpack may be lifted from the top through the lid studs or from the bottom by the inlet vents. After the loaded HI-STORM is lifted, it may be placed on a transport mechanism or continue to be lifted by the lid studs and transported to the storage location. The transport mechanism may be an air pad, crawler, railcar, heavy-haul trailer, dolly, etc. During the transport of the loaded HI-STORM, standard plant heavy load handling practices shall be applied including administrative controls for the travel path and tie-down mechanisms. Once in position at the storage pad, vent operability testing is performed to ensure that the system is functioning within its design parameters.

[redacted]

Unloading Operations

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 HI-TRAC and empty the MPC. Special precautions are outlined to ensure personnel safety during the unloading operations, and to prevent the risk of MPC overpressurization and thermal shock to the stored spent fuel assemblies.

The MPC is recovered from HI-STORM either at the cask transfer facility or the fuel building using any of the methodologies described in Section 8.1. *If it hasn't already been removed prior* *to entering the Part 50 facility,* the HI-STORM lid is removed *and, for the HI-STORM 100,* the vent duct shield inserts are installed. The MPC lift cleats are attached to the MPC and the MPC lift slings are attached to the MPC lift cleats. *For the HI-STORM 100S, the transfer doors are opened to avoid interfering with the MPC lift cleats.* HI-TRAC is raised and positioned on top of HI-STORM. The MPC is raised into HI-TRAC. Once the MPC is raised into HI-TRAC, the HI-TRAC transfer lid doors are closed and the locking pins are installed. HI-TRAC is removed from on top of HI-STORM.

The HI-TRAC is brought into the fuel building and manipulated for bottom lid replacement. The transfer lid is replaced with the pool lid. The MPC lift cleats and stays support the MPC during the transfer operations.

HI-TRAC and its enclosed MPC are returned to the designated preparation area and the MPC stays, MPC lift cleats, and HI-TRAC top lid are removed. The annulus is filled with plant demineralized water. The annulus shield is installed *and pressurized* to protect the annulus from debris produced from the lid removal process. Similarly, HI-TRAC top surfaces are covered with a protective fire-retarding blanket.

The MPC closure ring and vent and drain port cover plates are core drilled. Local ventilation is established around the MPC ports. The RVOAs are attached to the vent and drain port. The RVOAs allow access to the inner cavity of the MPC, while providing a hermetic seal. The MPC is cooled using a closed-loop heat exchanger to reduce the MPC internal temperature to allow water flooding. Following the fuel cool-down, the MPC is flooded with *borated or unborated* water *in accordance with the CoC*. The -MPC lid-to-MPC shell weld is removed. Then, all weld removal equipment is removed with the MPC lid left in place.

~~*The inflatable annulus seal is installed and pressurized*~~. The MPC lid is rigged to the lift yoke and the lift yoke is engaged to HI-TRAC lifting trunnions. If weight limitations require, the neutron shield jacket is drained. HI-TRAC is placed in the spent fuel pool and the MPC lid is removed. All fuel assemblies are returned to the spent fuel storage racks and the MPC fuel cells are vacuumed to remove any assembly debris. HI-TRAC and MPC are returned to the designated preparation area where the MPC water is *removed* ~~*pumped back into the spent fuel pool*~~. The annulus water is drained and the MPC and HI-TRAC are decontaminated in preparation for re-utilization.

1.2.2.3 Identification of Subjects for Safety and Reliability Analysis

1.2.2.3.1 Criticality Prevention

Criticality is controlled by geometry and neutron absorbing materials in the fuel basket. The MPC-24, *MPC-24E, and 24EF(all with lower enriched fuel)* and *the* MPC-68 do not rely on soluble boron credit during loading or the assurance that water cannot enter the MPC during storage to meet the stipulated criticality limits.

The MPC-68, *MPC-68FF, MPC-24E,MPC-24EF, and MPC-32* basket*s are* ~~is~~ equipped with Boral with a minimum ^{10}B areal density of 0.0372 g/cm^2. The MPC-24 basket is equipped with Boral with a minimum ^{10}B areal density of 0.0267 g/cm^2. Due to the lower reactivity of the fuel to be stored in the MPC-68F as specified ~~by the Technical Specifications in Chapter 12~~ *in Appendix B to the CoC*, the MPC-68F is equipped with Boral with a minimum ^{10}B areal density of 0.01 g/cm^2.

The MPC-24, MPC-24E and 24EF(all with higher enriched fuel) and the MPC-32 take credit for soluble boron in the MPC water for criticality prevention during wet loading and unloading operations. Boron credit is only necessary for these PWR MPCs during loading and unloading operations that take place under water. During storage, with the MPC cavity dry and sealed from the environment, criticality control measures beyond the fixed neutron poisons affixed to the storage cell walls are not necessary because of the low reactivity of the fuel in the dry, helium filled canister and the total assurance that no water can intrude into the canister during storage.

1.2.2.3.2 Chemical Safety

There are no chemical safety hazards associated with operations of the HI-STORM 100 dry storage system. A detailed evaluation is provided in Section 3.4.

1.2.2.3.3 Operation Shutdown Modes

The HI-STORM 100 System is totally passive and consequently, operation shutdown modes are unnecessary. Guidance is provided in Chapter 8, which outlines the HI-STORM 100 unloading procedures, and Chapter 11, which outlines the corrective course of action in the wake of postulated accidents.

1.2.2.3.4 Instrumentation

As stated earlier, the HI-STORM 100 confinement boundary is the MPC, which is seal welded and leak tested. The HI-STORM 100 is a completely passive system with appropriate margins of safety; therefore, it is not necessary to deploy any instrumentation to monitor the cask in the storage mode. At the option of the user, ~~a thermocouple~~ *temperature elements* may be utilized to monitor the air temperature of the HI-STORM overpack exit vent*s* in lieu of routinely inspecting the ducts for blockage. See Subsection 2.3.3.2 and the Technical Specifications in ~~Chapter 12~~ *Appendix A to the CoC* for additional details.

1.2.2.3.5 Maintenance Technique

Because of their passive nature, the HI-STORM 100 System requires minimal maintenance over its lifetime. No special maintenance program is required. Chapter 9 describes the acceptance criteria and maintenance program set forth for the HI-STORM 100.

1.2.3 Cask Contents

The HI-STORM 100 System is designed to house different types of MPCs. The MPCs are designed to store both BWR and PWR spent nuclear fuel assemblies. Tables 1.2.1 and 1.2.2 provide key design parameters for the MPCs. A description of acceptable fuel assemblies for storage in the MPCs is provided in Section 2.1 and the ~~Technical Specifications~~ *Approved Contents section of Appendix B to the CoC. This includes fuel assemblies classified as damaged fuel assemblies and fuel debris in accordance with the definitions of these terms in the CoC. A summary of the types of fuel authorized for storage in each MPC model is provided below. All fuel assemblies must meet the fuel specifications provided in Appendix B to the CoC. All fuel assemblies classified as damaged fuel or fuel debris must be stored in damaged fuel containers.* The quantity of damaged fuel containers with fuel debris is limited to meet the off-site transportation requirements of 10CFR71, specifically, 10CFR71.63(b).

~~*At this time, failed fuel assemblies discharged from Dresden Unit 1 and Humboldt Bay reactors have been evaluated and this application requests approval of these two types of damaged fuel assemblies and fuel debris as contents for storage in the MPC-68. Damaged fuel assemblies and fuel debris shall be placed in damaged fuel containers prior to loading into the MPC to facilitate handling and contain loose components. Any combination of damaged fuel assemblies in damaged fuel containers and intact fuel assemblies, up to a total of 68, may be stored in the standard MPC-68. The MPC-68 design to store fuel debris is almost identical to the MPC-68 design to store intact or damaged fuel, the sole difference being the former requires a lower minimum B^{10} areal density in the Boral. Therefore, an MPC-68 which is to store damaged fuel containers with fuel assemblies classified as fuel debris must be designated during fabrication to ensure the proper minimum B^{10} areal density criteria is applied. To distinguish an MPC-68 which is fabricated to store damaged fuel containers with fuel assemblies classified as fuel debris, the MPC shall be designated as an "MPC-68F".*~~

~~*Up to 4 damaged fuel containers with fuel assemblies classified as fuel debris and meeting the requirements in the Technical Specifications may be stored within an MPC-68F.*~~

MPC-24

The MPC-24 is designed to accommodate up to twenty-four (24) PWR fuel assemblies classified as intact fuel assemblies, with or without non-fuel hardware.

MPC-24E

The MPC-24E is designed to accommodate up to twenty-four (24) PWR fuel assemblies, with or without non-fuel hardware. Up to four (4) fuel assemblies may be classified as damaged fuel assemblies, with the balance being classified as intact fuel assemblies. Damaged fuel assemblies must be stored in fuel storage locations 3, 6, 19, and/or 22 (see Figure 1.2.4A).

MPC-24EF

The MPC-24EF is designed to accommodate up to twenty-four (24) PWR fuel assemblies, with or without non-fuel hardware. Up to four (4) fuel assemblies may be classified as damaged fuel assemblies or fuel debris, with the balance being classified as intact fuel assemblies. Damaged fuel assemblies and fuel debris must be stored in fuel storage locations 3, 6, 19, and/or 22 (see Figure 1.2.4A).

MPC-32

The MPC-32 is designed to accommodate up to thirty-two (32) PWR fuel assemblies classified as intact fuel assemblies, with or without non-fuel hardware.

MPC-68

The MPC-68 is designed to accommodate up to sixty-eight (68) BWR intact and/or damaged fuel assemblies, with or without channels. For the Dresden Unit 1 or Humboldt Bay plants, the number of damaged fuel assemblies may be up to a total of 68. For damaged fuel assemblies from plants other than Dresden Unit 1 and Humboldt Bay, the number of damaged fuel assemblies is limited to sixteen (16) and must be stored in fuel storage locations 1, 2, 3, 8, 9, 16, 25, 34, 35, 44, 53, 60, 61, 66, 67, and/or 68 (see Figure 1.2.2).

MPC-68F

The MPC-68F is designed to accommodate up to sixty-eight (68) Dresden Unit 1 or Humboldt Bay BWR fuel assemblies (with or without channels) made up of any combination of fuel assemblies classified as intact fuel assemblies, damaged fuel assemblies, and up to eight (8) fuel assemblies classified as fuel debris.

MPC-68FF

The MPC-68FF is designed to accommodate up to sixty-eight (68) BWR fuel assemblies with or without channels. Any number of these fuel assemblies may be Dresden Unit 1 or Humboldt Bay BWR fuel assemblies classified as intact fuel, damaged fuel, or fuel debris. For BWR fuel assemblies from plants other than Dresden Unit 1 and Humboldt Bay, the total number of fuel assemblies classified as damaged fuel assemblies or fuel debris is limited to sixteen (16), with up to eight (8) of the 16 fuel assemblies classified as fuel debris. These fuel assemblies must be stored in fuel storage locations 1, 2, 3, 8, 9, 16, 25, 34, 35, 44, 53, 60, 61, 66, 67, and/or 68 (see Figure 1.2.2). The balance of the fuel storage locations may be filled with intact BWR fuel assemblies, up to a total of 68.

Table 1.2.1

KEY SYSTEM DATA FOR HI-STORM 100 SYSTEM

ITEM	QUANTITY	NOTES
Types of MPCs included in this revision of the submittal	~~3~~ *7*	~~1~~ *4* for PWR ~~2~~ *3* for BWR
MPC storage capacity†:	MPC-24 *MPC-24E* *MPC-24EF* *MPC-32*	Up to 24 intact zircaloy or stainless steel clad PWR fuel assemblies *with or without non-fuel hardware. Up to four damaged fuel assemblies may be stored in the MPC-24E and up to four (4)damaged fuel assemblies and/or fuel assemblies classified as fuel debris may be stored in the MPC-24EF* ~~*Control components and non-fuel hardware are not authorized for loading.*~~ *OR* *Up to 32 intact zircaloy or stainless steel clad PWR fuel assemblies.*
	MPC-68	Any combination of *Dresden Unit 1 or Humboldt Bay* damaged fuel assemblies in damaged fuel containers and intact fuel assemblies, up to a total of 68. ~~*in the MPC-68*~~ . *For damaged fuel other than Dresden Unit 1 and Humboldt Bay, the number of fuel assemblies is limited to 16, with the balance being intact fuel assemblies.* OR

† *See Section 1.2.3 and Appendix B to the CoC for a complete description of cask contents and fuel specifications, respectively.*

Table 1.2.1 (continued)
KEY SYSTEM DATA FOR HI-STORM 100 SYSTEM

ITEM	QUANTITY	NOTES
MPC storage capacity:	MPC-68F *MPC-68FF*	Up to 4 damaged fuel containers with zircaloy clad *Dresden Unit 1 or Humboldt Bay* BWR fuel debris and the complement damaged zircaloy clad *Dresden Unit 1 or Humboldt Bay* BWR fuel assemblies in damaged fuel containers or intact *Dresden Unit 1 or Humboldt Bay* BWR intact fuel assemblies ~~within an MPC-68F~~. *OR* *As above for Dresden Unit 1 or Humboldt Bay fuel and up to 16 damaged fuel containers containing BWR damaged fuel and/or fuel debris with the complement intact fuel assemblies, up to a total of 68. The number of damaged fuel containers containing BWR fuel debris is limited to eight (8).*

Table 1.2.2

KEY PARAMETERS FOR HI-STORM 100 MULTI-PURPOSE CANISTERS

	PWR	BWR
Pre-disposal service life (years)	40	40
Design temperature, max./min. (°F)	725°†/-40°††	725°†/-40°††
Design internal pressure (psig) Normal conditions Off-normal conditions Accident Conditions	 100 100 ~~125~~ *200*	 100 100 ~~125~~ *200*
Total heat load, max. (kW)	~~20.88~~ *27.77* (MPC-24) *28.17 (MPC-24E & MPC-24EF)* *28.74 (MPC-32)*	~~21.4~~ *28.19* (MPC-68, *MPC-68F*, & *MPC-68FF*)
Maximum permissible peak fuel cladding temperature: Normal (°F) Short Term & Accident (°F)	 See Table 2.2.3 1058°	 See Table 2.2.3 1058°
MPC internal environment Helium fill (~~g-moles/l of free space~~ *psig*)	~~0.1212~~ *29.3 – 33.3*	~~0.1218.~~ *29.3 – 33.3*
Maximum permissible multiplication factor (k_{eff}) including all uncertainties and biases	< 0.95	< 0.95
Boral ^{10}B Areal Density (g/cm^2)	0.0267 (MPC-24, *MPC-24E & MPC-24EF*) *0.0372 (MPC-32)*	0.0372 (MPC-68 & *MPC-68FF*) 0.01 (MPC-68F)

† Maximum normal condition design temperatures for the MPC fuel basket. A complete listing of design temperatures for all components is provided in Table 2.2.3.

†† Temperature based on off-normal minimum environmental temperatures specified in Section 2.2.2.2 and no fuel decay heat load.

Table 1.2.2 (continued)

KEY PARAMETERS FOR HI-STORM 100 MULTI-PURPOSE CANISTERS

End closure(s)	Welded	Welded
Fuel handling	Opening compatible with standard grapples	Opening compatible with standard grapples
Heat dissipation	Passive	Passive

Table 1.2.3

BORAL EXPERIENCE LIST
DOMESTIC PRESSURIZED WATER REACTORS

Plant	Utility
Donald C. Cook	American Electric Power
Indian Point 3	New York Power Authority
Maine Yankee	Maine Yankee Atomic Power
Salem 1,2	Public Service Electric and Gas
Sequoyah 1,2	Tennessee Valley Authority
Yankee Rowe	Yankee Atomic Power
Zion 1,2	Commonwealth Edison Company
Byron 1,2	Commonwealth Edison Company
Braidwood 1,2	Commonwealth Edison Company
Three Mile Island I	GPU Nuclear
Sequoyah (rerack)	Tennessee Valley Authority
D.C. Cook (rerack)	American Electric Power
Maine Yankee	Maine Yankee Atomic Power Company
Connecticut Yankee	Northeast Utilities Service Company
Salem Units 1 & 2 (rerack)	Public Service Electric & Gas Company

Table 1.2.4

BORAL EXPERIENCE LIST
DOMESTIC BOILING WATER REACTORS

Browns Ferry 1,2,3	Tennessee Valley Authority
Brunswick 1,2	Carolina Power & Light
Clinton	Illinois Power
Dresden 2,3	Commonwealth Edison Company
Duane Arnold Energy Center	Iowa Electric Light and Power
J.A. FitzPatrick	New York Power Authority
E.I. Hatch 1,2	Georgia Power Company
Hope Creek	Public Service Electric and Gas
Humboldt Bay	Pacific Gas and Electric Company
LaCrosse	Dairyland Power
Limerick 1,2	Philadelphia Electric Company
Monticello	Northern States Power
Peachbottom 2,3	Philadelphia Electric Company
Perry 1,2	Cleveland Electric Illuminating
Pilgrim	Boston Edison Company
Susquehanna 1,2	Pennsylvania Power & Light
Vermont Yankee	Vermont Yankee Atomic Power
Hope Creek	Public Service Electric and Gas Company
Shearon Harris Pool B	Carolina Power & Light Company
Duane Arnold	Iowa Electric Light and Power
Pilgrim	Boston Edison Company
LaSalle Unit 1	Commonwealth Edison Company
Millstone Point Unit One	Northeast Utilities Service Company

Table 1.2.5

BORAL EXPERIENCE LIST
FOREIGN PLANTS

INTERNATIONAL INSTALLATIONS USING BORAL	
COUNTRY	PLANT(S)
France	12 PWR Plants
South Africa	Koeberg 1,2
Switzerland	Beznau 1,2 Gosgen
Taiwan	Chin-Shan 1,2 Kuosheng 1,2
Mexico	Laguna Verde Units 1,2
Korea	Ulchin Units 1, 2
Brazil	Angra 1
United Kingdom	Sizewell B

Table 1.2.6

HI-STORM 100 OPERATIONS SEQUENCE

Site-specific handling and operations procedures will be prepared, reviewed, and approved by each owner/user.

1	HI-TRAC and MPC lowered into the fuel pool without lids
2	Fuel assemblies transferred into the MPC fuel basket
3	MPC lid lowered onto the MPC
4	HI-TRAC/MPC assembly moved to the decon pit and MPC lid welded in place, volumetrically or multi-layer PTexamined, hydrostatically tested, and leak tested
5	MPC dewatered, vacuum dried, backfilled with helium, and the closure ring welded
6	HI-TRAC annulus drained and external surfaces decontaminated
7	MPC lifting cleats installed and MPC weight supported by rigging
8	HI-TRAC pool lid removed and transfer lid attached
9	MPC lowered and seated on HI-TRAC transfer lid
10	HI-TRAC/MPC assembly transferred to atop HI-STORM overpack
11	MPC weight supported by rigging and transfer lid doors opened
12	MPC lowered into HI-STORM overpack, ~~*HI-TRAC transfer lid doors closed,*~~ and HI-TRAC removed from atop HI-STORM overpack
13	HI-STORM overpack lid installed and bolted in place
14	HI-STORM overpack placed in storage at the ISFSI pad

[redacted]

[redacted]

[redacted]

[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]
[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]
[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]
[redacted]	[redacted]	[redacted]	[redacted]	[redacted]	[redacted]
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† [redacted]

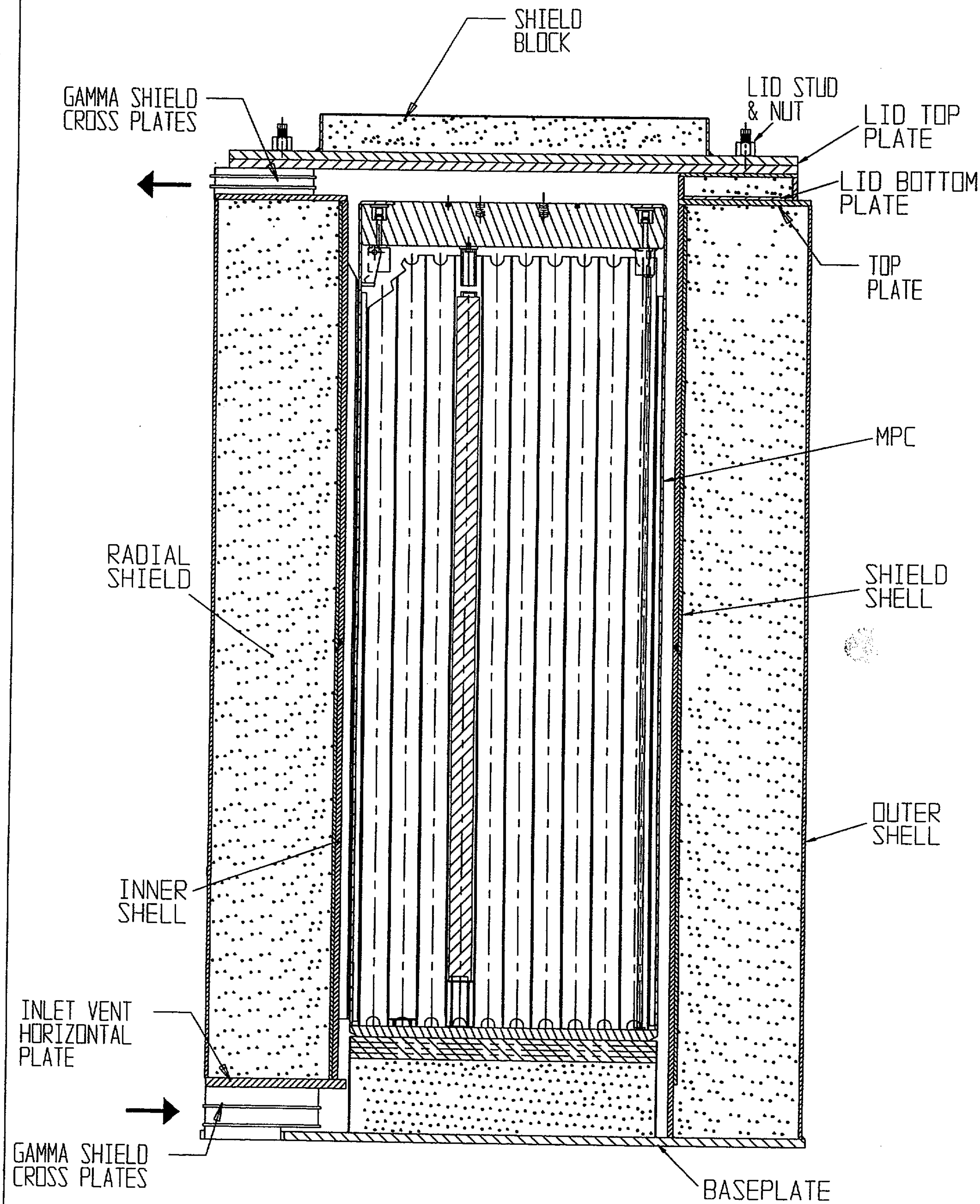

FIGURE 1.2.1A; CROSS SECTION VIEW OF THE HI-STORM 100S SYSTEM

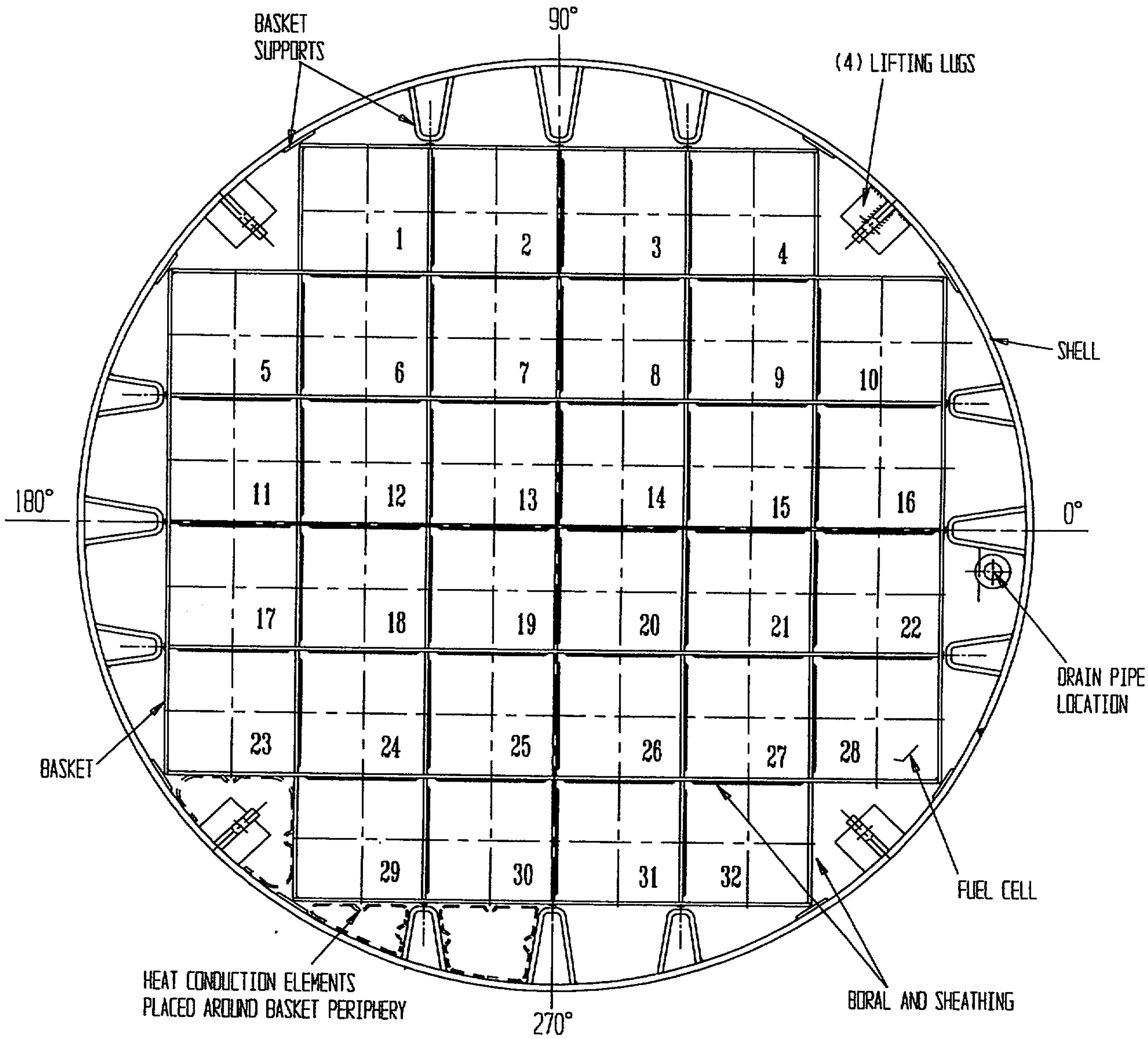

FIGURE 1.2.3; MPC-32 CROSS SECTION

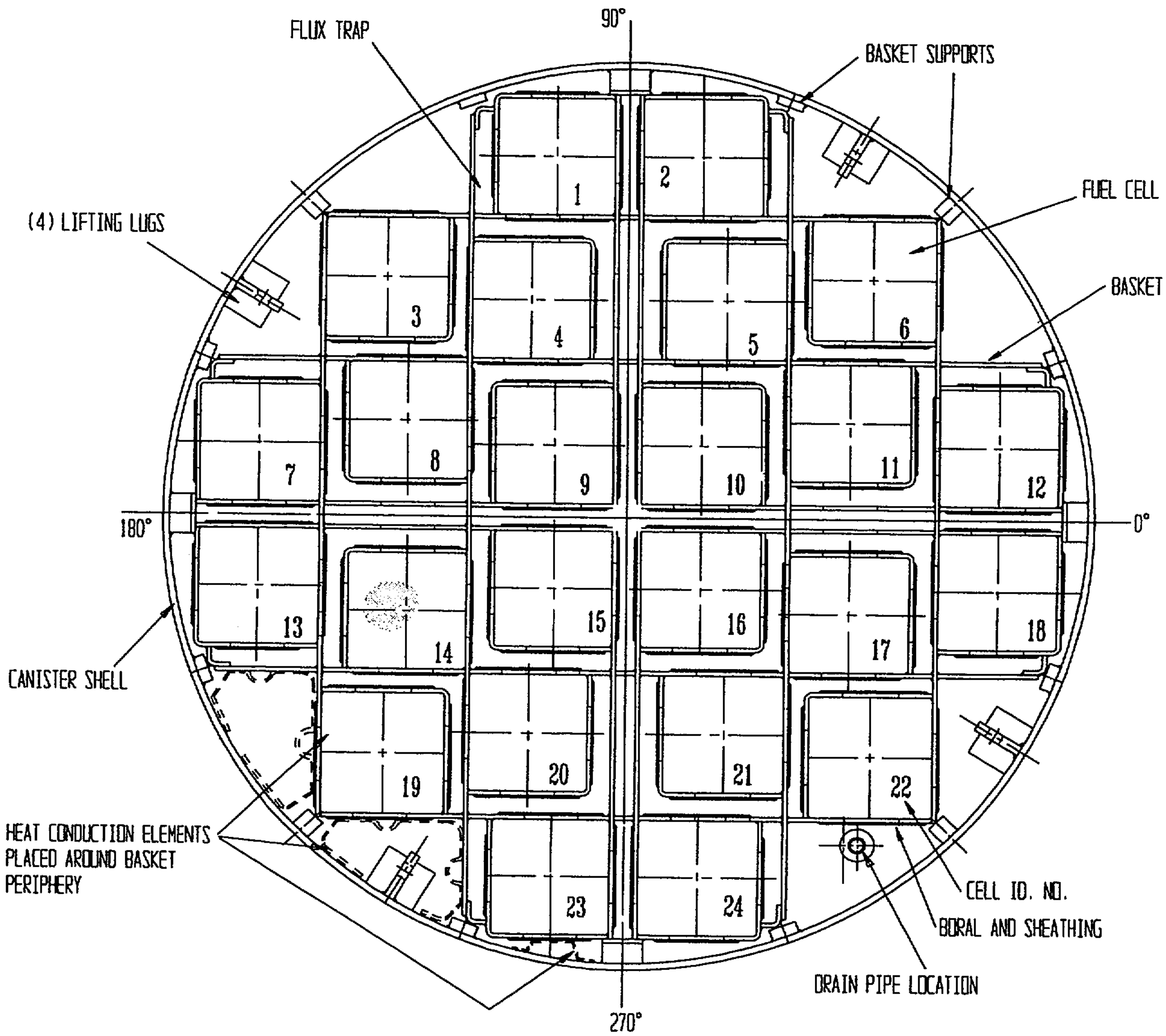

FIGURE 1.2.4A; MPC-24E/24EF CROSS SECTION VIEW

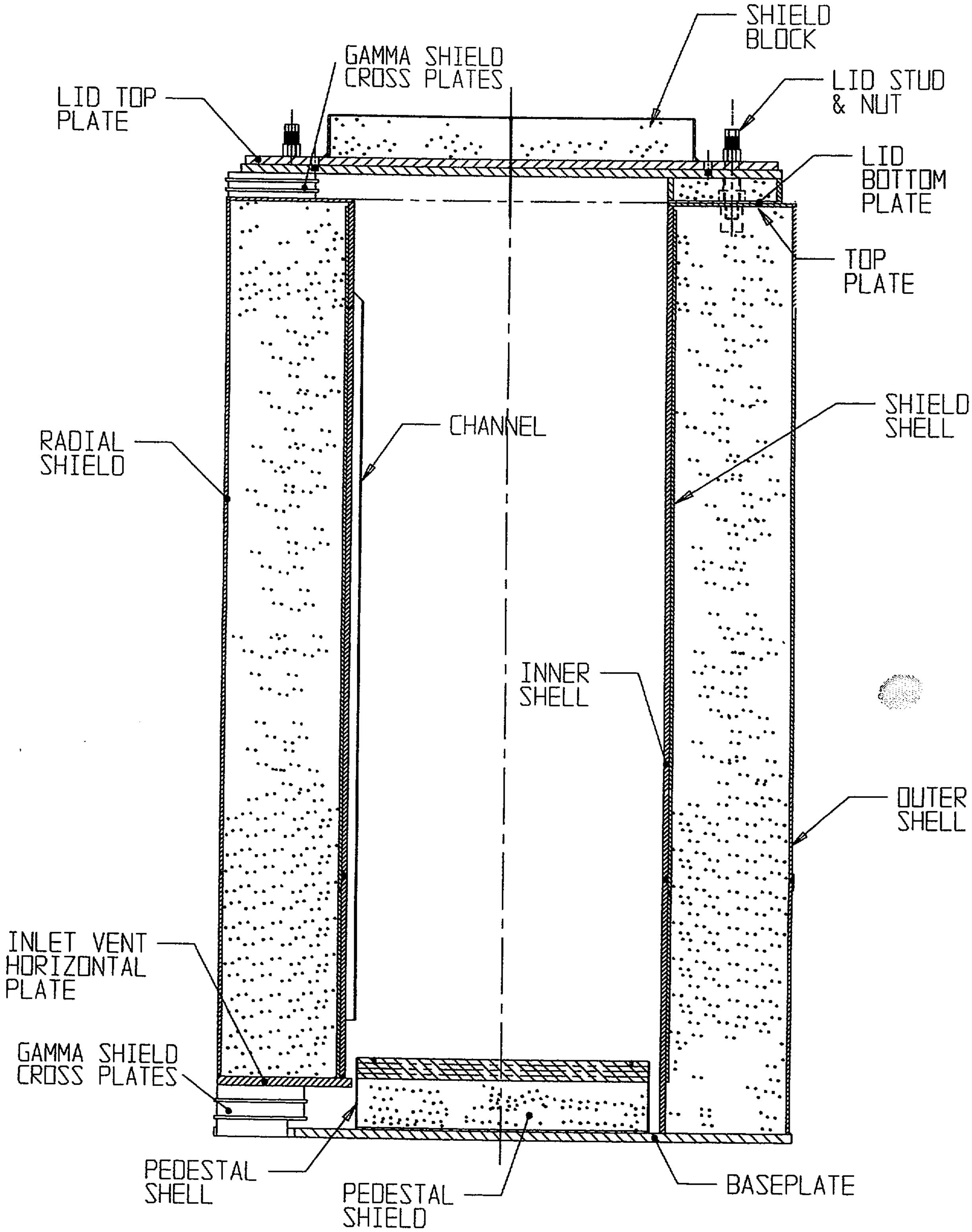

FIGURE 1.2.8A; HI-STORM 100S OVERPACK CROSS SECTIONAL ELEVATION VIEW

1.4 GENERIC CASK ARRAYS

The HI-STORM 100 System is stored in a vertical configuration. The required center-to-center spacing between the modules (layout pitch) is guided by ~~heat transfer~~ *operational* considerations. Tables 1.4.1 and 1.4.2 provide the *nominal layout pitch information.* ~~minimum pitch requirements~~. *Site-specific pitches are* determined by *practical operation with supporting* heat transfer calculations in Chapter 4. The pitch values *in Tables 1.4.1 and 1.4.2* are *nominal* ~~minimums~~ and may be *varied* ~~increased~~ to suit the user's specific needs. ~~If MPC transfer operations between the HI-TRAC transfer cask and HI-STORM overpack are to be performed on the ISFSI pad, the minimum cask pitch values may not provide sufficient clearance to perform the transfer operations. An alternative to performing the MPC transfer operations on the ISFSI pad is to provide a separate MPC transfer area at the ISFSI.~~

Table 1.4.1 provides recommended cask spacing data for array(s) of two by N casks. *T*he pitch between adjacent rows of casks ~~(P1)~~ and between each adjacent column of casks ~~(P2)~~ *are denoted by* P_1 *and* P_2 *In Table 1.4.1.* ~~shall be in accordance with Table 1.4.1.~~ There may be an unlimited number of rows. The distance between adjacent arrays of two by N casks (P3) shall be as specified in Table 1.4.1. See Figure 1.4.1 for further clarification. The pattern of required pitches and distances may be repeated for an unlimited number of columns.

For a square array of casks the pitch between adjacent casks *may* ~~shall~~ be in accordance with Table 1.4.2. See Figure 1.4.2 for further clarification. ~~The total quantity of rows and columns is unlimited provided the minimum pitch specified in Table 1.4.2 is met.~~ *The data in Table 1.4.2 provide nominal values for large ISFSIs (i.e., those with hundreds of casks in a uniform layout), where access of feed air to the centrally located casks may become a matter of thermal consideration. From a thermal standpoint, regardless of the size of the ISFSI, the casks should be arrayed in such a manner that the tributary area for each cask (open ISFSI area attributable to a cask) is a minimum of 225* ft^2. *Subsection 4.4.1.1.7 provides the detailed thermal evaluation of the required tributary area. For specific sites, a smaller tributary area can be utilized after appropriate thermal evaluations for the site-specific conditions are performed, subject to evaluation under 10 CFR 72.48.*

Table 1.4.1

CASK LAYOUT ~~MINIMUM~~ PITCH DATA FOR 2 BY N ARRAYS

Orientation	~~**Minimum**~~ ***Nominal*** **Cask Pitch (ft.)**
Between adjacent rows, P1, and adjacent columns, P2	13.5
Between adjacent sets of two columns, P3	38

Table 1.4.2

CASK LAYOUT ~~MINIMUM~~ PITCH DATA FOR SQUARE ARRAYS

Orientation	~~Minimum~~ *Nominal* Cask Pitch (ft.)
Between adjacent casks	18 - 8"

APPENDIX 1.B: HOLTITE ™ MATERIAL DATA
(Total of 20 Pages Including This Page)

The information provided in this appendix describes the neutron absorber material, Holtite-A ~~(also known commercially as NS-4-FR)~~ for the purpose of confirming its suitability for use as a neutron shield material in spent fuel storage casks. *Holtite-A is one of the family of Holtite neutron shield materials denoted by the generic name Holtite™. It is currently the only neutron shield material approved for installation in the HI-STAR 100 cask. It is chemically identical to NS-4-FR* which was originally developed by Bisco Inc. and used for many years as a shield material with B_4C or Pb added.

~~NS-4-FR~~ *Holtite-A* contains aluminum hydroxide ($Al(OH)_3$) in an epoxy resin binder. Aluminum hydroxide is also known by the industrial trade name of aluminum tri-hydrate or ATH. ATH is often used commercially as a fire-retardant. ~~hence the "FR" designation in NS-4-FR. NS-4-FR~~ *Holtite-A* contains approximately 62% ATH supported in a typical 2-part epoxy resin as a binder. Holtite-A ~~the Holtec International version of NS-4-FR,~~ contains 1% (nominal) by weight B_4C, a chemically inert material added to enhance the neutron absorption property. Pertinent properties of Holtite-A are listed in Table 1.B.1.

The essential properties of Holtite-A are:

1. the hydrogen density (needed to thermalize neutrons),

2. thermal stability of the hydrogen density, and

3. the uniformity in distribution of B_4C needed to absorb the thermalized neutrons.

ATH and the resin binder contain nearly the same hydrogen density so that the hydrogen density of the mixture is not sensitive to the proportion of ATH and resin in the ~~NS-4-FR~~ *Holtite-A* mixture. B_4C is added ~~(1% in Holtite-A)~~ as a finely divided powder and does not settle out during the resin curing process. Once the resin is cured (polymerized), the ATH and B_4C are physically retained in the hardened resin. ~~Analysis~~ *Qualification testing* for B_4C throughout a column of Holtite-A has confirmed ~~(Holtec International qualification tests)~~ that the B_4C is uniformly distributed with no evidence of settling or non-uniformity. Furthermore, an excess of B_4C is specified in *the* Holtite-A *mixing and pouring procedure* as a precaution to assure that the B_4C concentration is always adequate throughout the mixture.

~~NS-4-FR material has been extensively tested for thermal stability, as indicated in the following documents (copies of these documents are attached).~~

- ~~Letter dated 4/20/87 from Mr. Larry Dietrick, Bisco Products to Mr. Tod Lesser, NAC International, "Weight loss of NS-4-FR under extreme temperature conditions"~~

- ~~"Experimental Studies On Long-Term Thermal Degradation of Enclosed Neutron Shielding Resin", Asano, Ryoji and Nagao Niomura~~

- ~~"Thermal Testing of Solid Neutron Shielding Materials", Boonstra, Richard H.~~

~~The specific gravity specified in Table 1.B.1 does not include an allowance for weight loss. The specific gravity specified in Chapter 1 includes a 4% reduction to conservatively account for potential weight loss at the design temperature of 300°F. However,. the BISCO letter dated 4/20/87 provides information stating that samples had been exposed to a continuous temperature of 338°F for 146 days and a maximum of 3.15% weight loss had been experienced. Thus, there is a substantial level of conservatism in the Holtec allowance for weight loss.~~

The specific gravity specified in Table 1.B.1 does not include an allowance for weight loss. The specific gravity assumed in the shielding analysis includes a 4% reduction to conservatively account for potential weight loss at the design temperature of 300°F or an inability to reach theoretical density. Tests on the stability of Holtite-A were ~~also~~ performed by Holtec International. *The results of the tests are summarized in Holtec Reports HI-2002396, "Holtite-A Development History and Thermal Performance Data" and HI-2002420, "Results of Pre- and Post-Irradiation Test Measurements." The information provided in these reports demonstrates that Holtite-A™ possesses the necessary thermal and radiation stability characteristics to function as a reliable shielding material in the HI-STAR 100 overpack.*

~~The paper entitled "Experimental Studies on Long-Term Thermal Degradation of Enclosed Neutron Shielding Resin " provides information which corroborates the information provided in the BISCO letter dated 4/20/87. The paper suggests that enclosures of the NS-4-FR material can further decrease the percent weight reduction at elevated temperatures.~~ The ~~NS-4-FR~~ *Holtite-A* is encapsulated in the HI-STAR 100 overpack and, therefore, should experience a very small weight reduction during the design life of the HI-STAR 100 System. ~~It should be noted that the shielding analysis conservatively assumes 4% loss in density.~~

~~The paper entitled "Thermal Testing of Solid Neutron Shielding Materials" provides information regarding NS-4-FR material stability during a fire accident. Results of the study suggests that NS-4-FR could withstand a fire accident with minimal damage. This data is provided for information only, as the post-accident shielding analysis very conservatively assumes complete degradation of the Holtite-A neutron shield and replaces the neutron shield with a void.~~

The data and test results ~~presented here~~ confirm that *Holtite-A remains stable under design thermal and radiation conditions, the material properties meet or exceed that assumed in the shielding analysis, and the B_4C remains uniformly distributed with no evidence of settling or non-uniformity.*

~~1. Holtite-A with 1% B_4C is thermally stable. has the same thermal stability and characteristics as the previously approved NS-4-FR material;~~

~~2. The hydrogen density meets or exceeds minimum NS-4-FR specifications (measured at 0.105 gH_2/cc compared to the NS-4-FR specification of 0.096 gH_2/cc), and~~

~~3. The B_4C is uniformly distributed, with no evidence of settling or non-uniformity.~~

Based on the information described above, Holtite-A meets all of the requirements for an acceptable neutron shield material. ~~in the manner of NS-4-FR, which was licensed previously in Docket No. 71-9235 (NAC-STC).~~

Table 1.B.1

REFERENCE PROPERTIES OF HOLTITE-A NEUTRON SHIELD *MATERIAL*

PHYSICAL PROPERTIES ~~(Reference: NAC International Brochure)~~	
% ATH	62 *nominal* ~~maximum (confirmed by Holtec in independent analyses)~~
Specific Gravity	1.68 g/cc nominal
~~Thermal Conductivity~~	~~0.373 Btu/hr/ft-°F~~
Max. Continuous Operating Temperature	300°F
~~Specific Heat†~~	~~0.39 Btu/lb-°F~~
Hydrogen Density	0.096 g/cc minimum ~~(confirmed by Holtec in independent analyses)~~
Radiation Resistance	Excellent
~~Ultimate Tensile Strength~~	~~4,250 psi~~
~~Tensile elongation~~	~~0.65%~~
~~Ultimate Compression Strength~~	~~10,500 psi~~
~~Compression Yield Strength~~	~~8,780 psi~~
~~Compression Modulus~~	~~561,000 psi~~
CHEMICAL PROPERTIES (Nominal)	
wt% Aluminum	21.5 ~~(confirmed by Holtec)~~
wt% Hydrogen	6.0 ~~(confirmed by Holtec)~~
wt% Carbon	27.7
wt% Oxygen	42.8
wt% Nitrogen	2.0
wt% B_4C	~~up to 6.5 (Holtite-A uses 1% B_4C)~~ *1.0*

H ~~BISCO Products Data from Docket M-55, NAC-STC TSAR.~~

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