HOLTEC INTERNATIONAL

HI-STORM 100 CERTIFICATE OF COMPLIANCE 72-1014

LICENSE AMENDMENT REQUEST 1014-2

REVISION 1

OCTOBER, 2002

(NON-PROPRIETARY VERSION)

INFORMATION NEEDED TO REVIEW HI-STORM 100, AMENDMENT 2

Technical Specifications

1. Provide the methodology used to determine allowable surface dose rate limits in conjunction with Technical Specification 5.7, *Radiation Protection Program.*

   Page 31 of NUREG-1745, *Standard Format and Content for Technical Specifications for 10 CFR Part 72 Cask Certificates of Compliance*, states that the process for establishing a limit may be removed to an administrative program if a method of evaluation acceptable to the NRC is presented in the Safety Analysis Report (SAR). Surface dose rate limits assure proper loading and consistency with the site-specific off-site dose analysis, which in turn ensure compliance with radiological requirements in 10CFR Part 20, and 10CFR Part 72.

   A user that determines allowable surface dose limits under proposed Technical Specification 5.7, in lieu of numeric dose limits specified in the Technical Specifications, should use an NRC-approved methodology for determining allowable dose rates. The methodology should specify appropriate shielding design considerations, acceptable analytical methods, baseline inputs and assumptions, and qualifications of personnel. A methodology is required in the SAR, with a discussion of an administrative program to measure and control dose limits in order to satisfy the requirements of 10 CFR 72.104, and 72.106.

   <u>Response</u>

   There is no requirement proposed in TS 5.7, *Radiation Protection Program*, for users to determine a maximum allowable cask surface dose rate limit because neither 10 CFR 20 nor 10 CFR 72 include a cask surface dose rate limit or a requirement to establish such a limit. This balance of this response addresses each issue in the question separately.

   10 CFR 20

   Compliance with 10 CFR 20 is demonstrated by dose monitoring under the licensee's individual ALARA-based dose control program and is unrelated to any dose rate limit in the generic cask Technical Specifications (TS). This is because licensees handle a wide variety of radioactive materials as a normal part of plant operations, many of which have contact dose rates significantly higher than those of a dry storage cask. Cask operations would be treated similarly. That is, appropriate dose minimization measures would be established in the Radiation Work Permit (RWP) and procedures governing the activities taking place, based on the actual radiation dose rates measured using the "time, distance, and shielding" approach.

10 CFR 72.104

The regulations require that users comply with 10 CFR 72.104 for controlled-area boundary doses due to ISFSI operation and other uranium fuel cycle operations that may contribute additional dose (such as plant operations). Calculations performed site-specifically to demonstrate compliance with 10 CFR 72.104 (as required by 10 CFR 72.212(b)(2)(i)(C)) may use plant-specific fuel and non-fuel hardware characteristics, site boundary distances, and ISFSI arrays, or may use the bounding inputs found in the HI-STORM FSAR and CoC. However, it remains up to the user to confirm that the ultimate conclusion of the 72.104 analysis remains valid (i.e., compliance is demonstrated). For example, a particular cask may have measured surface dose rates above the calculated value, but other casks at the ISFI may have lower than calculated surface dose rates, resulting in overall compliance with the regulations.

Confirmation of Proper Cask Loading

Compliance with surface dose rate limits in the TS is not a reliable indicator of proper cask loading or consistency with the site-specific off-site dose analysis. More specifically, if a measured cask surface dose rate exceeds the cask TS limit, certainly a mis-loading has occurred. However, measuring a surface dose rate less than the limit in no way assures that all contents loaded meet the CoC requirements. This is because the actual contents of a cask loaded at a given general licensee's facility will never match the bounding design basis contents used in the licensing basis shielding analyses. Individual fuel assemblies or non-fuel hardware not meeting the CoC could be loaded with the overall effect on dose rate being insignificant. Compliant cask loading is only assured through administrative controls requiring appropriate verification and documentation of all cask contents before loading.

Shielding Methodology

The methodology used to calculate single cask and whole ISFSI dose rates that would be used in a radiation protection program is described in Chapter 5 of the approved HI-STORM 100 FSAR. We believe FSAR Chapter 5 includes sufficiently detailed descriptions of shielding methodology to ensure appropriate conservatisms in the modeling are used in the site specific calculations. The shielding methodology described in the FSAR is approved for use in evaluating proposed changes under 10 CFR 72.48 and changing the methodology itself is appropriately limited by the 10 CFR 72.48 regulation. Therefore, the shielding methodology as described in FSAR Chapter 5 and controlled through 10 CFR 72.48, should be acceptable for use in determining site-specific cask dose rate limits. FSAR Chapter

5 will continue to retain the results of the generic shielding analyses for design basis fuel and non-fuel hardware.

Proposed Changes

Notwithstanding the above arguments, based on discussions with the SFPO staff, the proposed CoC changes have been modified as follows:

a. Similar to existing requirements in CoC Appendix B, Sections 5.5.a.2 and 5.5.b.1 for user-performed structural calculations, proposed new TS 5.7 has been revised to explicitly require those who calculate transfer cask and overpack dose rates to use methodologies consistent with those in the HI-STORM 100 FSAR.

b. In order to balance the removal of the dose rate LCOs from the HI-STORM TS, proposed new TS 5.7 has been revised to include more specific methodology and input requirements applicable to anyone who performs shielding analyses or evaluations of the HI-STORM 100 System, including the evaluations required to demonstrate compliance with 10 CFR 72.104 and evaluations of changes proposed under 10 CFR 72.48. Including these methodology and input requirements directly in the TS limits the flexibility of users to change inputs and methodology even more than the 10 CFR 72.48 regulation already does.

2. Provide additional information to clarify the statement in Technical Specification 5.7, that a user may "establish a separate radiation protection program" from its 10 CFR Part 50 radiation protection program. Provide information in the SAR concerning the programmatic structure and necessary elements of an alternate radiation protection program including establishment of contamination limits, for an ISFSI using the HI-STORM 100.

<u>Response</u>

After further consideration, we have determined that this proposed change is unnecessary and has been withdrawn from the amendment request. Proposed Technical Specification 5.7 has been revised accordingly.

3. Provide the methodology used to determine if contents can satisfy the criteria listed in Technical Specification 2.3 of Appendix B, *Deviations from Cask Content Requirements*.

A user that requests a deviation in allowable cask contents should use an appropriate/approved methodology for determining whether the proposed contents satisfy criteria number one and two of the proposed technical specification. The methodology should establish acceptance criteria that define "an equivalent level of safety" for new or different contents. The methodology should also discuss limitations on important content parameters when proposing a change in cask contents (e.g., what changes trigger the need for an amendment request). The methodology should also specify appropriate design considerations and baseline inputs and assumptions that are required to determine whether proposed new contents provide "an equivalent level of safety" and satisfy applicable NRC requirements.

At a minimum, an approved NRC methodology is one which contains: in depth descriptions of acceptable analytical and computer code methods, assumptions, and validations at all burnup levels; calculational packages and complete models; qualifications demonstrating ability to utilize the NRC approved methodology; examples of the types of changes which could be made; and any limitations on methodology applicability.

<u>Response</u>

The basis for this request is consistent with the intent of the provision in Section 2.2 of NUREG-1745, "Standard Format and Content for Technical Specifications for 10 CFR Part 72 Cask Certificates of Compliance", yet is more conservative as proposed HI-STORM 100 in LAR 1014-2. Proposed HI-STORM TS 2.3, "Deviations from Cask Content Requirements" is more restrictive than the analogous provision NUREG-1745 because the list of approved contents remains the same as that currently in the CoC rather than being relocated to the FSAR.

The TS 2.3 provision is requested to avoid unnecessarily impacting a general licensee's fuel loading schedule to accommodate a CoC amendment for a non-safety significant deviation from the cask contents listed in the CoC. This provision will also eliminate the need for licensees to request exemptions from the regulations for reasons of schedule, to implement such a change. In many respects, this proposed CoC provision is the same as the existing CoC provision for gaining NRC approval of alternatives to the ASME Code.

In order to clarify this request, we provide the following information:

a. All requests for approval of cask contents deviations will be submitted to the NRC by Holtec International in accordance with proposed TS 2.3. General licensees will not be permitted to independently request approval of such deviations.

b. All requests will include a basis and justification for approval that addresses the effect of the change on the various technical bases described in the FSAR. These evaluations will be performed using the previously approved technical methodologies, as described in the FSAR and implemented in supporting calculations.

c. The justification for each request will show that the requirements proposed in TS 2.3.1 and 2.3.2 are met.

d. Each request will be applicable to one or more general licensees, specifically identified in the request, with the schedular need clearly articulated.

e. All approved deviations will be included in the next scheduled CoC amendment request as permanent changes.

Proposed TS 2.3 has been modified in LAR 1014-2, Revision 1, to clarify and add requirements consistent with the above list. The Summary of Proposed Changes in the LAR package has also been revised to clarify the justification for this requested TS change.

4. Provide a clarification of whether or not the HI-STORM 100 design is compliant with the recently revised staff guidance regarding cladding temperature limits, burnup levels, etc., contained in ISG-11, Rev. 2, *Cladding Considerations for the Transportation and Store of Spent Fuel,* dated July 30, 2002.

If the design is not in compliance with this latest staff guidance, it will be necessary to change all references of "Zircalloy (or other alloy of zirconium)" throughout the Technical Specifications and the SAR, to clearly specify which alloy is being requested ; i.e., zirc-2, zirc-4, ZIRLO, M5, OPTIN, etc. For those alloys other than zirc-2 and zirc-4, it will be necessary to provide all data regarding the mechanical, creep, and hydride properties of the specific cladding type for the design temperature/stress regimes.

<u>**Response**</u>

Holtec has reviewed the revised guidance in ISG-11, Revision 2 as it applies to both the existing approved licensing basis for the HI-STORM 100 System and the changes proposed in LAR 1014-2. We have modified the proposed changes in LAR 1014-2 and associated FSAR material to comport with the guidance in ISG-11, Rev. 2. The new, single spent fuel cladding temperature limit of 400°C has permitted us to eliminate the FSAR detail that currently describes how the permissible spent fuel cladding temperatures were computed for low burnup fuel. The changes in the guidance pertaining to peak clad temperature limits, cladding strain, and cladding oxide thickness have also allowed us to completely eliminate FSAR Appendix 4.A for high burnup fuel. The Summary of Proposed Changes has been modified to add this item as new Proposed Change Number 15a in LAR 1014-2.

Based on clarifying discussions with the NRC on ISG-11, the following positions are also incorporated into LAR 1014-2, Revision 1:

a. New definitions for the terms "commercial spent fuel", "design heat load", "long-term storage", "short-term operating conditions", "thermal capacity", "threshold heat load", and "ZR" have been added to the FSAR glossary (Table 1.0.1).

b. The permissible fuel cladding temperature limit for stainless steel clad fuel for long term storage and short-term operating condition is set at 400°C, which we understand is conservative compared to the limit currently contemplated for stainless steel clad fuel. An alternative to this limit for moderate burnup fuel is described in item 'd' below.

c. The permissible fuel cladding temperature limit for all off-normal and accident conditions remains at 570°C (1058°F).

d. HI-STORM Amendment 1 was previously licensed for MPCs with decay heat loads up to approximately 29 kW to be loaded. Vacuum drying is currently permitted as the method of MPC cavity drying up to this heat load for moderate burnup fuel ($\leq$ 45,000 MWD/MTU), based on a permissible fuel cladding temperature limit of 570°C. In order to maintain this option of MPC drying and this temperature limit, a lower threshold heat load is established whereby, for higher heat loads, a "best-estimate" calculation of fuel cladding hoop stress is required to demonstrate that the hoop stress is less than 90 MPa if fuel cladding temperatures exceed 400°C but still remain less than 570°C. Please see proposed changes to TS LCO 3.1.1, Surveillance Requirement SR 3.1.1.1 for the details of MPC cavity drying.

e. There is no longer a distinction made in the CoC among the various zirconium-based fuel cladding alloys. Consistent with ISG-11, revision 2, all zirconium-based fuel cladding materials approved for reactor operations by the NRC's Division of Nuclear Reactor Regulation are considered authorized for loading, subject to the other restrictions of the CoC.

Chapter One, General Description

Provide a drawing of the MPC-32F.

__Response__

The MPC-32F design (like its approved cousins, the MPC-24EF and MPC-68F, and MPC-68FF), differ from the base MPC-32 design __only__ in the following ways (see Proposed Change No. 2 in the LAR submittal):

a. The enclosure vessel shell is thickness is increased from ½ inch to 1 inch in the top 9 inches.

b. The MPC lid diameter is reduced by one inch to accommodate the thicker upper shell.

c. The MPC lid-to-shell weld size is increased from 3/4 -inch to 1 inch.

None of these differences affect 10 CFR 72 certification. All of these differences exist solely to address a 10 CFR 71 structural load combination where the MPC acts as the second containment barrier per 10 CFR 71.63(b) for canisters containing fuel debris. Because the MPC-32F will ultimately be dual-purpose certified under 10 CFR 71, fuel debris must be __stored__ at the ISFSI under 10 CFR 72 in an "F" model MPC to avoid re-packaging for transportation. Proposed changes to FSAR Section 2.1.3 and proposed new FSAR Figure 2.1.9 included in LAR 1014-2 explain and depict the differences between all "F" model MPCs and their standard design counterparts. Therefore, no drawing is necessary for the review. After approval of Amendment 2 to the CoC, appropriate changes to the design and licensing drawings for the MPC enclosure vessel will be made to reflect MPC-32F.

Chapter Three, Structural

1. Provide calculations similar to those found in Appendix 3.AS for the MPC-32F.

2. Provide the updated calculation package which contains the supporting analytical approaches and calculations which were previously included in Appendices 3.B through 3.AS.

__Response__

These Holtec-proprietary structural calculations are being provided under separate cover. Separate calculation packages are provided for the MPC, HI-STORM overpack, and the HI-TRAC transfer cask.

Chapter Four, Thermal

1. Provide the following supporting documentation, calculations, etc.:

    a. ANSYS analysis models in .db or .inp format which were used to obtain the bounding PWR and BWR MPC regional effective thermal properties.

    b. Provide the new Figure 4.4.27.

    c. Provide the updated calculation package which contains the supporting analytical approaches and calculations for all the thermal analyses described in Chapter 4 and Chapter 11.

    <u>Response</u>

    The Holtec-proprietary ANSYS models in .db and .inp format are being provided under separate cover on a compact disk (CD). The CD also includes a "read me" file providing instructions on how to use the model files. Figure 4.4.27 is included in Revision 1 of LAR 1014-2. The thermal calculation package is being provided under separate cover.

2. Revise Figures 4.4.16, 4.4.17, 4.4.19, 4.4.20, 4.4.26, 4.5.2, to clarify the units used to report analysis results.

    <u>Response</u>

    The figures have been revised and are included in Revision 1 of LAR 1014-2.

3. Either change the "List of Effective Pages for Proposed FSAR Revision 2" to reflect that Figures 4.4.16, 4.4.17, 4.4.19, 4.4.20, 4.4.26, are of "Revision 1," or submit the updated figures. Currently, the list shows the above figures as "Revision 2."

    <u>Response</u>

    The List of Effective Pages is correct. The missing figures are now included in Revision 1 of LAR 1014-2.

Chapter Five, Shielding

Provide a full description of the "vibration suppressor inserts," and clearly describe the differences and similarities between these components with burnable poison rod assemblies.

<u>Response</u>

Vibration suppressor inserts, on the whole, are essentially identical to BPRAs. The top portion of the inserts is identical to a BPRA. They are handled in the same manner as a BPRA. The vibration suppressor inserts contain long rodlets similar to BPRAs, with the main difference being that the rodlets in the vibration suppressor inserts do not contain poison materials (i.e., the rodlets may be made of Zircaloy). Therefore, as far as the analysis in the FSAR is concerned, the vibration suppressor inserts are considered to be BPRAs and must meet the burnup and cooling time limits for BPRAs in the CoC.

Additional clarifying text has been added to the end of the first paragraph in the proposed changes to FSAR Section 5.2.4.1.

Chapter Six, Criticality

1. Provide the locations of the missing rods for the MPC-32 damaged fuel analysis as was done in Figures 6.4.2 to 8 for the MPC-24.

   <u>Response</u>

   FSAR Figures 6.4.2 through 6.4.8 show the missing fuel rod patterns used in the MPC-68 analysis, as described in the titles of these figures. Damaged fuel is not permitted to be loaded in the MPC-24, so no such analyses are performed. For the MPC-24E, missing fuel rod patterns were analyzed, and the results are included in FSAR Figure 6.4.14. However, since the bounding model for damaged fuel/fuel debris (bare fuel pellets) results in significantly higher reactivities than missing rods, the missing rod patterns were not included in the chapter for MPC-24E. For the MPC-32, only the bounding bare fuel rod model is used and no missing fuel rod patterns were analyzed, since the calculations for the MPC-24E show that the bare fuel model is bounding. The main difference between the MPC-24E and the MPC-32 that could affect the results of the analyses is the soluble boron concentration, which is much higher for the MPC-32 due primarily to the absence of flux traps in the basket design. To show that, even for the high soluble boron concentration, the optimum moderation occurs in the MPC-32, a study has been performed for a variation in water density inside the DFC and the results are shown in FSAR Table 6.4.13.

Calculation packages are being provided separately that provide additional information about the damaged fuel analysis approach. For the MPC-24E in fresh water, this information is located in Report HI-951321, Attachment F, and for borated water in HI-2012771, Appendix D. For the MPC-32, the information is located in HI-2012771, Appendix C.

2. Provide sample input files for the MCNP runs of the MPC-32F, which describe how the fuel debris was modeled. Provide a sketch of the model with a vertical slice showing the key dimensions and the important features like the active fuel, hardware, poison plates, MPC and cask structure, etc.

<u>Response</u>

Sample input files are being provided under separate cover. A vertical slice of the principle geometry used in the criticality calculations is provided in FSAR Figure 6.3.7 for the HI-TRAC transfer cask and HI-STORM overpack, and in the HI-STAR FSAR (HI-2012610), Chapter 6, Figure 6.3.7 for the HI-STAR (the HI-STAR is used in most criticality analyses since the effect of the overpack on reactivity, in either case, is negligible). A horizontal slice of the MPC-32 (and MPC-32F) basket and HI-STORM overpack model is shown in Revision 1 of the HI-STORM FSAR, Figure 6.3.5. An individual cell of the MPC-32 (with intact fuel) is shown in FSAR Figure 6.3.2. The damaged fuel/fuel debris model in the MPC-32 is essentially the same as the model shown in FSAR Figure 6.4.9 for the MPC-68, with the DFC dimensions from new proposed FSAR Figure 2.1.2D used. The various bare fuel rod arrays are described in Report HI-2012771, Appendix C. An additional figure of an MPC-32 cell model with a DFC has been added as FSAR Figure 6.4.17.

3. Provide information to quantify and apply the reactivity effect for off-center fuel in the basket cells, per page 6-3 of NUREG-1536, *Dry Cask Storage Systems.* Consider off-center configurations beyond the gratis example given in NUREG-1536 (see NUREG-1567, *Spent Fuel Dry Storage Facilities*, Section 8.4.3.1, for other examples).

<u>Response</u>

Calculations were performed to evaluate the off-center fuel in the MPC-24E and MPC-32 fuel baskets. New FSAR Subsection 6.4.12 has been added to discuss this condition. For the MPC-32, there is a slight increase in reactivity when all assemblies are assumed to move toward the center of the cask. Reactivity remains below the limit of $k_{eff} \leq 0.95$. Calculations are documented in HI-2012771, Appendix I.

4. Provide information for the cases in Figure 6.4.14 to enable independent calculations. Include the variations in pitch or the fuel arrays, location of missing rods for damaged fuel, configuration of collapsed fuel, and configuration of fuel debris. Show that this figure applies to the 5% fuel case with borated water.

<u>Response</u>

The information for the cases in Figure 6.4.14 is documented in calculation package HI-951321, Rev. 13, which was submitted to the NRC with previous license amendment request 1014-1 (see Holtec letter to the NRC dated July 3, 2001). Revision 16 of this calculation package is being provided under separate cover. Also, see the reply to question 1 for further discussions on the configurations analyzed. FSAR Figure 6.4.14, which was added to the chapter in FSAR Revision 1, shows results for the MPC-24E with unborated water. The principal trend shown in this figure, specifically the distinct reactivity peak, are also found in the calculations for the MPC-24E with borated water and the MPC-32.

Chapter Seven, Confinement Boundary

1. Provide information to justify the gravitational settling values used in the confinement analysis. The confinement analysis uses gravitational settling values to reduce the amount of fines, volatiles and crud available for release from the canister. Use of gravitational settling values in the confinement analysis is a deviation from NUREG-1536, and Interim Staff Guidance (ISG) 5, *Confinement Evaluation*. Deviations from the NUREG-1536, and ISG-5 must be described and justified.

Revise the SAR to provide the assumptions and calculations used to develop the gravitational settling values for the amended HI-STORM 100 design. The assumptions and calculations should be in sufficient detail such that the staff can independently confirm if the gravitational settling values used in the analysis are appropriate for the HI-STORM 100 design and its contents. In addition, the assumptions and calculations should be of sufficient detail such that the staff can re-create these gravitational settling values.

<u>Response</u>

Proposed changes to Chapter 7 of the FSAR have been modified to describe the methodology used to calculate the gravitational settling values credited in the confinement analysis. Sufficient detail has been provided such that the staff will be able to independently confirm the appropriateness of the gravitational settling values. Table 7.2.2 (normal, off-normal conditions) and Table 7.3.9 (accident conditions) have been added to summarize the results of the calculation of gravitational settling values.

2. Provide information to explain how the gravitational settling values were incorporated into the confinement analyses.

The confinement analysis incorporates gravitational settling values which are not described in NUREG-1536 and ISG-5. Any deviations from NUREG-1536 and ISG-5, must be described and justified. Revise the SAR to describe how the gravitational settling values were incorporated with the methodology of ISG-5. The confinement evaluation should be in sufficient detail such that the staff can independently confirm that the deviation from ISG-5 is appropriate for the HI-STORM 100 design and its contents. In addition, the assumptions and calculations should be of sufficient detail such that the staff can re-create the HI-STORM 100 confinement analysis.

<u>Response</u>

Proposed changes to Chapter 7 of the FSAR have been modified to describe how the gravitational settling values were incorporated into the confinement analysis. Sufficient detail has been provided such that the staff will be able to independently confirm the appropriateness of the application of gravitational settling to the HI-STORM 100 design and contents. The confinement calculation is being provided under separate cover.

Chapter Nine, Acceptance Tests and Maintenance Program

Provide the qualification and acceptance tests for the neutron absorber (METAMIC™), including the statistical acceptance criteria. Provide engineering, testing evidence that the material functions according to the design requirements; i.e., Boron content, temperature resistance over time, etc. Submit manufacturing procedures to verify the quality of the material; i.e., how to avoid voids, assuring the Boron content, etc.

<u>Response</u>

EPRI Report 1003137 "Qualification of METAMIC® for Spent Fuel Storage Application" provides the information requested pertaining to qualification and acceptance testing of METAMIC® as a neutron absorber in dry fuel storage applications. Holtec-proprietary Report HI-2022871, "Use of METAMIC® in Fuel Pool Applications", includes a detailed discussion of the use of METAMIC® in wet storage applications, but also includes information germane to dry storage. Both of these reports conclude that METAMIC® is well-suited for use in spent fuel storage casks. Furthermore, METAMIC® has already been approved for use in dry storage under docket 72-1004 for the NUHOMS 61BT dry storage canister. We have confirmed that the NRC possesses a copy of the EPRI report, and the Holtec report is being provided under separate cover.

The quality of the manufactured material will be verified in the same manner as the quality of BORAL® neutron absorber material has been verified in the past. The tests and inspections required of the manufacturer are discussed in FSAR Section 9.1.5.3. These tests and inspections are implemented via the manufacturer's procedures, which are subject to audit and inspection by Holtec, its clients, and the NRC. The test results are maintained in the quality documentation package for the MPC.

Chapter Ten, Radiation Protection

Provide occupational dose assessments that are based on the bounding burnup and cooling times for the new proposed contents. Note, page 10-3 of NUREG-1536, states that the following, "The applicant should use these data [SAR sections 5 and 8] to estimate the dose received by occupational personnel during cask loading and transportation to the ISFSI."

<u>Response</u>

Chapter 10 has been revised as requested to reflect dose rates that bound the burnups and cooling times for the revised cask contents.

TECHNCIAL ISSUES FOR FURTHER EVALUATION

Technical Specifications

1. Provide an appropriate provision in Technical Specification 3.1, LCO 3.3.1.e, if the applicant's intent is to request damaged fuel or fuel debris with enrichments less than or equal to 4.0 wt%, as contents for the MPC-24E or MPC-24EF.

   Response

   Loading of damaged fuel and fuel debris with enrichments less than or equal to 4.0 wt% in the MPC-24E/EF was previously approved by the NRC in HI-STORM CoC Amendment 1 and is not proposed to be modified in this amendment request. No soluble boron is required for damaged fuel and fuel debris in the MPC-24E or MPC-24EF with enrichments less than or equal to 4.0 wt% (see Table 2.1-2 in Appendix B to the CoC, Row 4 and Note 7). Therefore, a provision in Technical Specification 3.1, LCO 3.3.1.e, is not necessary for this condition.

2. For the Technical Specification, Appendix B, Table 2.1-1, Item VIII.C., "Neutron sources," either add this item to the section in the SAR entitled, "CoC Markup," or delete it from the section entitled, "Revised CoC." If the applicant's intent is to add Item VIII.C., "Neutron sources," then clearly describe what "neutron sources" are being requested for contents in the MPC-32F. Provide isotopic information, such as activity, decay time, etc. Perform a detailed shielding analysis in Chapter Five of the SAR.

   Response

   This is an editorial error. No additional neutron sources are being requested for authorization for loading in the HI-STORM 100 System. Therefore, "Neutron Sources" has been deleted from LAR Attachment 4 entitled "Revised CoC."

Chapter Five, Shielding

1 Provide information in Section 5.2, which specifies the expected error in source term estimates for actinides and fission products important in shielding (e.g., Cs-134 and Cm-244), and source term estimates for total decay heat, as a function of requested burnup between 45 and 75 GWd/MTU.

   Response

   During the RAI process for the recently-approved HI-STORM Amendment 1, a similar RAI (number 5-2) was received and answered by Holtec That RAI response provides detailed information on this subject (see Holtec's RAI response to the NRC dated July 3, 2001). Under Amendment 1 the allowable maximum burnups for PWR and BWR that were approved by the NRC are 68,200 MWD/MTU for PWR and 59,900 MWD/MTU for BWR fuel. In LAR 1014-2, Holtec is requesting an extension of these burnups to 75,000 MWD/MTU and 70,000 MWD/MTU for PWR and BWR fuel, respectively. Since the extension in burnups being requested is relatively small, it is reasonable to assume that the analytical methods, SAS2H and ORIGEN-S, which were approved for high burnup fuel in LAR 1014-1, are still valid without additional studies.

2. Clarify how the burnup and cooling time combinations in Table 5.2.1 were determined to bound the burnup and cooling time combinations in Appendix B of the proposed CoC.

<u>Response</u>

The burnup and cooling times in FSAR Table 5.2.1 were meant to only refer to the HI-STORM shielding analysis. The limits in the CoC are also a function of the storage configuration (i.e., uniform or regionalized fuel storage). This approach may have created some confusion in this amendment request because of the different burnup and cooling times being analyzed for the HI-TRAC and HI-STORM. Therefore, these two lines in the table have been removed. The discussion in Section 5.1 provides the burnup and cooling time combinations that were analyzed for the various MPCs in the HI-STORM overpack and the HI-TRAC transfer casks. Section 5.1 also provides the explanation as to how these burnup and cooling times were chosen. The burnups for the HI-STORM analysis reported in Section 5.1 were chosen at the 3 year cooling time to be slightly higher than the values used for the HI-TRAC analysis. In both cases (HI-STORM and HI-TRAC) the burnups used for the analysis bound the values in the CoC.

3. Justify in Section 5.4, why a burnup and cooling time combination of 43.5 GWd/MTU and 3 years is specified to produce the highest dose rates in the MPC-24 in the 100-ton HI-TRAC, whereas Table 5.2.1 specifies the bounding burnup as 47.5 GWd/MTU.

<u>Response</u>

As a result of the changes in the thermal analysis, the allowable burnups have changed. Therefore, the analysis in Section 5.4 has been revised with results presented at different burnup and cooling time combinations. As stated in Section 5.1, the burnup and cooling time combinations for the HI-TRAC analysis were chosen to bound the allowable burnup and cooling time combinations in the CoC. Please also see the response to Question 2.

4. Justify in Section 5.4, why a burnup and cooling time combination of 70 GWd/MTU and 10 years was used to calculate dose rates from the MPC-24 in the 125 ton HI-TRAC, whereas Appendix B of the proposed CoC indicates fuel with a burnup and cooling time of 75 GWD/MTU and 6 years may be authorized contents.

<u>Response</u>

The analysis for the 125-ton HI-TRAC has been changed to reflect burnup and cooling time combinations for the minimum cooling time and maximum burnups. The new burnup and cooling time combinations for the 125-ton HI-TRAC bound the values in the CoC.

5. Justify in Section 5.4, why a burnup and cooling time combination of 45.5 GWd/MTU and 4 years cooling was used to calculate dose rates from the MPC-32 in the 100-ton HI-TRAC, whereas Table 5.2.1 specifies the bounding burnup and cooling time as 33 GWd/MTU and 3 years.

<u>Response</u>

As a result of the revised thermal analysis, the burnup and cooling time combinations used in the shielding analysis have changed and a burnup and cooling time combination at 4 years is no longer analyzed. However, the previously reported values were correct and bounding. The previous analysis was performed by choosing multiple burnup and cooling time combinations for cooling times ranging from 3 to 20 years. It happened that the burnup and cooling time combination at 4 years produced slightly higher dose rate than the burnup and cooling time combination at 3 years and therefore the results for 4 years were reported in the FSAR. Please also see the response to question 2.

Chapter Six, Criticality

1. Indicate which version of the HI-TRAC was modeled in the evaluations cited on page 6.1-4 (the 100 ton model with 2.875" thick lead or the 125 ton model with 4.5" thick lead). Provide sketches of this model.

<u>Response</u>

The 125-ton HI-TRAC design was modeled in the evaluations. Sketches of this model are shown in FSAR Figure 6.3.7 (axial configuration) and Figures 6.3.4 through 6.3 6 (cross sections for the different baskets). Note that the differences in transfer cask designs has a statistically insignificant effect on reactivity calculations.

2. Provide information to document the statement on page 6.2-1, which indicates that k_{eff} is maximum for maximum active fuel length.

   Response

   The information is provided in Report HI-951321, Attachment F, Page F-1-46. Additional information has also been provided in FSAR Chapter 6 on pages 6.2-2 and 6.2-20.

3. Clarify the fuel debris model for case 3 on page 6.4-8 (e.g., describe the number of pellets which change to powder. Indicate if clumps or chucks were considered).

   Response

   A mixture of fuel and water was used in this analysis, i.e., other materials such as fuel cladding were neglected. The fuel content was approximately 19 volume percent. Clumps or chunks were not considered.

4. Provide statistical uncertainties to evaluate the significance of differences in summary of results, for comparison of the detailed results in Appendix 6C.

   Response

   Statistical uncertainties for most calculations not listed in Appendix 6C are listed in the corresponding calculation packages. To assist the reviewer in locating information in the calculation packages, cross-reference tables are provided in Report HI-2012771, Appendix J

5. Clarify how the data were obtained for Figure 6.4.10. Explain how it differs from the data in Table 6.4.1.

   Response

   As stated in FSAR Section 6.4.2.1.1, data in FSAR Figure 6.4.10 is for 100 percent external water reflection, whereas data in Table 6.4.1 are for various different values of external water reflection.

6. Explain how the mass of fuel per unit length was varied as cited on page 6.4-12. Indicate if the density or diameter was changed, or if fuel rods were added to the lattice.

Response

The mass of fuel per unit length was varied by changing the number of fuel rods in the DFC and the fuel rod diameter. The density remained unchanged (see corresponding assumption in FSAR Section 6.1). For details see report HI-951321, Attachment F, Pages F-1-105 through F-1-112.

7. Explain if the hypothetical fuel debris configuration referenced in the last paragraph in Section 6.4.4.2.6, is the same as that described in item no. 4 of Enclosure 1, or how it is different.

   Response

   The "hypothetical fuel debris configurations, i.e. various arrays of bare fuel rods" discussed in the last paragraph in FSAR Section 6.4.4.2.6 for the MPC-32/32F is principally the same as that used in the analyses for the MPC-24E/EF shown in FSAR Figure 6.4.14. The sole difference is a difference in the pitch for a given array size of bare fuel rods made necessary by the difference in the inner dimension between the MPC-24E/EF DFC and the MPC-32/32F DFC.

8. Describe the studies referenced in the third sentence in the last paragraph on page 6.4-9.

   Response

   These studies are described in Report HI-951321, Attachment J.

LAR 1014-2, REVISION 1 SUMMARY OF PROPOSED HI-STORM 100 SYSTEM CHANGES

SECTION I - PROPOSED CHANGES TO CERTIFICATE OF COMPLIANCE 1014

Proposed Change No. 1

Certificate of Compliance, Section 1.b and Appendix B, Section 3.2:

Remove the specific reference to BORAL® neutron poison material to allow the use of an alternate, equivalent neutron poison material, METAMIC®, as defined in the FSAR

Reason for Proposed Changes

This change is proposed to allow flexibility in choosing the neutron absorber material used in the MPC basket. The neutron absorber material METAMIC® is proposed as an alternative to BORAL®. Because of the absence of interconnected porosities, the time required to dry a METAMIC®-equipped MPC is expected to be less compared to an MPC containing the rolled cermet class of neutron absorbers such as BORAL®.

Justification for Proposed Changes

METAMIC® neutron poison material has been demonstrated to be equivalent to Boral in performing the design function of absorbing thermal neutrons. METAMIC® is also equivalent to BORAL® in its thermal, structural, and shielding performance. The dimensions and tolerances for the fabrication and installation of the METAMIC® neutron absorber panels are identical to the current BORAL® dimensions and tolerances.

EPRI Report 1003137, "Qualification of METAMIC® for Spent-Fuel Storage Application" provides the pertinent qualification tests data for this material. Holtec International proprietary Report HI-2022871, "Use of METAMIC® in Fuel Pool Applications", includes a detailed discussion of the use of METAMIC® in wet storage applications, but also includes information germane to dry storage. Both of these reports support the conclusion that METAMIC® is well-suited for use in spent fuel storage casks. The EPRI report is already in NRC's possession and the Holtec report is being submitted under separate cover. See proposed revisions to FSAR Sections 1.2.1.3, 4.2, 5.3, 6.4.11, and 9.1 in Attachment 5 for additional discussion.

Note: Appropriate conforming editorial changes to the MPC design drawings will be made after approval of the CoC amendment.

Proposed Change No. 2

Certificate of Compliance, Section 1.b and 9; Appendix A, LCO 3 3.1; and Appendix B, Table 2.1-1:

a. Modify CoC Section 1.b, Appendix A, LCO 3.3.1, and Appendix B, Section V of Table 2.1-1; and add new Section VIII to Appendix B, Table 2.1-1 to authorize damaged fuel for loading into the MPC-32 and damaged fuel and fuel debris for loading into the MPC-32F.

b. Revise LCO 3.3.1 to re-format the required minimum soluble boron requirements for MPC-32/32F to provide the appropriate values for soluble boron based on fuel assembly array/class, intact vs. damaged fuel, and initial enrichment.

Reason for Proposed Changes

a. Damaged fuel and fuel debris currently are not authorized for loading in the MPC-32. Users currently must load PWR damaged fuel and fuel debris in the MPC-24E and -24EF. This change would enable customers to load all MPC-32 canisters on their ISFSI if they choose to do so.

b. The reformatting of the MPC-32/32F soluble boron requirements reduces the current, across-the-board soluble boron concentration of 2,600 ppmb for MPC-32 to account for differences in fuel types and enrichments. This change can help reduce the amount of radioactive waste produced at a plant if the boron concentration in the spent fuel pool must be temporarily increased for cask loading.

Justification for Proposed Change

a. The addition of damaged fuel and fuel debris as authorized contents in the MPC-32 and MPC-32F has been analyzed and found to be acceptable. The creation of MPC-32F entails only the thickening of the MPC shell at the top (with an associated reduction in the diameter of the MPC lid) and increasing the size of the lid-to-shell weld. This design difference is exclusively needed for qualification of the dual-purpose MPC for 10 CFR 71 transport loads - see proposed changes to FSAR Section 2.1.3 and new FSAR Figure 2.1.9: The rest of the MPC-32 and MPC-32F shell and basket designs are identical. This is the same design detail previously approved for the MPC-68F, MPC-68FF, and MPC-24EF in earlier CoC amendments. Allowing users to load damaged fuel and fuel debris into 32-assembly MPCs instead of 24-assembly MPCs reduces the risk of operating events and reduces the overall dose to personnel from ISFSI operations by reducing the total number of casks required to store a given amount of spent fuel. The MPC-32/32F damaged fuel container is shown in new FSAR Figure 2.1.2D. The technical evaluation is summarized below by discipline.

Structural

The generic MPC-32/32F damaged fuel container (DFC) design is different in two respects from the previously approved generic MPC-24E PWR DFC: 1) the wall thickness is reduced from 0.075 inch to 0.0239 inch and 2) there is one additional spot weld per side in the MPC-32/32F DFC baseplate. A structural evaluation for the MPC-32/32F DFC is documented in the MPC structural calculation package (Report HI-2012787, Supplement 25) and all safety factors remain greater than 1.0.

Thermal

The storage of DFCs containing damaged fuel assemblies in the peripheral fuel cells in MPC-32/32F is acceptable because, in a bounding evaluation, the effect of the presence of DFCs on peak fuel cladding temperature is negligible (i.e., much less than 1°F).

Shielding

Generic damaged fuel has been analyzed in the MPC-24 and the MPC-68. The effect of storing damaged fuel and the post-accident consequences of collapsed damaged assemblies has been analyzed for the MPC-24 and the MPC-68. The results presented in the FSAR for the MPC-24 and the MPC-68 conclude that there is little effect on the external dose rates as a result of storing damaged fuel assemblies in these baskets.

Since storage of damaged fuel in the MPC-32 is similar to the MPC-24 and MPC-68 in that a limited number of assemblies are stored on the periphery of the basket, the effect on the external dose rates from storing damaged fuel in the MPC-32 will be similar to the effect seen in the MPC-24 and the MPC-68. Based on the results for the MPC-24 and MPC-68, it is concluded that the effect on the external dose rates from storing damaged fuel in the MPC-32 will be small. Therefore, storage of damaged fuel in the MPC-32 is acceptable from a shielding perspective without performing explicit MCNP calculations. Section 5.4.2 of the FSAR has been modified to add the above discussion pertaining to MPC-32.

Criticality

Criticality evaluations were performed for the MPC-32/32F with intact fuel and damaged fuel/fuel debris using the same bounding fuel debris model developed in HI-STORM Amendment 1 for the MPC-68/68FF and the MPC-24E/EF. Additional calculations were performed to demonstrate that this model is conservative in the presence of soluble boron. Details of the damaged fuel model and calculations are discussed in general in FSAR Section 6.4.4.2, and calculations for the MPC-32/32F are specifically addressed in Section 6.4.4.2.6. These proposed FSAR changes may be found in Attachment 5.

Note that some of the reactivities reported in Tables 6.1.5 and 6.1.6 have increased slightly, although the corresponding soluble boron requirement were not changed. This is the result of a more extensive and slightly more conservative set of evaluations regarding the water density and the fill status of the guide tubes (see Tables 6.4.10, 6.4.11 and 6.4.14). These evaluations were necessary for consistency between the assembly classes and soluble boron levels.

Confinement

There is no impact on the confinement analysis since damaged fuel and fuel debris are not treated differently than intact fuel. The confinement analysis described in FSAR Revision 1 is applicable to the MPC-32 with damaged fuel and the MPC-32F with damaged fuel and fuel debris.

b. The re-formatting of the minimum boron concentration is consistent with the supporting criticality evaluations. FSAR Section 6.4, and specifically Section 6.4.2.1.2, (Attachment 5) contain the details of the supporting evaluations. Users who previously may have had to increase the boron concentration in the spent fuel pool to load an MPC-32, may not need to do so if their normal spent fuel pool soluble boron concentration is sufficiently high. The eliminates the radioactive waste produced when boron concentration is temporarily increased for cask loading and subsequently decreased for normal pool operation.

Proposed Change No. 3

Certificate of Compliance, Sections 1.a and 1.b:

Revise the wording in these two CoC sections as follows:

a. In Section 1.a and the first paragraph of Section 1.b, delete the "100 or 100S" designation in the references to the HI-STORM overpack

b. In the second paragraph of Section 1.b, clarify that the aluminum heat conduction elements (AHCEs) are optional hardware for MPCs loaded under Revision 0 or Amendment 1 to the CoC and are prohibited for MPCs loaded under Amendment 2 or later amendments.

Note: Conforming editorial changes to the affected MPC drawings will be made after approval of this amendment request.

c. In the third paragraph of Section 1.b, delete all information pertaining to the authorized contents of each MPC model and add a statement defining the suffix to the MPC model number.

d. In the fourth paragraph of Section 1.b, change the word "types" to "sizes" in two places in reference to the HI-TRAC transfer cask.

e. In the fifth paragraph of Section 1.b: i) clarify the description of the differences between the HI-STORM 100S and HI-STORM 100 overpacks, ii) specify number of overpack air inlets and outlets as minimums, and iii) delete the terms "standard" and "short" from the discussion of HI-STORM 100A.

Reason and Justification for Proposed Changes

a. These changes for the overpack description are proposed for consistency with the discussion of the HI-TRAC transfer cask and MPCs in these portions of the CoC.

b. For those MPCs loaded under CoC Amendment 2 or later, the AHCEs are prohibited because they have not been included in the thermal evaluation model. In the thermal evaluation for those MPCs loaded under the original CoC or Amendment 1, the aluminum heat conduction elements <u>were</u> conservatively modeled as a flow restriction, but no credit was taken for heat transfer through them in the bounding thermal analysis presented in FSAR Revision 1; therefore, the AHCEs are optional equipment for MPCs loaded under the original CoC or Amendment 1. There are a number of MPCs that are, or will be loaded under the original CoC or Amendment 1 that contain AHCEs. Therefore, this proposed change is consistent with past and future MPCs and the supporting thermal analyses. Sections 1.2.1.1 and 4.4.1.1.b of the proposed FSAR (Attachment 5) have been modified appropriately to address this change.

c. This information currently duplicates Section 6 of the CoC, which refers to Appendix B of the CoC for approved contents. Appendix B of the CoC contains detailed specifications for the contents of each MPC model, including all of the information contained in the material proposed for deletion. This changes eliminates redundancy in the CoC.

d. This wording change provides clarification in distinguishing between the 125-ton and the 100-ton HI-TRAC transfer casks. The term "types" is too general and subject to misinterpretation. The term "sizes" is more correct for distinguishing between the 100-ton and 125-ton transfer casks.

e. These wording changes provide i) clarification of the major differences between the 100S and 100 overpack designs, ii) flexibility regarding the number of air inlets and outlets for potential future modifications, and iii) clarification by removing redundant terms for the HI-STORM 100 and HI-STORM 100S overpack designs.

Proposed Change No. 3a

Certificate of Compliance, Section 11:

Revise the wording in this CoC section as shown in the attached CoC markup to reflect Amendment 2 as the latest amendment.

Reason and Justification for Proposed Change

Administrative change.

Proposed Change No. 4

Certificate of Compliance, Appendix A, SR 3.1.1.1, SR 3.1.1.3 and Table 3-1:

a. Revise Surveillance Requirement (SR) 3.1.1.1 and Table 3-1, and relocate information previously in Table 3-1 to new Table 3-2 as shown in the attached markup CoC to reflect necessary changes in requirements for MPC cavity drying.

b. Revise SR 3.1.1.3 to move the leakage rate acceptance criterion from previous Appendix A, Table 3-1 to the SR itself.

c. Revise the helium backfill requirements in new Table 3-2 (previously located in Table 3-1) as shown in the attached mark-up of the CoC.

Reason for Proposed Changes

a. These proposed changes in MPC cavity drying requirements are necessary as a consequence of higher authorized heat loads and the new peak fuel cladding temperature limit suggested by ISG-11, Revision 2[1]. There are now a variety of requirements and options based on the decay heat load of the MPC and the burnup of fuel being stored (i.e., moderate burnup versus high burnup fuel).

b. This is a human factors improvement. There is no change to the acceptance criterion.

c. This proposed change is a result of lessons learned in the field on implementing the current helium backfill pressure requirement of 29.3 to 33.3 psig. Due to the accuracy of instruments available for performing this activity in then field, more precision was required in establishing the appropriate range in the Technical Specifications.

Justification for Proposed Changes

a. The proposed changes in MPC cavity drying requirements create the necessary controls to ensure the peak fuel cladding temperature limit of 400°C is not exceeded during short term loading operations. They also provide optional requirements (fuel cladding hoop stress calculations) for MPC containing all moderate burnup fuel (≤ 45,000 MWD/MTU) to all the fuel cladding temperature to approach the previous licensing basis limit of 570°C during vacuum drying, for heat loads up to those already licensed in Amendment 1 to the CoC. Any MPC containing one or more high burnup fuel assemblies must be dried using the forced helium dehydration method, in which case the 400°C temperature limit is ensured for all authorized heat loads. See proposed changes to FSAR Section 4.5 in Attachment 5 and Holtec calculation HI-2002407 for details of the thermal analyses.

b. Human factors improvement.

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[1] The modified MPC cavity drying requirements also reflect the fuel cladding hoop stress calculation option to retain the 570°C temperature limit for moderate burnup fuel (≤ 45,000 MWD/MTU) This is expected to be consistent with the soon-to-be-published ISG-11, Revision 3.

c. The TS requirements for helium backfill more accurately account for the potential range of instrument accuracies in the field, the different MPC cavity drying methods, and the supporting thermal analyses. The thermal analyses evaluate a lower bound helium backfill value that ensures a sufficient density of helium is in the MPC to promote adequate thermosiphon heat transfer. They also evaluate an upper bound value in determining the pressure consequences of certain accident events, such as extreme temperature and 100 percent fuel rod failure. See proposed changes to FSAR Section 4.4.1 in Attachment 5 for additional justification.

Proposed Change No. 5

Certificate of Compliance, Appendix A, LCO 3.1.3 and associated Bases in FSAR Appendix 12.A:

Revise this LCO and associated Technical Specification Bases as shown in the attached markup of the CoC and FSAR Appendix 12.A to:

a. Provide appropriate requirements for ensuring MPC cavity bulk helium temperature is less than 200 degrees F prior to re-flooding, instead of the existing "helium gas exit temperature." Revise associated bases in the FSAR accordingly.

b. Change the Completion time of Required Action A.2 from 22 hours to "Immediately."

Reason for Proposed Changes

a. Using a forced helium recirculation system to cool the MPC cavity gas for low decay heat load casks may be unnecessary in the unlikely event that an MPC must be unloaded. This change provides appropriate flexibility for users who may have to unload an MPC with low a decay heat load.

b. This change is required as a result of the new, lower peak fuel cladding temperature limit of 400°C during short-term operating conditions, including unloading operations.

Justification for Proposed Change

a. Depending upon the decay heat in the cask at the time of unloading, it may not be necessary to cool the contained helium with a recirculating helium cooldown system prior to re-flooding with water. The helium temperature of very low decay heat load casks could be less than 200 degrees F at the time of re-flooding with no action required. Alternatively, adequate cooling of the helium inside the MPC may be able to be accomplished by non-intrusive means, such as air or water applied to the outside surface of the MPC.

The bulk temperature of the helium in the MPC and the fuel cladding itself, is predicted using a computational fluid dynamics computer program (FLUENT, FSAR Chapter 4) to license the cask for normal, off-normal, and accident conditions of storage. These same analytical techniques, accepted as the basis for loading operations and long-term fuel storage, can be used to predict the bulk helium temperature of an MPC designated for unloading. The actual characteristics of the MPC contents (i.e., fuel type, presence of non-fuel hardware, time in storage) can be used to conservatively predict the bulk helium temperature prior to re-flooding. The results of that prediction would then be used to determine the appropriate means (if any are necessary) and time frame to cool the bulk helium down to 200°F prior to re-flooding in order to minimize thermal stress in the fuel cladding.

b. The thermal analyses described in FSAR Section 4.5 indicate that there are restrictions on the time an MPC may reside in a HI-TRAC transfer cask, based on the heat load of the MPC. Furthermore, unloading operations taking place in a decontamination pit or fuel vault require additional restrictions. The specifics of what actions, if any, are required are detailed in the proposed modifications to the TS Bases in FSAR Appendix 12.A (Attachment 5).

For loaded transfer casks located in an open area, such as a fuel floor, the time limits in CoC Appendix B, Section 3.4.10 apply. For transfer casks in a pit or vault, the thermal analysis shows that, for MPC heat loads up to 25 kW, the steady state fuel cladding temperature does not exceed 400°C, so adequate cooling is provided and no further action is required. For heat loads above 25 kW with the transfer cask in a pit or vault, users must ensure the applicable fuel cladding temperature limit is not exceeded either by placing a limit on the length of time the cask can reside in the pit or vault, or by provided augmented cooling. The time limit or type of augmented cooling is necessarily site-specific and is left to the user to determine, using the thermal methodologies in the HI-STORM FSAR.

Proposed Change No. 6

Certificate of Compliance, Appendix A, LCOs 3.2.1, 3.2.2, and 3.2.3; Action B.1 of LCO 3.1.2; and Section 5.0:

Delete LCOs 3.2.1, 3.2.2, and 3.2.3 and associated bases in FSAR Appendix 12.A and replace them with new Technical Specification Program 5.7 for radiation protection, located in CoC Appendix A, Section 5.0. Modify the Required Action in LCO 3.1.2 to conform with this change.

Reason for Proposed Change

The current Required Actions for LCOs 3.2.1 and 3.2.3 do not lead to an end point that results in compliance with the LCO requirements. For example, if dose rates on the HI-TRAC transfer cask exceed one of the LCO 3.2.1 limits, Required Actions A.1 and A.2 of that LCO require the cask user to administratively verify correct fuel loading and to perform an evaluation to verify compliance with 10 CFR 20 and 10 CFR 72, respectively. Once these actions are complete, operations are permitted to continue, yet the cask surface dose rates would remain out of compliance with the LCO limits. The same logic applies to LCO 3.2.3 for HI-STORM overpack dose rates.

In addition, this change is proposed to be consistent with the guidance of NUREG-1745, "Standard Format and Content for Technical Specifications for 10 CFR Part 72 Cask Certificates of Compliance" and with many plants' Part 50 technical specifications. General licensees' radiation protection programs that implement the ALARA philosophy are considered sufficient to protect operations personnel and the public and to ensure compliance with regulatory dose limits.

The program description also includes specific requirements on the methodology and key inputs used in any shielding analyses and evaluations performed under the program. These additional requirements provide continued NRC control over certain aspects of any shielding analyses and evaluations performed to demonstrate compliance with off-site dose limits and in support of changes made under the provisions of 10 CFR 72.48.

Justification for Proposed Change

There are no numerical regulatory limits on contact dose rates from a spent fuel storage cask in 10 CFR 72 or 10 CFR 20. For normal and off-normal operations, however, general licensees must demonstrate compliance with 10 CFR 72.104 for dose at the controlled area boundary. Compliance with 10 CFR 72.104 is site-specific, based on the dose rate from reactor operations, contents of the casks, the number of casks at the ISFSI, local meteorology, and the distance to the site boundary. The contact dose rate limits in current LCOs 3.2.1 and 3.2.3 serve no function for licensees in demonstrating compliance with 10 CFR 72.104.

Compliance with surface dose rate limits in the Technical Specifications (TS), or elsewhere, are not a reliable indicator of proper cask loading or consistency with the site-specific off-site dose analysis. Specifically, if a measured cask surface dose rate exceeds the cask TS value, certainly a mis-loading has occurred. However, measuring a surface dose rate less than the limit in no way assures that all contents loaded meet the CoC requirements. This is because the actual contents of a cask loaded at a given general licensee's facility will never match the bounding design basis contents used in the licensing basis shielding analyses. Individual fuel assemblies or non-fuel hardware not meeting the CoC could be loaded with the overall effect on dose rate being insignificant. The administrative controls used to select and document fuel assemblies and non-fuel hardware chosen for loading in a cask (equivalent to those used to store fuel in a plant's spent fuel pool) are the only reliable way to ensure the fuel loading requirements of the CoC are met.

Contact dose rates from the casks are a factor in determining occupational exposures during cask loading operations. Occupational exposure regulatory limits are set by 10 CFR 20 and exposures to personnel are generally controlled to even lower limits through the users' ALARA-based radiation protection programs. The dose rates to personnel from a loaded HI-STORM overpack or HI-TRAC transfer cask are necessarily site-specific, and cask specific, based on the particular contents of the cask. Part 50 licensees are well-versed at handling radioactive containers, many of which emit much higher levels of radiation than a dry storage cask. Similarly, contamination control and handling of contaminated containers lies within the core expertise of Part 50 licensees. Therefore, these requirements are more appropriately controlled through a Technical Specification program.

Proposed Change No. 7

Certificate of Compliance, Appendix B, Section 1.0, Definitions; Table 2.1-1, Note 1 in Sections I, IV, V, VII, and VIII; and Note 3 of Table 2.1-8

Revise the definition of NON-FUEL HARDWARE as shown in the attached mark-up of the CoC to include vibration suppressor inserts. Revise the subject notes as shown to allow the storage of vibration suppressor inserts as integral non-fuel hardware that may be stored in the MPC with a fuel assembly.

Reason for Proposed Change

Vibration suppressor inserts have been identified by a number of Holtec's clients as non-fuel hardware that is integral to the fuel assemblies and must be qualified for storage. Vibration suppressor inserts were added by certain fuel vendors as a design feature to address a vibration-induced failure problem in operating reactors.

Justification for Proposed Change

The vibration suppressor inserts contain no fissile material and have been evaluated as activated hardware (BPRAs). See Section 5.2.4 of the proposed FSAR changes (Attachment 5) for additional information. Table 2.1-8 of CoC Appendix B has been modified to include the vibration suppressor inserts with the existing approved fuel insert burnups and cooling times.

Proposed Change No. 8

Certificate of Compliance, Appendix A, LCO 3.3.1; Appendix B, Table 2.1-1, Section IV; and Appendix B, Table 2.1-2:

Increase the maximum authorized initial enrichment for PWR damaged fuel and fuel debris to 5.0 wt.% as shown in the attached mark-ups of the CoC

Reason for Proposed Change

PWR users have damaged fuel and fuel debris up to 5 wt.% initial enrichment that needs to be placed into dry storage.

Justification for Proposed Change

Damaged fuel and fuel debris up to 5.0 wt.% ^{235}U has been evaluated and found to be acceptable for loading in the PWR MPCs. See Section 6.4.4.2.5 in the attached proposed FSAR changes for detailed justification.

Proposed Change No. 9

Deleted

Proposed Change No. 10

Certificate of Compliance, Appendix B, new proposed Section 2.3:

Provide a process for the certificate holder to request and receive NRC approval of case-specific alternatives to the cask contents on behalf of a cask user, as shown in the attached markup of the CoC.

Reason for Proposed Change

To provide necessary flexibility for the NRC to review and approve, upon request by Holtec, small deviations from the cask contents limits in the CoC that have been shown to have little or no safety significance. This change process will eliminate the need for licensees to request exemptions from the regulations or significantly delay their fuel loading schedules for small, non-safety significant changes to the CoC cask contents on a case-specific basis.

Justification for Proposed Change

This proposed change is consistent with NUREG-1745, "Standard Format and Content for Technical Specifications for 10 CFR Part 72 Cask Certificates of Compliance." NUREG-1745 also suggests moving some of the cask contents limits to the FSAR. However, Holtec has chosen to leave all cask contents parameters the same as currently found in the CoC (although some of the values for those parameters are proposed to be changed in this amendment request). This is conservative since a CoC amendment would still be necessary to permanently change any of the parameters (or the values) in the CoC, including any case-specific changes approved by the NRC under this process. The flexibility permitted by this proposed change is appropriate because there may be instances where cask users' fuel or other contents have slight deviations from the limits in the CoC, where there is little or no safety concern with granting the deviation on a case-specific basis.

This change process allows Holtec to support our customers' fuel loading schedules without the users having to request exemptions from the regulations for small deviation from the approved contents section of the CoC. Holtec will then pursue a permanent change to the parameter or value on a normal priority schedule, using the CoC amendment process. Overall, this change process eliminates unnecessary regulatory burden in an area of little or no safety significance, but retains the requirement for prior NRC approval of cask contents changes.

Proposed Change No. 11

Certificate of Compliance, Appendix A, LCO 3.1.2 - Required Actions B.2.1 and B.2.2, and SR 3.1.2.1; and Appendix B, Tables 2.1-1 and 2.1-4 through 2.1-7:

Revise Table 2.1-1, delete Tables 2.1-4 through 2.1-7, and create new Section 2.4 in Appendix B as shown in the attached markup of the CoC to provide new (higher) limits for fuel assembly decay heat and burnup as a function of cooling time and as a function of fuel array/class. Modify the Completion Times for Required Actions B 3.2.1 and B 3.2.2 to reflect the revised blocked duct accident analysis. Modify the acceptance criterion for temperature measurement in SR 3.1.2.1 to be 158°F to conform to these changes. See also Proposed Change 15a.

Reason for Proposed Changes

Based on user input, the existing limits unnecessarily penalize certain fuel types due to only grouping by reactor type (PWR or BWR). The previous limits did not meet the entire spectrum of users' needs to store fuel with higher heat emission rates. Other changes are conforming changes made necessary by the higher heat loads.

Justification for Proposed Change

Thermal

The previous burnup and decay heat limits were distinguished only by PWR or BWR fuel type for each MPC model. The revised limits are specified by fuel array/class and MPC model to provide an improved specificity for the various fuel types. The new limits appropriately reflect the ability of the HI-STORM 100 System to reject more heat than previously authorized, while still retaining adequate margins to the various limits (see revised FSAR Section 4.4 in Attachment 5). Placing the higher burnup fuel in the central core of the basket, surrounded by lower burnup fuel reduces the overall dose to personnel and the public from ISFSI operations due to the self-shielding phenomenon of the fuel assemblies. See proposed revisions to FSAR Section 4.4.1.1.9 (Attachment 5) for additional justification. The permissible fuel cladding temperature limit used to determine the maximum cask heat loads are consistent with ISG-11, Revision 2. (see also Proposed Change Number 15a).

Shielding

The shielding analysis in Chapter 5 of the FSAR has been modified to reflect the changes in the allowable burnup and cooling times by changing all dose rate calculations using the design basis fuel assemblies, B&W15x15 and GE7x7. The source terms have also been changed appropriately. The choice of design basis fuel assembly for the shielding analysis remains the same. Section 5.2 has been modified slightly to address the fact that the different array classes have different burnup and cooling times as a result of this change. The design basis assemblies remain valid because the analysis in Chapter 5 uses the maximum burnup from all array classes for a given cooling time. This is described in Section 5.1 of the proposed Revision 2 FSAR (Attachment 5).

In conjunction with calculating the allowable burnups for the different array classes, Tables 5.2.25 and 5.2.26 have been slightly modified. In Table 5.2.25, the pellet diameter and resulting uranium loadings of three of the assemblies have been increased to be consistent with the maximum permissible value in the CoC. In Table 5.2.26, the 9x9 assembly has been modified to reflect the 9x9 array class which now has the highest decay heat load for the specified burnup and cooling time in that table.

In the calculation of the allowable burnups for the different array/classes an additional change was made in the shielding analysis. Rather than use the same power level of 40 MW/MTU for all array/classes, the power per assembly was calculated for each reactor type and increased by 10 or 20% to account for potential power uprates for the PWR and BWR plants, respectively. Tables 5.2.25 and 5.2.26 reflect this change as does Section 5.2.5 in Attachment 5.

Accidents

Placing the relatively hotter fuel assemblies in the center of the MPC basket by design obviates the need to analyze a fuel assembly mis-loading accident. This is because, as described in FSAR Section 4.4.1.1.9, the inadvertent loading of a hotter fuel assembly on the periphery is actually more advantageous from a thermal perspective (i.e., the heat rejection of the cask system would be better with hotter assemblies on the periphery of the fuel basket). From a shielding perspective, the mis-loading of an assembly would result in a small, localized increase in the contact dose rate on the cask and would be imperceptible at the controlled area boundary.

The 100% air duct blockage accident was re-analyzed for two threshold heat loads. This re-analysis is discussed in FSAR Section 11.2.13 (Attachment 5). The results of the analyses show that, for heat loads ≤ 27.74 kW (the Amendment 1 maximum heat load), no components reach their short term temperature limit. For a bounding heat load of 41.22 kW, no components reach their short term temperature limit for 33 hours. The Completion Times for Required Actions B3.2.1 and B3.2.2 have been revised to reflect these results. Note also that the basis for the revised Completion Times no longer includes the assumption that the complete blockage of all inlet ducts occurs immediately after completion of the last surveillance. This change is consistent with the bases for Completion Times in power reactor technical specifications, which are developed assuming that the degraded condition begins at the time the component or system is declared inoperable[2]. It is not required to assume the component or system has been inoperable since the last successful completion of the Surveillance Requirement. See also the Bases for LCO 3.1.2 in FSAR Appendix 12.A (Attachment 5).

Proposed Change No. 12

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

Revise the maximum allowable uranium masses for certain fuel assemblies as shown in the attached markup to the CoC. These changes are made to maintain consistency with the revised shielding analyses in Chapter 5.

Reason for Proposed Changes

As Proposed Change 11 discusses, the allowable burnups are being calculated in this LAR for different array classes rather than a single PWR or BWR array/class. Explicit analysis has been performed to determine the allowable burnups for each array/class. The change to the allowable uranium mass loadings is being made to reflect the actual uranium mass loadings used in the calculation of the allowable burnups for each array/class.

Justification for Proposed Change

The allowable burnups as a function of cooling time are calculated using the allowable decay heat as an input. Source term calculations are performed to determine the burnup that produces the allowable decay heat for a specified cooling time. One of the key inputs in the source term calculations is the uranium mass loading. The maximum uranium mass loading has historically been specified in the CoC for the HI-STORM system and has always been a quantity derived from the shielding analysis. Therefore, this change is being made to maintain the uranium mass loading values consistent with the shielding analysis used to determine the allowable burnups. The maximum uranium mass loadings in the CoC are not based on the criticality analysis or the thermal analysis and changes to these mass values do not reflect changes in the criticality or thermal areas.

[2] This is not to say the actual point of inoperability is not an issue to be investigated through the root cause evaluation conducted in accordance with the corrective action program, if necessary.

Proposed Change No. 13

Certificate of Compliance, Appendix B, Table 2.1-8:

Revise the maximum allowable burnup for non-fuel hardware inserts as shown in the attached markup to the CoC. These changes are made to maintain consistency with the revised shielding analyses in Chapter 5.

Reason for Proposed Changes

Feedback from our clients has indicated that the allowable burnups versus cooling time for some of the non-fuel hardware is unnecessarily restrictive. Therefore, the allowable burnups for a given cooling time are proposed to be increased for non-fuel hardware inserts.

Justification for Proposed Change

The allowable burnups for the non-fuel hardware are derived from the shielding analysis where a maximum activity of Cobalt-60 is specified for the non-fuel hardware and the burnups are chosen at a given cooling time to assure that the calculated Cobalt-60 activity remains less than the maximum value used in the shielding evaluation. In order to increase the burnups for the non-fuel hardware inserts, the maximum permissible Cobalt-60 activity was increased as identified in Section 5.2.4 and Table 5.2.31 (Attachment 5). The dose rates reported in Chapter 5 of the FSAR were modified to account for this increase in source term.

Proposed Change No. 14

Certificate of Compliance, Appendix B, Section 3.3 and Table 3-1:

a. Change "Exceptions" to "Alternatives" throughout the section.

b. Revise Section 3.3 as shown in the attached mark-up of the CoC to clarify the ASME Code Edition of record for the HI-STORM 100 System. This clarification is proposed to allow the latest effective versions of ASME Sections V and IX to govern the performance of non-destructive examination (NDE) and welding, respectively.

c. Add the new and revised ASME Code alternatives as shown in the attached markup of the CoC.

d. Add "on a case-specific basis" to the requirements related to alternatives to the Code as shown in the attached markup of the CoC.

e. In the second paragraph of the justification for the alternative to Code Article NB-6111, change "process" to "results" and add "relevant" before "findings."

Reason and Justification for Proposed Changes

a. This is an editorial change to make the CoC agree with the regulatory guidance terminology contained in ISG-10.

b. Code Sections V and IX are periodically revised by the ASME to more closely reflect the state of the art in NDE and welding. It is prudent to require the performance of these activities to be in accordance with the latest techniques endorsed by ASME. This change does not affect the design or analysis of the storage system in any manner and is consistent with the current practice of the fabricator of the components governed by the Code.

c. These new and revised Code alternatives are needed to reflect the design drawings and are identical to those previously reviewed and approved by the NRC under separate cover for certain serial number cask components. See NRC letters to Holtec dated March 5th and March 7th, 2002.

d. Based on ISG-10, the intent of this flexibility is to authorize additional Code alternatives on a case-specific basis. This change provides the necessary clarification.

e. This is an editorial change to make the CoC agree with the regulatory guidance terminology contained in ISG-4.

Proposed Change No. 15

Certificate of Compliance, Appendix B, Section 3.5:

Revise the text in Appendix B, Section 3.5.1 as shown in the attached markup to the CoC. These changes are made to maintain consistency with similar wording in CoC Appendix A, Section 5.5.

Reason and Justification Proposed Change

This proposed change clarifies the text to state that the Cask Transfer Facility design criteria requirements do not apply to lifting devices integral to structures governed by the regulation of 10 CFR 50. Our users have stated that the use of the word "outside" as currently written in Section 3.5.1 could be misconstrued to mean anywhere "outdoors", which could include outdoor cranes integral to the Part 50 facility and governed by Part 50 regulatory requirements. This is not the intent of this CoC requirement. The intent of the requirement is to distinguish between 10 CFR Part 50 and Part 72 jurisdiction.

Proposed Change No. 15a

Certificate of Compliance, Appendix A, LCOs 3.1.1, 3.1.2, 3.1.3, and Section 5.6; and Appendix B, Section 1.0, Section 2.1.2, Section 2.1.3, Tables 2.1-1 through 2.1-3, Tables 2.4-4 through 2.4-8, and new Section 3.4.10:

Revise the affected portions of the referenced CoC sections and tables to reflect incorporation of the review guidance contained in ISG-11.

Reason and Justification Proposed Changes

These changes reflect the necessary modifications to the CoC to adopt the revised review guidance contained in ISG-11, Revision 2. The changes to the CoC are consistent with the guidance with regard to the authorization for storage of any fuel cladding material previously approved by NRR for use in a commercial reactor, the elimination of fuel cladding oxidation thickness as a criterion for classifying fuel as damaged, the elimination of the confinement source term penalty factors for high burnup fuel, and the use of a single temperature limit for long-term storage and short-term operating conditions (e.g., vacuum drying). The supporting justification for these changes is contained in proposed changes to FSAR Chapter 4, with conforming changes in FSAR Chapters 1, 2, 5, 8, 10 and 12, and in the thermal-hydraulic calculation package, HI-2002407. Please also see the response to Question 4 in Attachment 1 to this submittal in the section entitled "Information Needed To Review HI-STORM 100, Amendment 2 ."

SECTION II – PROPOSED CHANGES TO THE FSAR

Proposed Change No. 16

Changes to FSAR Chapter 2, Tables 2.2.1 and 2.2.3:

a. Revise the off-normal MPC internal design pressure from 100 psig to 110 psig as shown in FSAR Table 2.2.1 (Attachment 5).

b. Revise the normal temperature limit for the overpack lid top plate as shown in FSAR Table 2.2.3 from 350°F to 450°F in the attached proposed FSAR markups (Attachment 5).

Reason for Proposed Change

a and b. Increasing these pressure and temperature limits is necessary to support the increased thermal loads being proposed elsewhere in this amendment request.

Justification for Proposed Change

a. The off-normal condition is defined as the upset or Level B condition in the ASME Code for which the allowable stresses are 10% greater than for normal service conditions. Therefore, the associated permissible pressure may be increased by 10%.

b. The higher lid top plate temperature has been evaluated and found to be acceptable. The overpack lid lifting evaluations contained in FSAR Section 3.4.3.7 address the adequacy of the threaded holes used to lift the lid for placement on the loaded overpack. This evolution occurs with the lid at ambient temperature. Therefore, this evaluation is unaffected by this change.

The change in the design lid material temperature does not affect the vertical drop or tipover analyses.

The missile impact on the top lid was re-evaluated for the increased lid plate design temperature. The allowable stress is slightly reduced and the safety factor is reduced accordingly, but still shows a safety margin of 33 percent. See proposed FSAR Section 3.4.8.1 (Attachment 5) for more detailed discussion of this event.

Proposed Change No. 17

Changes to FSAR Chapter 3 and Chapter 7

Delete Appendices 3.B thru 3.AS, and 7.A in their entirety and re-locate this information to the supporting calculation package. Create new FSAR Subsection 3.4.4.3.1.8 to address some of these calculation results.

Reason and Justification for Proposed Change

These detailed calculations are of a level of detail that is not appropriate for the FSAR. The supporting calculation packages have been updated as necessary to include the appropriate information deleted from the FSAR appendices.

Proposed Change No. 18

Changes to FSAR Chapter 7

a. Revise the confinement methodology to account for gravitational settling of certain isotopes in the MPC cavity.

b. Revise the confinement analysis to remove the source term penalty factors previously included in ISG-11, Revision 1 for high burnup fuel.

Reason for Proposed Changes

a. The current confinement methodology is unnecessarily conservative by not accounting for the gravitational settling of the fines, volatiles, and crud inside the MPC. This may cause some plants, with large ISFSIs or relatively close site boundaries to significantly overestimate effluent doses from the ISFSI.

b. This is a conforming change to make the confinement analysis agree with the relevant review guidance in ISG-11, Revision 2. Otherwise, the analysis remains consistent with ISG-5, Revision 1.

Justification for Proposed Changes

a. The Holtec MPCs are redundant welded stainless steel canisters designed and manufactured in accordance with the most rigorous ASME Code (Section III, Subsection NB). Analyses show that leakage from the MPC confinement boundary is not credible. However, the licensing basis currently includes analyses of hypothetical (non-mechanistic) leakage during normal, off-normal, and accident conditions of storage. This new methodology incorporates the effects of the gravitational settling, as discussed in NRC Report SMSAB-00-03, "Best-Estimate Offsite Dose from Dry Storage Cask Leakage" while still maintaining other conservatism previously included in the confinement analysis. This methodology is described in the proposed changes to the FSAR contained in Attachment 5. This is also a deviation from ISG-5 in that ISG-5 does not currently recognize gravitational settling. This deviation has been listed in FSAR Table 1.0.3.

b. This change is consistent with ISG-11, Revision 2.

See also Holtec Report HI-992258, "HI-STORM Confinement Analysis" submitted under separate cover

Proposed Change No. 19

Change to FSAR Chapter 11

In Section 11.1.4.3, remove discussion of the three-ducts blocked condition. Remove results currently presented in Table 11.1.2

Reason and Justification for Proposed Change

The three-ducts blocked condition was previously included in the FSAR for comparison purposes only. This comparison is now being removed. The design basis off-normal condition is two ducts blocked and the design basis accident condition (FSAR Section 11.2.13) is all ducts blocked.

Proposed Change No. 20

Changes to FSAR Chapter 13

Replace the detailed discussion of the Holtec QA program throughout Chapter 13 with a short discussion of the program and a reference to the current NRC-approved QA program in Section 13.0 (see Attachment 5). Sections 13.1 through 13.3 and 13.5 are deleted in their entirety. Section 13.4 and Appendices 13.A and 13.B were removed in FSAR Revision 1 after Revision 13 of the Holtec QA Program Manual was approved by the NRC.

Reason for Proposed Change

To remove redundant information.

Justification for Proposed Change

The NRC has approved Revision 13 of Holtec's QA program under 10 CFR 71 (Approval 71-0784, Rev. 3). Holtec also uses this QA program to control activities important to safety that are performed under 10 CFR 72 as permitted by 10 CFR 72.140(d). Including the same, or similar QA program information in FSAR Chapter 13 is unnecessarily redundant. This change is similar to that approved for other Part 72 general certifications (e.g., Fuel Solutions, Docket 72-1026). In accordance with 10 CFR 72.140(d) The Holtec QA program also meets the supplemental recordkeeping requirements of 10 CFR 72.174 for use under Part 72.

NRC FORM 651
(3-1999)
10 CFR 72

U.S. NUCLEAR REGULATORY COMMISSION

CERTIFICATE OF COMPLIANCE FOR SPENT FUEL STORAGE CASKS

The U.S. Nuclear Regulatory Commission is issuing this Certificate of Compliance pursuant to Title 10 of the Code of Federal Regulations, Part 72, "Licensing Requirements for Independent Storage of Spent Nuclear Fuel and High-Level Radioactive Waste" (10 CFR Part 72) This certificate is issued in accordance with 10 CFR 72.238, certifying that the storage design and contents described below meet the applicable safety standards set forth in 10 CFR Part 72, Subpart L, and on the basis of the Final Safety Analysis Report (FSAR) of the cask design. This certificate is conditional upon fulfilling the requirements of 10 CFR Part 72, as applicable, and the conditions specified below.

Certificate No	Effective Date	Expiration Date	Docket Number	Amendment No.	Amendment Date	Package Identification No
1014	05/31/00	06/01/20	72-1014	42		USA/72-1014

Issued To: (Name/Address)

Holtec International
Holtec Center
555 Lincoln Drive West
Marlton, NJ 08053

Safety Analysis Report Title

Holtec International Inc., Final Safety Analysis Report for the HI-STORM 100 Cask System
Docket No. 72-1014

CONDITIONS

This certificate is conditioned upon fulfilling the requirements of 10 CFR Part 72, as applicable, the attached Appendix A (Technical Specifications) and Appendix B – (Approved Contents and Design Features), and the conditions specified below:

1. CASK

   a. Model No : HI-STORM 100 Cask System

   The HI-STORM 100 Cask System (the cask) consists of the following components: (1) interchangeable multi-purpose canisters (MPCs), which contain the fuel; (2) a storage overpack (HI-STORM ~~100 or 100S~~), which contains the MPC during storage; and (3) a transfer cask (HI-TRAC), which contains the MPC during loading, unloading and transfer operations. The cask stores up to 32 pressurized water reactor (PWR) fuel assemblies or 68 boiling water reactor (BWR) fuel assemblies

   b. Description

   The HI-STORM 100 Cask System is certified as described in the Final Safety Analysis Report (FSAR) and in the U. S Nuclear Regulatory Commission's (NRC) Safety Evaluation Report (SER) accompanying the Certificate of Compliance. The cask comprises three discrete components the MPCs, the HI-TRAC transfer cask, and the HI-STORM ~~100 or 100S~~ storage overpack.

   The MPC is the confinement system for the stored fuel It is a welded, cylindrical canister with a honeycombed fuel basket, a baseplate, a lid, a closure ring, and the canister shell. It is made entirely of stainless steel except for the neutron absorbers and ~~optional~~ aluminum heat conduction elements *(AHCEs) AHCEs are optional for those MPCs loaded under the original CoC or Amendment 1. AHCEs are not permitted for those MPCs loaded under Amendment 2 or later amendments to this CoC.* The canister shell, baseplate, lid, vent and drain port cover plates, and closure ring are the main confinement boundary components The honeycombed basket, which is equipped with ~~Boral~~ neutron absorbers, provides criticality control.

1. b. Description (continued)

There are ~~seven~~ *eight* types of MPCs: the MPC-24, MPC-24E, MPC-24EF, MPC-32, *MPC-32F*, MPC-68, MPC-68F, and MPC-68FF. *The number suffix indicates the maximum number of fuel assemblies permitted to be loaded in the MPC.* ~~The MPC-24 and MPC-32 hold up to 24 and 32 PWR fuel assemblies, respectively, that must be intact. The MPC-24E holds up to 24 PWR fuel assemblies, up to four of which may be classified as damaged fuel assemblies. The MPC-24EF holds up to 24 PWR fuel assemblies, up to four of which may be classified as damaged fuel assemblies or in the form of fuel debris. The MPC-68 holds up to 68 BWR fuel assemblies that may be intact or damaged (i.e., with known or suspected cladding defects greater than hairline cracks or pinholes). The number of damaged fuel assemblies is limited to sixteen unless they are Dresden Unit 1 or Humboldt Bay fuel assemblies. The MPC-68F holds up to 68 Dresden Unit 1 or Humboldt Bay BWR fuel assemblies that may be intact, damaged, with up to four in the form of fuel debris (i.e., with known or suspected defects such as ruptured fuel rods, severed fuel rods, and loose fuel pellets). The MPC-68FF holds up to 68 BWR fuel assemblies, up to sixteen of which may be classified as damaged fuel or fuel debris. A maximum of eight fuel assemblies may be in the form of fuel debris. All fuel to be stored in the HI-STORM 100 System must comply with the limits specified in Appendix B to this CoC.~~ All ~~seven~~ *eight* MPC models have the same external dimensions.

The HI-TRAC transfer cask provides shielding and structural protection of the MPC during loading, unloading, and movement of the MPC from the spent fuel pool to the storage overpack. The transfer cask is a multi-walled (carbon steel/lead/carbon steel) cylindrical vessel with a water jacket attached to the exterior. Two ~~types~~ *sizes* of HI-TRAC transfer casks are available, the 125 ton-HI-TRAC and the 100 ton HI-TRAC. The weight designation is the maximum weight of a loaded transfer cask during any loading, unloading or transfer operation. Both transfer cask ~~types~~ *sizes* have identical cavity diameters. The 125 ton HI-TRAC transfer cask has thicker lead and water shielding and larger outer dimensions than the 100 ton HI-TRAC transfer cask.

The HI-STORM 100 or 100S storage overpack provides shielding and structural protection of the MPC during storage. The HI-STORM 100S is a ~~shortened version~~ *variation* of the *HI-STORM 100 overpack design that includes* ~~with~~ a modified lid ~~design incorporating~~ *which incorporates* the air outlet ducts into the lid, *allowing the overpack body to be shortened*. The overpack is a heavy-walled steel and concrete, cylindrical vessel Its side wall consists of plain (un-reinforced) concrete that is enclosed between inner and outer carbon steel shells. The overpack has *a minimum of* four air inlets at the bottom and *a minimum of* four air outlets at the top to allow air to circulate naturally through the cavity to cool the MPC inside. The inner shell has channels attached to its interior surface to guide the MPC during insertion and removal, provide a flexible medium to absorb impact loads, and allow cooling air to circulate through the overpack. A loaded MPC is stored within the HI- STORM 100 or 100S storage overpack in a vertical orientation The HI-STORM 100A is a variant of the HI-STORM 100 family and is outfitted with an extended baseplate and gussets to enable the overpack to be anchored to the concrete storage pad in high seismic applications. The HI-STORM 100A applies to both the ~~standard (~~HI-STORM 100~~)~~ and ~~short (~~HI-STORM 100S~~)~~ overpacks that are classified as the HI-STORM 100A and HI-STORM 100SA, respectively

2. OPERATING PROCEDURES

Written operating procedures shall be prepared for cask handling, loading, movement, surveillance, and maintenance. The user's site-specific written operating procedures shall be consistent with the technical basis described in Chapter 8 of the FSAR

3. ACCEPTANCE TESTS AND MAINTENANCE PROGRAM

Written cask acceptance tests and maintenance program shall be prepared consistent with the technical basis described in Chapter 9 of the FSAR.

4. QUALITY ASSURANCE

Activities in the areas of design, purchase, fabrication, assembly, inspection, testing, operation, maintenance, repair, modification of structures, systems and components, and decommissioning that are important to safety shall be conducted in accordance with a Commission-approved quality assurance program which satisfies the applicable requirements of 10 CFR Part 72, Subpart G, and which is established, maintained, and executed with regard to the cask system.

5. HEAVY LOADS REQUIREMENTS

Each lift of an MPC, a HI-TRAC transfer cask, or a HI-STORM 100 or 100S overpack must be made in accordance to the existing heavy loads requirements and procedures of the licensed facility at which the lift is made. A plant-specific safety review (under 10 CFR 50.59 or 10 CFR 72.48, if applicable) is required to show operational compliance with existing plant specific heavy loads requirements. Lifting operations outside of structures governed by 10 CFR Part 50 must be in accordance with Section 5.5 of Appendix A and/or Sections 3 4.6 and Section 3.5 of Appendix B to this certificate, as applicable.

6. APPROVED CONTENTS

Contents of the HI-STORM 100 Cask System must meet the fuel specifications given in Appendix B to this certificate.

7. DESIGN FEATURES

Features or characteristics for the site, cask, or ancillary equipment must be in accordance with Appendix B to this certificate

8. CHANGES TO THE CERTIFICATE OF COMPLIANCE

The holder of this certificate who desires to make changes to the certificate, which includes Appendix A (Technical Specifications) and Appendix B (Approved Contents and Design Features), shall submit an application for amendment of the certificate.

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 Cask Systems (for each thermally unique MPC basket design - MPC-24/24E/24EF, MPC-32/*32F*, and MPC-68/68F/68FF) placed into service by any user 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 FSAR.

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. PRE-OPERATIONAL TESTING AND TRAINING EXERCISE

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 step 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, moisture removal (by vacuum drying or forced helium dehydration, as applicable), helium backfilling, and leakage testing. (A mockup may be used for this dry-run exercise.)

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 Cask System at the ISFSI.

j. HI-STORM 100 Cask System unloading, including cooling fuel assemblies, flooding MPC cavity, removing MPC lid welds. (A mockup may be used for this dry-run exercise.)

11. 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 All HI-STORM 100 Cask Systems must be fabricated and used in accordance with CoC No 1014, Amendment No ~~1~~2; except that general licensees may use the HI-STORM 100 Cask Systems that were fabricated in accordance with the original CoC *or Amendment 1*.

FOR THE U S. NUCLEAR REGULATORY COMMISSION

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

Attachments
1. Appendix A
2. Appendix B

CERTIFICATE OF COMPLIANCE NO. 1014

APPENDIX A

TECHNICAL SPECIFICATIONS

FOR THE HI-STORM 100 CASK SYSTEM

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 or demoisturizer exit gas temperature 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 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
	<u>AND</u>	
	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	~~For those MPCs containing all moderate burnup (≤ 45,000 MWD/MTU) fuel assemblies, verify MPC cavity vacuum drying pressure is within the limit specified in Table 3-1 for the applicable MPC model.~~ ~~OR~~ ~~For those MPCs containing fuel assemblies of any authorized burnup, while using the recirculating helium method to dehydrate the MPC cavity, verify that the gas temperature exiting the demoisturizer is ≤ 21°F for ≥ 30 minutes.~~ *Verify that the MPC cavity has been dried in accordance with the applicable limits in Table 3-1.*	Once, prior to TRANSPORT OPERATIONS
SR 3.1.1.2	Verify MPC helium backfill ~~density or pressure~~ *quantity* is within the limit specified in Table 3-~~1~~*2* 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 ≤ *5.0 x 10*$^{-6}$ atm-cc/sec. ~~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.~~ *Measure SFSC dose rates in accordance with the Radiation Protection Program.* AND B.2.1 Restore SFSC Heat Removal System to operable status. OR B.2.2 Transfer the MPC into a TRANSFER CASK.	Immediately and once per 12 hours thereafter ~~48~~*64* hours, *if MPC heat load is ≤ 28.74 kW* *OR* *25 hours, if MPC heat load is > 28.74 kW* ~~48~~*64* hours, *if MPC heat load is ≤ 28.74 kW* *OR* *25 hours, if MPC heat load is > 28.74 kW*

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 ≤ ~~126~~ *158* °F.	24 hours

3.1 SFSC INTEGRITY

3.1.3 Fuel Cool-Down

LCO 3.1.3 The MPC *cavity bulk* 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 *cavity bulk* helium ~~gas exit~~ temperature not within limit.	A.1 Establish MPC *cavity bulk* 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	~~22 hours~~ *Immediately*

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.1.3.1	~~Verify~~*Ensure via analysis or direct measurement of MPC exit gas temperature that* MPC *cavity bulk* helium ~~gas exit~~ temperature *is* within limit.	Prior to MPC re-flooding operations.

3.2 ~~SFSC RADIATION PROTECTION~~ *Deleted.*

3.2.1 ~~TRANSFER CASK Average Surface Dose Rates~~ *Deleted.*

LCO 3.2.1 *Deleted.* ~~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 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~~ ~~48 hours~~

~~(continued)~~

~~ACTIONS~~
~~(continued)~~

~~CONDITION~~	~~REQUIRED ACTION~~	~~COMPLETION TIME~~
~~B. Required Action and associated Completion Time not met.~~	~~B.1 Remove all fuel assemblies from the TRANSFER CASK~~	~~30 days~~

~~SURVEILLANCE REQUIREMENTS~~

~~SURVEILLANCE~~		~~FREQUENCY~~
~~SR 3.2.1.1~~	~~Verify average surface dose rates of the TRANSFER CASK loaded with an MPC containing fuel assemblies are within limits.~~ ~~A minimum of 12 dose rate measurements shall be taken on the side of the TRANSFER CASK in three sets of four measurements. One measurement set shall be taken approximately at the cask mid-height plane. The second and third measurement sets shall be taken approximately 60 inches above and below the mid-height plane, respectively. Within each set, the measurement locations shall be approximately 90 degrees apart around the circumference of the cask. Dose rates shall be measured between the radial ribs of the water jacket. The average of the 12 dose rate measurements shall be compared to the limit specified in LCO 3.2.1 a.i or b.i, as applicable.~~ ~~A minimum of four (4) top lid dose rates shall be measured at locations approximately half way between the edge of the hole in the top lid and the outer edge of the top lid, 90 degrees apart around the circumference of the top lid. The average of these four dose rates shall be compared to the limit specified in LCO 3.2.1.a.ii or b.ii, as applicable.~~	~~Once, prior to TRANSPORT OPERATIONS~~

~~FIGURE 3 2 1-1 INTENTIONALLY DELETED~~

3.2 ~~SFSC RADIATION PROTECTION~~ *Deleted.*

3.2.2 ~~TRANSFER CASK Surface Contamination~~ *Deleted.*

LCO 3.2.2 *Deleted.* ~~Removable contamination on the exterior surfaces of the TRANSFER CASK and accessible portions of the MPC shall each not exceed:~~

~~a. 1000 dpm/100 cm² from beta and gamma sources~~

~~b. 20 dpm/100 cm² from alpha sources.~~

~~NOTE~~

~~This LCO is not applicable to the TRANSFER CASK if MPC transfer operations occur inside the FUEL BUILDING.~~

~~APPLICABILITY: During TRANSPORT OPERATIONS.~~

~~ACTIONS~~

~~NOTE~~

~~Separate Condition entry is allowed for each TRANSFER CASK.~~

~~CONDITION~~	~~REQUIRED ACTION~~	~~COMPLETION TIME~~
~~A. TRANSFER CASK or MPC removable surface contamination limits not met.~~	~~A.1 Restore removable surface contamination to within limits.~~	~~7 days~~

~~SURVEILLANCE REQUIREMENTS~~

~~SURVEILLANCE~~		~~FREQUENCY~~
~~SR 3.2.2.1~~	~~Verify that the removable contamination on the exterior surfaces of the TRANSFER CASK and accessible portions of the MPC containing fuel is within limits.~~	~~Once, prior to TRANSPORT OPERATIONS~~

3.2 ~~SFSC RADIATION PROTECTION~~ *Deleted.*

3.2.3 ~~OVERPACK Average Surface Dose Rates~~ *Deleted.*

LCO 3.2.3 *Deleted* ~~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. 45 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.~~ ~~AND~~ ~~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.~~	~~24 hours~~ ~~48 hours~~
~~B. Required Action and associated Completion Time not met.~~	~~B.1 Remove all fuel assemblies from the SFSC.~~	~~30 days~~

~~SURVEILLANCE REQUIREMENTS~~

~~SURVEILLANCE~~		~~FREQUENCY~~
~~SR 3.2.3.1~~	~~Verify average surface dose rates of the OVERPACK loaded with an MPC containing fuel assemblies are within limits.~~	~~Once, within 24 hours after beginning STORAGE OPERATIONS~~
	~~A minimum of 12 dose rate measurements shall be taken on the side of the OVERPACK in three sets of four measurements. One measurement set shall be taken approximately at the cask mid-height plane, 90 degrees apart around the circumference of the cask. The second and third measurement sets shall be taken approximately 60 inches above and below the mid-height plane, respectively, also 90 degrees apart around the circumference of the cask. The average of the 12 dose rate measurements shall be compared to the limit specified in LCO 3.2.3.a.~~	
	~~A minimum of five (5) dose rate measurements shall be taken on the top of the OVERPACK. One dose rate measurement shall be taken at approximately the center of the lid and four measurements shall be taken at locations on the top concrete shield, approximately half way between the center and the edge of the top shield, 90 degrees apart around the circumference of the lid. The average of the 5 dose rate measurements shall be compared to the limit specified in LCO 3.2.3.b.~~	
	~~A dose rate measurement shall be taken adjacent to each inlet and outlet vent duct. The average of the 8 inlet and outlet duct dose rates shall be compared to the limit specified in LCO 3.2.3.c.~~	

~~Figure 3.2.3-1 INTENTIONALLY DELETED~~

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 ≤ 5 0 wt% ^{235}U. ≥ 400 ppmb

b. MPC-24E *or MPC-24EF (all INTACT FUEL ASSEMBLIES)* 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 ≤ 5.0 wt% ^{235}U: ≥ 300 ppmb

c. *Deleted.* ~~MPC-32 with all fuel assemblies having an initial enrichment ≤ 4.1 wt% ^{235}U: ≥ 1900 ppmb~~

d. *Deleted.* ~~MPC-32 with one or more fuel assemblies having an initial enrichment > 4.1 and ≤ 5.0 wt% ^{235}U: ≥ 2600 ppmb~~

e. *MPC-24E or MPC-24EF (one or more DAMAGED FUEL ASSEMBLIES or FUEL DEBRIS) with one or more fuel assemblies having an initial enrichment > 4.0 wt% ^{235}U and ≤ 5.0 wt% ^{235} U: ≥ 600 ppmb*

f. *MPC-32/32F: Minimum soluble boron as required by the table below.*

Fuel Assembly Array/Class	*All INTACT FUEL ASSEMBLIES*		*One or more DAMAGED FUEL ASSEMBLIES or FUEL DEBRIS*	
	Initial Enrichment ≤ 4.1 wt% ^{235} U (ppmb)	*Initial Enrichment ≤ 5.0 wt% ^{235} U (ppmb)*	*Initial Enrichment ≤ 4.1 wt% ^{235} U (ppmb)*	*Initial Enrichment ≤ 5.0 wt% ^{235} U (ppmb)*
14x14A/B/C/D/E	*1,300*	*1,900*	*1,500*	*2,300*
15x15A/B/C/G	*1,800*	*2,400*	*1,900*	*2,700*
15x15D/E/F/H	*1,900*	*2,600*	*2,100*	*2,900*
16x16A	*1,300*	*1,900*	*1,500*	*2,300*
17x17A/B/C	*1,900*	*2,600*	*2,100*	*2,900*

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
	AND	
	A.2 Suspend positive reactivity additions.	Immediately
	AND	
	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. --- SR 3.3.1.1 Verify boron concentration is within the applicable limit using two independent measurements.	Within 4 hours of entering the Applicability of this LCO. AND Once per 48 hours thereafter.

Table 3-1
MPC Cavity Drying Limits (Note 1)

Fuel Burnup (MWD/MTU)	MPC Heat Load (kW)	Method of Moisture Removal (Notes 2 and 3)	Other Requirements (Notes 4 and 5)
All Assemblies ≤ 45,000	≤ 10	VDS or FHD	None
All Assemblies ≤ 45,000	> 10 and ≤ 15	VDS or FHD	If VDS is used, annulus water recirculation is required
All Assemblies ≤ 45,000	> 10 and ≤ 25	VDS or FHD	If VDS is used, fuel cladding hoop stress shall be shown by analysis to be ≤ 90 MPa
All Assemblies ≤ 45,000	> 25 and ≤ 29	VDS or FHD	If VDS is used, annulus water recirculation is required AND fuel cladding hoop stress shall be shown by analysis to be ≤ 90 MPa
All Assemblies ≤ 45,000	> 29 and ≤ 40	FHD	None
One or more assemblies > 45,000	≤ 40	FHD	None

Notes:

1. If the limits in two or more rows apply, the user may choose one set of limits to implement.

2. VDS means Vacuum Drying System. The acceptance criterion for VDS is MPC cavity pressure shall be ≤ 3 torr for ≥ 30 minutes.

3. FHD means Forced Helium Dehydration System. The acceptance criterion for the FHD System is gas temperature exiting the demoisturizer shall be ≤ 21°F for ≥ 30 minutes.

4. Annulus water recirculation means a constant flow of water through the annulus between the MPC and the HI-TRAC inner shell during moisture removal operations (i.e., beginning prior to 10 torr descending and ending when helium backfill operations commence).

5. Fuel cladding hoop stress calculations may be performed using "best estimate" inputs.

Table 3-~~1~~2
MPC *Helium Backfill* ~~Model-Dependent~~ Limits[1]

MPC MODEL	LIMITS
~~1.~~ MPC-24/24E/24EF	
~~a. MPC Cavity Vacuum Drying Pressure~~	~~≤ 3 torr for ≥ 30 min~~
~~b. MPC Helium Backfill†~~	
i. Cask Heat Load ≤ 27.77 kW (MPC-24) or ≤ 28.17 kW (MPC-24E/EF)	~~0.1212 +0/-10% g-moles/l~~ ~~OR~~ ≥ 29.3 psig and ≤ ~~33.3~~*61.4* psig
ii Cask Heat Load > 27.77 kW (MPC-24) or > 28.17 kW (MPC-24E/EF)	*≥ 40.9 psig and ≤ 61.4 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†~~	
i. Cask Heat Load ≤ 28.19 kW	0.1218 +0/-10% g-moles/l OR ≥ 29 3 psig and ≤ ~~33.3~~*76.3* psig
ii. Cask Heat Load > 28.19 kW	*≥ 40.9 psig and ≤ 76.3 psig*
~~c. MPC Helium Leak Rate~~	~~≤ 5.0E-6 atm cc/sec (He)~~
~~3.~~ MPC-32/32F	
~~a. MPC Cavity Vacuum Drying Pressure~~	~~≤ 3 torr for ≥ 30 min~~
~~b. MPC Helium Backfill Pressure~~[1]	
i. Cask Heat Load ≤ 28.74 kW	≥ 29.3 psig and ≤ ~~33.3~~*47.9* psig
ii. Cask Heat Load > 28.74 kW	*≥ 41.3 psig and ≤ 47.9 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%. Pressure range is at a reference temperature of 70°F

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 FSAR 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 FSAR 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 Final Safety Analysis Report for the HI-STORM 100 Cask System. The program shall ensure that these alternative analyses are documented and controlled.

(continued)

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.

b. For the transport of OVERPACKS to be anchored to the ISFSI pad, the following requirements apply:

1. Except as provided in 5.5.b.2, user shall determine allowable OVERPACK lift height limit(s) above the transport route surface(s) based on site-specific transport route conditions. The lift heights shall be determined by evaluation or analysis, based on limiting the design basis cask deceleration during a postulated drop event to ≤ 45 g's at the top of the MPC fuel basket. Evaluations and/or analyses shall be performed using methodologies consistent with those in the HI-STORM 100 FSAR.

2. The 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.

(continued)

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 cask/lid assemblage)
2. See Technical Specification 5.5.a.3 and 4
3. See Technical Specification 5.5.a.3.

(continued)

5.6 ~~Fuel Cladding Oxide Thickness Evaluation Program~~ *Deleted.*

~~A Fuel Cladding Oxide Thickness Evaluation Program shall be developed and implemented to determine the average fuel cladding oxide thickness of high burnup (> 45,000 MWD/MTU) spent nuclear fuel assemblies proposed to be stored in the HI-STORM 100 System. The program may use direct physical measurements or an appropriate predictive methodology with due consideration of all significant variables (e.g., in-core flux, cycle length and number, power history, coolant temperature profile, coolant chemistry, and metallurgy of the fuel cladding material) to determine the average oxide thickness on the fuel cladding. If a predictive methodology is used to determine average fuel cladding oxide thickness, a sufficient number of fuel cladding thickness measurements shall be made to adequately benchmark the methodology.~~

~~In order to classify a high burnup spent fuel assembly as an INTACT FUEL ASSEMBLY, the loss of fuel cladding to oxidation must not increase the fuel cladding inner radius-to-fuel cladding thickness ratio above 10.5 for PWR fuel assemblies or 9.5 for BWR fuel assemblies. The criterion is met if the computed or measured average oxidation layer thickness of all fuel rods is less than the maximum allowable average fuel cladding oxidation thickness. The maximum allowable average fuel cladding oxidation layer thickness shall be calculated using the following formula:~~

$$\sout{t_{ox} = \left(t_{nom} - \frac{0.5 \times d_{nom} - t_{nom}}{W} \right) \times 25{,}400}$$

~~where:~~

~~t_{ox} = the maximum allowable average oxidation layer thickness (micrometers)~~

~~W = the applicable maximum allowable fuel cladding inner radius-to fuel cladding thickness ratio (10.5 or 9.5)~~

~~t_{nom} = the nominal, pre-irradiated fuel cladding thickness (inches)~~

~~d_{nom} = the nominal, pre-irradiated fuel cladding outer diameter (inches)~~

~~A high burnup spent fuel assembly shall be considered a DAMAGED FUEL ASSEMBLY if the computed or measured average oxidation layer thickness on any fuel rod exceeds the limit determined above.~~

(continued)

5.7 Radiation Protection Program

5.7.1 Each cask user shall establish and maintain a radiation protection program governing cask loading and unloading, as well as ISFSI operations, including transport of the loaded OVERPACK or TRANSFER CASK outside of facilities governed by 10 CFR Part 50. The radiation protection program shall include appropriate controls for measuring dose rates, ensuring compliance with applicable regulations, and implementing actions to maintain personnel occupational exposures As Low As Reasonably Achievable (ALARA). The program shall also address the control of loose contamination of cask components. Contamination limits shall be established, as appropriate, in accordance with applicable guidance for the release of contaminated material from the Part 50 facility (i.e., NRC I&E Circular 81-07 or other appropriate guidance).

5.7.2 In performing shielding calculations or evaluations under this program to demonstrate compliance with regulatory dose limits or in support of changes to the cask design or procedures made under 10 CFR 72.48, the methodology described in the HI-STORM FSAR shall be used, and the following methods and input values shall be used:

a. A three-dimensional radiation transport code shall be used.

b. Source terms shall be shall be calculated using computer codes that have been reviewed and approved by the NRC for the burnups being analyzed.

c. Cobalt-59 impurity levels in source components shall be $\geq$ 0.8 gm/kg for stainless steel and $\geq$ 1.0 gm/kg for Inconel.

CERTIFICATE OF COMPLIANCE NO. 1014

APPENDIX B

APPROVED CONTENTS AND DESIGN FEATURES

FOR THE HI-STORM 100 CASK SYSTEM

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 that 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), Wet Annular Burnable Absorbers (WABAs), Rod Cluster Control Assemblies (RCCAs), Control Element Assemblies (CEAs), water displacement guide tube plugs, ~~and~~, orifice rod assemblies, *and vibration suppressor inserts*.

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 OPERATIONS include 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.
ZR	*ZR means any zirconium-based fuel cladding material authorized for use in a commercial nuclear power plant reactor.*

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 (or other alloy of zirconium)~~ *ZR* 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 MPCs partially loaded with array/class 6x6A, 6x6B, 6x6C, 7x7A, or 8x8A fuel assemblies, all remaining ~~Zircaloy (or other alloy of zirconium)~~ *ZR* 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 (or other alloy of zirconium)~~ *ZR* channels with the exception of array/class 10x10D and 10x10E fuel assemblies, which may be stored with or without ~~Zircaloy~~*ZR* or stainless steel channels.

2.1.2 Uniform Fuel Loading

Any authorized fuel assembly may be stored in any fuel cell location, subject to other restrictions related to DAMAGED FUEL, FUEL DEBRIS, and NON-FUEL HARDWARE specified in the CoC. ~~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 (≥ 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.~~

(continued)

2.0 Approved Contents

2.1 Fuel Specifications and Loading Conditions (cont'd)

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)~~ *ZR* cladding. Figures 2.1-1 through 2.1-4 define the regions for the MPC-24, MPC-24E, MPC-24EF, MPC-32, *MPC-32F,* MPC-68, and MPC-68FF models, respectively[1]. Fuel assembly burnup, decay heat, and cooling time limits for regionalized loading are specified in *Section 2.4.2.* ~~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.

2.3 Deviations from Cask Contents Requirements

Proposed alternatives to the contents listed in Section 2.0 may be authorized on a case-specific basis by the Director of the Office of Nuclear Material Safety and Safeguards or designee. The request for such alternative contents should demonstrate that:

2.3.1 The proposed alternative contents would provide an equivalent level of safety, and

2.3.2 The proposed alternative contents are consistent with the applicable requirements.

Requests for case-specific NRC approval of alternatives to contents shall be submitted in accordance with 10 CFR 72.4 by the certificate holder. Case-specific alternatives approved pursuant to this section shall be incorporated permanently into the CoC by the certificate holder in accordance with 10 CFR 72.244, as part of the next scheduled amendment request.

[1] These figures are only intended to distinguish the fuel loading regions. Other details of the basket design are illustrative and may not reflect the actual basket design details. The design drawings should be consulted for basket design details.

LEGEND:

REGION 1:

REGION 2:

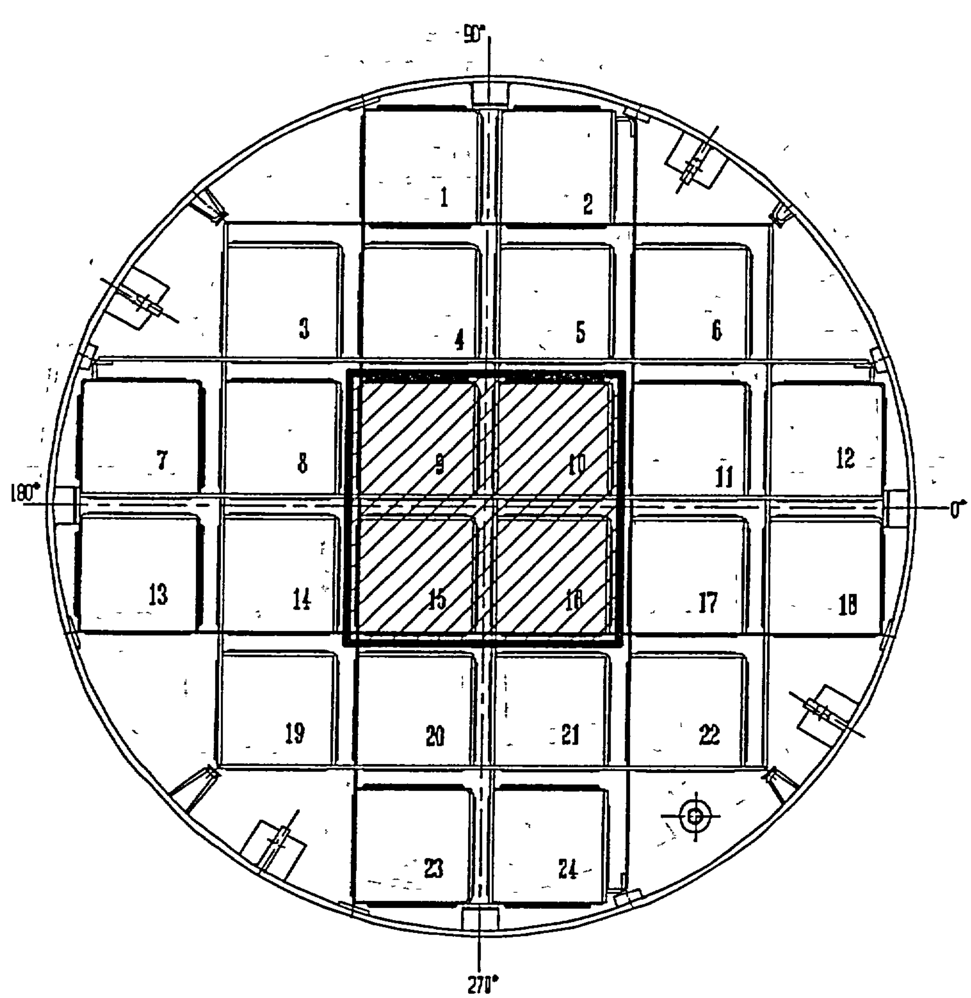

FIGURE 2.1-1
FUEL LOADING REGIONS - MPC-24

LEGEND:

REGION 1: (hatched)

REGION 2: (blank)

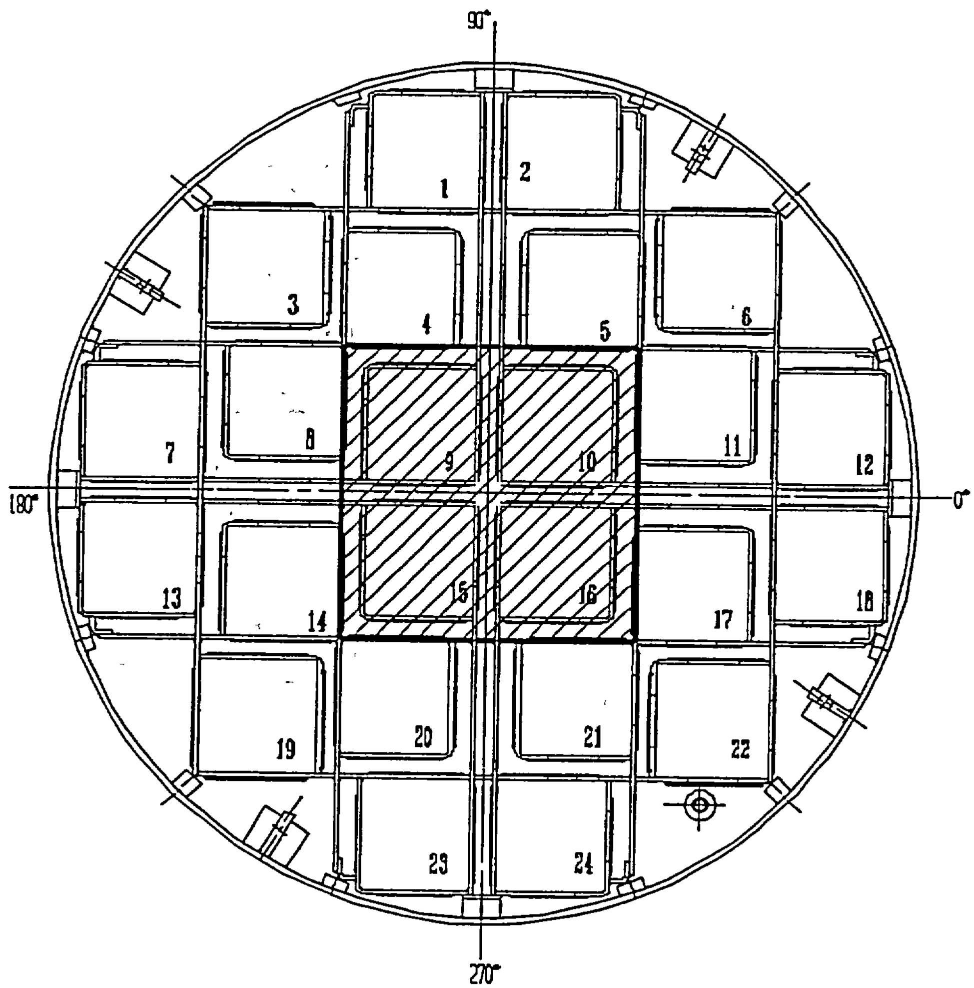

FIGURE 2.1-2

FUEL LOADING REGIONS - MPC-24E/24EF

LEGEND:

REGION 1: (hatched)

REGION 2: (blank)

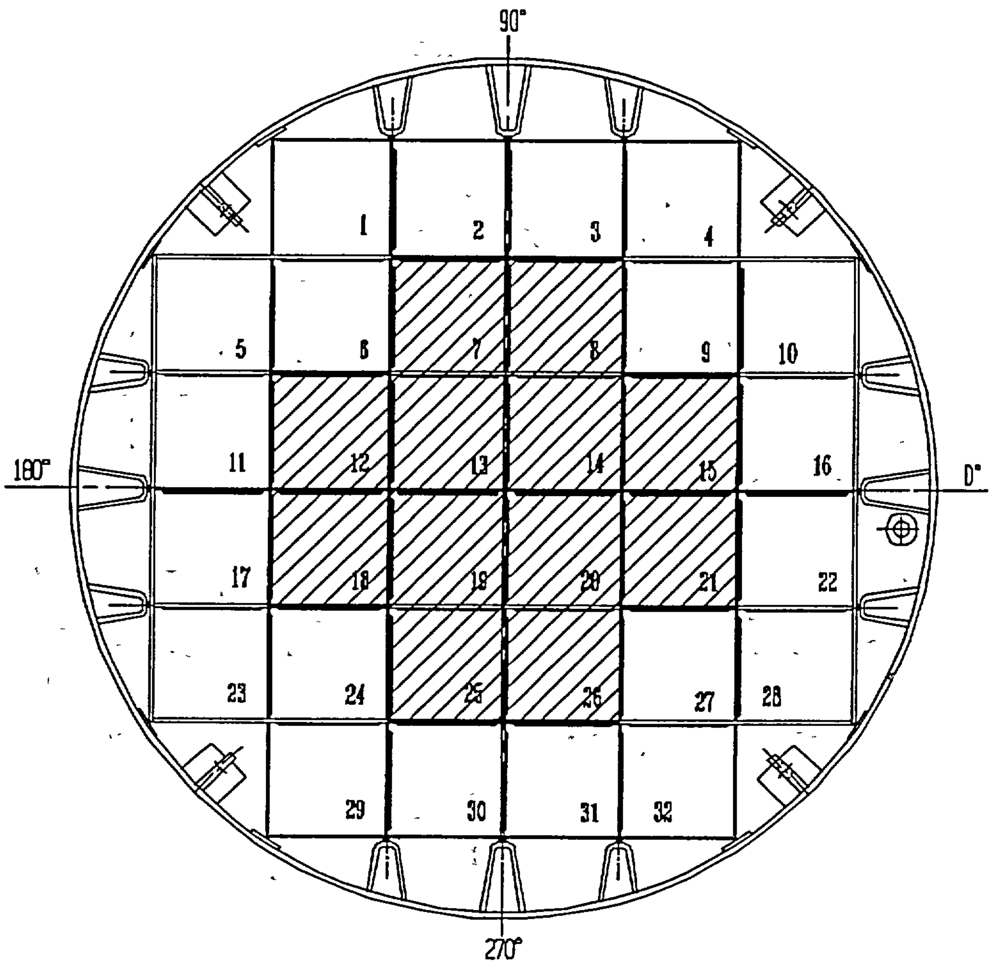

FIGURE 2.1-3

FUEL LOADING REGIONS - MPC-32/32F

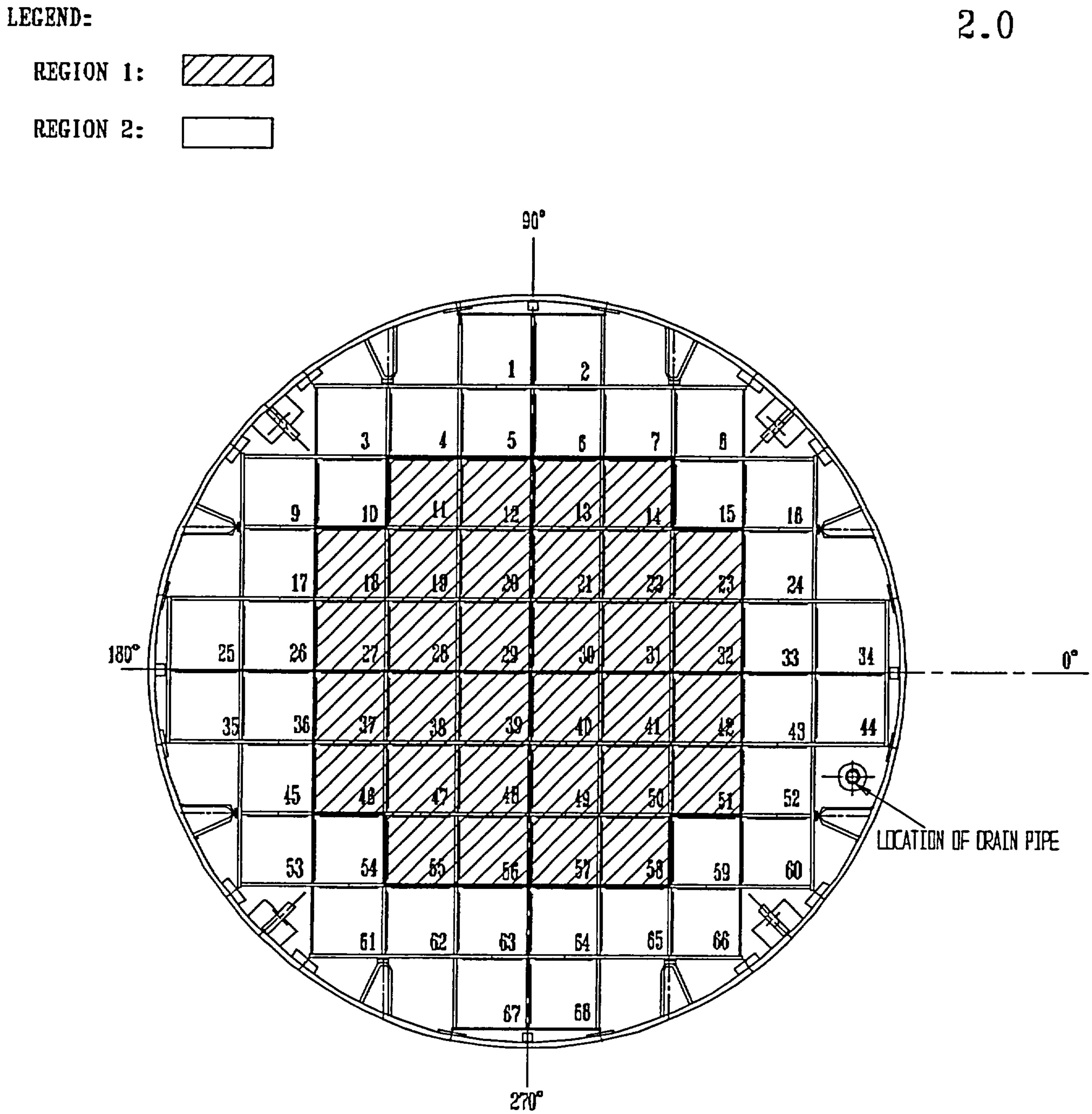

FIGURE 2.1-4

FUEL LOADING REGIONS - MPC-68/68FF

Table 2.1-1 (page 1 of 3~~3~~*9*)
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)~~ *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	Cooling time and average burnup as specified in *Section 2.4.* ~~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 3~~3~~9)
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 *Section 2.4.* ~~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. Deleted.

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

Note 1: Fuel assemblies containing BPRAs, TPDs, WABAs, water displacement guide tube plugs, ~~or~~ orifice rod assemblies, *or vibration suppressor inserts* may be stored in any fuel cell location. Fuel assemblies containing CRAs, RCCAs, CEAs, 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~~9)
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 channels, and meeting the following specifications:

a. Cladding Type:	~~Zircaloy (Zr)~~ *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 *Section 2.4*. ~~Tables 2.1-4 or 2.1-6.~~

Table 2.1-1 (page 4 of 3~~3~~9)
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 *Section 2.4.* ~~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 ~~39~~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)~~ *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 *Section 2.4.* ~~Tables 2.1-4 or 2.1-6.~~

Table 2.1-1 (page 6 of ~~33~~9)
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 *Section 2.4.* ~~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~~9)
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)~~ *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 8 of ~~33~~9)
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)~~ *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 ≥ 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:	≤ 550 lbs, including channels and DFC

Table 2.1-1 (page 9 of ~~3~~39)
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)~~ *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 3~~3~~9)
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 DAMAGED 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.

E. FUEL DEBRIS is not authorized for loading in the MPC-68.

Table 2.1-1 (page 11 of ~~33~~9)
Fuel Assembly Limits

III. MPC MODEL: MPC-68F

A. Allowable Contents

1. Uranium oxide, BWR INTACT FUEL ASSEMBLIES, with or without ~~Zircaloy (Zr)~~ *ZR* 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)~~ *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~~9)
Fuel Assembly Limits

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

A. Allowable Contents (continued)

2. Uranium oxide, BWR DAMAGED FUEL ASSEMBLIES, with or without ~~Zircaloy (Zr)~~ *ZR* 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)~~ *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:	≤ 550 lbs, including channels and DFC

Table 2.1-1 (page 13 of ~~33~~9)
Fuel Assembly Limits

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

A. Allowable Contents (continued)

3. Uranium oxide, BWR FUEL DEBRIS, with or without ~~Zircaloy (Zr)~~ *ZR* 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)~~ *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~~9)
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 (Zr)~~ *ZR* 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)~~ *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~~9)
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 (Zr)~~ *ZR* 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)~~ *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~~9)
Fuel Assembly Limits

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

A. Allowable Contents (continued)

6. Mixed Oxide (MOX), BWR FUEL DEBRIS, with or without ~~Zircaloy (Zr)~~ *ZR* 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)~~ *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 ≥ 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:	≤ 550 lbs, including channels and DFC

Table 2.1-1 (page 17 of ~~33~~9)
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)~~ *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 339)
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~~9)
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)~~ *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 *Section 2.4.* ~~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 3~~3~~9)
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 *Section 2.4.* ~~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~~9)
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)~~ *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.~~ *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 *Section 2.4.* ~~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 3~~3~~9)
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 *Section 2.4.* ~~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, TPDs, WABAs, water displacement guide tube plugs, ~~or~~ orifice rod assemblies, *or vibration suppressor inserts* may be stored in any fuel cell location. Fuel assemblies containing CRAs, RCCAs, CEAs, 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 23 of 3~~3~~9)
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)~~ *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 *Section 2.4.* ~~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 3~~3~~9)
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 *Section 2.4.* ~~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 245 of 339)
Fuel Assembly Limits

V. MPC MODEL: MPC-32 (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:	*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 Section 2.4.*
iii. NON-FUEL HARDWARE	*As specified in Table 2.1-8.*

Table 2.1-1 (page 2~~2~~6 of 3~~3~~9)
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	*≤ 710 Watts.*
ii. All Other Array/Classes	*As specified in Section 2.4.*
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 eight (8) DAMAGED FUEL ASSEMBLIES in DAMAGED FUEL CONTAINERS, stored in fuel storage locations 1, 4, 5, 10, 23, 28, 29, and/or 32. The remaining MPC-32 fuel storage locations may be filled with PWR INTACT FUEL ASSEMBLIES meeting the applicable specifications.*

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

Note 1: Fuel assemblies containing BPRAs, TPDs, WABAs, water displacement guide tube plugs, ~~or~~ orifice rod assemblies, *or vibration suppressor inserts* may be stored in any fuel cell location. Fuel assemblies containing CRAs, RCCAs, CEAs, or APSRs may only 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~~7 of ~~33~~9)
Fuel Assembly Limits

VI. MPC MODEL: MPC-68FF

A. Allowable Contents

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

a. Cladding Type:	~~Zircaloy (Zr)~~ *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, 6x6B, 6x6C, 7x7A, and 8x8A	Cooling time ≥ 18 years and an average burnup ≤ 30,000 MWD/MTU (or MTU/MTIHM)
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 *Section 2.4.* ~~Tables 2.1-4 or 2.1-6.~~

Table 2.1-1 (page 268 of 339)
Fuel Assembly Limits

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

A. Allowable Contents (continued)

e. Decay Heat Per Assembly

i. Array/Classes 6x6A, 6X6b, 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 *Section 2.4.* ~~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 ~~27~~9 of ~~33~~9)
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)~~ *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	≤ 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 ≥ 18 years and an average burnup ≤ 30,000 MWD/MTU (or MWD/MTIHM).
ii. Array/Class 8x8F	Cooling time ≥ 10 years and an average burnup ≤ 27,500 MWD/MTU.
iii. Array/Class 10x10D and 10x10E	Cooling time ≥ 10 years and an average burnup ≤ 22,500 MWD/MTU.
iv. All Other Array/Classes	As specified in *Section 2.4.* ~~Tables 2.1-4 or 2.1-6.~~

Table 2.1-1 (page ~~28~~ *30* of 33~~9~~)
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 *Section 2.4.* ~~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 ~~29~~ 31 of 3~~3~~9)
Fuel Assembly limits

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

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

1. For fuel assembly array/classes 6x6A, 6X6B, 6x6C, 7x7A, or 8x8A, up to 68 BWR INTACT FUEL ASSEMBLIES and/or DAMAGED FUEL ASSEMBLIES. Up to eight (8) DFCs containing FUEL DEBRIS from these array/classes may be stored.

2. For all other array/classes, 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:

   i. Uranium Oxide BWR INTACT FUEL ASSEMBLIES; or

   ii. 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 3~~0~~2 of 3~~3~~9)
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)~~ *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 *Section 2.4.* ~~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~~ 3 of ~~33~~ 9)
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 *Section 2.4.* ~~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 3~~2~~*4* of 3~~3~~*9*)
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)~~ *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.~~ *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 *Section 2.4.* ~~Tables 2.1-4 or 2.1-6.~~
iii. NON-FUEL HARDWARE	As specified in Table 2.1-8.

Table 2.1-1 (page 335 of 339)
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 *Section 2.4.* ~~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, TPDs, WABAs, water displacement guide tube plugs, ~~or~~ orifice rod assemblies, *or vibration suppressor inserts* may be stored in any fuel cell location. Fuel assemblies containing CRAs, RCCAs, CEAs, 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 36 of 39)
Fuel Assembly Limits

VIII. MPC MODEL: MPC-32F

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:	*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 Section 2.4.*
iii. NON-FUEL HARDWARE	*As specified in Table 2.1-8.*

Table 2.1-1 (page 37 of 39)
Fuel Assembly Limits

VIII. MPC MODEL: MPC-32F (cont'd)

A. Allowable Contents (cont'd)

d. Decay Heat Per Assembly	
i. Array/Classes 14x14D, 14x14E, and 15x15G	*≤ 710 Watts.*
ii. All Other Array/Classes	*As specified in Section 2.4.*
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)*

Table 2.1-1 (page 38 of 39)
Fuel Assembly Limits

VIII. MPC MODEL: MPC-32F (cont'd)

A. Allowable Contents (cont'd)

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:	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 Section 2.4.
iii. NON-FUEL HARDWARE	As specified in Table 2.1-8.

Table 2.1-1 (page 39 of 39)
Fuel Assembly Limits

VIII. MPC MODEL: MPC-32F (cont'd)

A. Allowable Contents (cont'd)

d. Decay Heat Per Assembly

i. Array/Classes 14x14D, 14x14E, and 15x15G	≤ 710 Watts.
ii. All Other Array/Classes	As specified in Section 2.4.
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 eight (8) DAMAGED FUEL ASSEMBLIES and/or FUEL DEBRIS in DAMAGED FUEL CONTAINERS, stored in fuel storage locations 1, 4, 5, 10, 23, 28, 29, and/or 32. The remaining MPC-32 fuel storage locations may be filled with PWR INTACT FUEL ASSEMBLIES meeting the applicable specifications.

Note 1: Fuel assemblies containing BPRAs, TPDs, WABAs, water displacement guide tube plugs, orifice rod assemblies, or vibration suppressor inserts may be stored in any fuel cell location. Fuel assemblies containing CRAs, RCCAs, CEAs, or APSRs may only 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 2.1-2 (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~~ *ZR*	~~Zr~~ *ZR*	SS	SS
Design Initial U (kg/assy.) (Note 3)	≤ ~~407~~ *361*	≤ ~~407~~ *408*	≤ ~~425~~ *433*	≤ 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, 32, or *32F* with soluble boron credit - see Note~~s~~ 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*	~~Zr~~ *ZR*	~~Zr~~ *ZR*	~~Zr~~ *ZR*
Design Initial U (kg/assy.) (Note 3)	≤ ~~464~~ *468*	≤ ~~464~~ *468*	≤ ~~464~~ *468*	≤ ~~475~~ *495*	≤ ~~475~~ *495*	≤ ~~475~~ *495*
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, 32, or *32F* with soluble boron credit - see Note~~s~~ 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~~ *ZR*	~~Zr~~ *ZR*	~~Zr~~ *ZR*
Design Initial U (kg/assy.) (Note 3)	≤ 420	≤ ~~475~~ *495*	≤ 443	≤ ~~467~~ *428*	≤ ~~467~~ *469*	≤ ~~474~~ *475*
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, 32, or *32F* with soluble boron credit - see Note~~s~~ 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~~ *Deleted*.

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. These pitches are 0.441 inches and 0.453 inches.

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, DAMAGED FUEL ASSEMBLIES and FUEL DEBRIS is 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*	~~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*	~~Zr~~ *ZR*	~~Zr~~ *ZR*	~~Zr~~ *ZR*
Design Initial U (kg/assy.) (Note 3)	≤ ~~191~~ *192*	≤ ~~191~~ *183*	≤ ~~191~~ *183*	< ~~191~~ *183*	≤ 191	≤ ~~179~~ *180*
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*	~~Zr~~ *ZR*	~~Zr~~ *ZR*	~~Zr~~ *ZR*
Design Initial U (kg/assy.) (Note 3)	≤ ~~179~~ *173*	≤ ~~179~~ *175*	≤ ~~179~~ *175*	≤ ~~179~~ *183*	≤ ~~179~~ *183*	≤ ~~179~~ *157*
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~~ *ZR*	~~Zr~~ *ZR*	SS	SS
Design Initial U (kg/assy.) (Note 3)	≤ 188	≤ 188	≤ ~~188~~ *172*	≤ 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~~ *Deleted.*

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

TABLE DELETED

~~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~~	~~36,700~~
~~≥6~~	~~45,000~~	~~45,000~~	~~43,700~~	~~36,500~~	~~41,600~~	~~39,900~~
~~≥7~~	~~45,900~~	~~46,300~~	~~44,500~~	~~37,500~~	~~42,300~~	~~40,700~~
~~≥8~~	~~48,300~~	~~48,900~~	~~46,900~~	~~39,900~~	~~44,800~~	~~42,900~~
~~≥9~~	~~50,300~~	~~50,700~~	~~48,700~~	~~41,500~~	~~46,600~~	~~44,700~~
~~≥10~~	~~51,600~~	~~52,100~~	~~50,100~~	~~42,900~~	~~48,000~~	~~46,100~~
~~≥11~~	~~53,100~~	~~53,700~~	~~51,500~~	~~44,100~~	~~49,600~~	~~47,200~~
~~≥12~~	~~54,500~~	~~55,100~~	~~52,600~~	~~45,000~~	~~50,800~~	~~48,500~~
~~≥13~~	~~55,600~~	~~56,100~~	~~53,800~~	~~45,700~~	~~51,800~~	~~49,800~~
~~≥14~~	~~56,500~~	~~57,100~~	~~54,900~~	~~46,500~~	~~52,700~~	~~50,700~~
~~≥15~~	~~57,400~~	~~58,000~~	~~55,800~~	~~47,200~~	~~53,900~~	~~51,500~~

~~Notes: 1. Linear interpolation between points is permitted.~~

~~2. Burnup for fuel assemblies with cladding made of materials other than Zircaloy-2 or Zircaloy-4 is limited to 45,000 MWD/MTU or the value in this table, whichever is less.~~

Table 2.1-5

TABLE DELETED

~~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~~	~~393~~
~~≥6~~	~~1123~~	~~1138~~	~~1081~~	~~873~~	~~394~~	~~374~~
~~≥7~~	~~1030~~	~~1043~~	~~991~~	~~805~~	~~363~~	~~345~~
~~≥8~~	~~1020~~	~~1033~~	~~981~~	~~800~~	~~360~~	~~342~~
~~≥9~~	~~1010~~	~~1023~~	~~972~~	~~794~~	~~358~~	~~340~~
~~≥10~~	~~1000~~	~~1012~~	~~962~~	~~789~~	~~355~~	~~337~~
~~≥11~~	~~996~~	~~1008~~	~~958~~	~~785~~	~~353~~	~~336~~
~~≥12~~	~~992~~	~~1004~~	~~954~~	~~782~~	~~352~~	~~334~~
~~≥13~~	~~987~~	~~999~~	~~949~~	~~773~~	~~350~~	~~332~~
~~≥14~~	~~983~~	~~995~~	~~945~~	~~769~~	~~348~~	~~331~~
~~≥15~~	~~979~~	~~991~~	~~941~~	~~766~~	~~347~~	~~329~~

~~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)

TABLE DELETED

~~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~~

~~Notes: 1. Linear interpolation between points is permitted.~~

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

~~3. Burnup for fuel assemblies with cladding made of materials other than Zircaloy-2 or Zircaloy-4 is limited to 45,000 MWD/MTU or the value in this table, whichever is less.~~

Table 2.1-6 (page 2 of 2)

TABLE DELETED

~~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~~

~~Notes: 1. Linear interpolation between points is permitted.~~

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

~~3. Burnup for fuel assemblies with cladding made of materials other than Zircaloy-2 or Zircaloy-4 is limited to 45,000 MWD/MTU or the value in this table, whichever is less.~~

Table 2.1-7 (page 1 of 2)

TABLE DELETED

~~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).~~
~~4. These limits apply to INTACT FUEL ASSEMBLIES, DAMAGED FUEL ASSEMBLIES, and FUEL DEBRIS.~~

Table 2.1-7 (page 2 of 2)

TABLE DELETED

~~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)	~~NEUTRON POISON~~ INSERTS (Note 3) BURNUP (MWD/MTU)	GUIDE TUBE HARDWARE (Note 4) BURNUP (MWD/MTU)	CONTROL COMPONENT (Note 5) BURNUP (MWD/MTU)	APSR BURNUP (MWD/MTU)
≥ 3	≤ ~~20,000~~ *24,635*	NA (Note 6)	NA	NA
≥ 4	≤ ~~25,000~~ *30,000*	≤ 20,000	NA	NA
≥ 5	≤ ~~30,000~~ *36,748*	≤ 25,000	≤ 630,000	≤ 45,000
≥ 6	≤ ~~40,000~~ *44,102*	≤ 30,000	-	≤ 54,500
≥ 7	≤ ~~45,000~~ *52,900*	≤ 40,000	-	≤ 68,000
≥ 8	≤ ~~50,000~~ *60,000*	≤ 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. Includes Burnable Poison Rod Assemblies (BPRAs), ~~and~~ Wet Annular Burnable Absorbers (WABAs), *and vibration suppressor inserts*
4. Includes Thimble Plug Devices (TPDs), water displacement guide tube plugs, and orifice rod assemblies
5. Includes Control Rod Assemblies (CRAs), Control Element Assemblies (CEAs), and Rod Cluster Control Assemblies (RCCAs)
6. NA means not authorized for loading

2.4 Allowable Cask and Fuel Assembly Bumup, Decay Heat, and Cooling Time

2.4.1 Uniform Fuel Loading

Tables 2.4-1 through 2.4-3 provide the maximum allowable decay heat and bumup per assembly as a function of minimum cooling time for uniform loading for each MPC model.

Table 2.4-1
Maximum Allowable Decay Heat and Bumup for the MPC-24/24E/24EF - Uniform Loading

		Maximum Burnup per Assembly (MWD/MTU) (By Array/Class)							
Minimum Cooling Time (yrs)	Maximum Decay Heat per Assembly (kW)	14x14A	14X14B	14X14C	15X15 A/B/C	15X15 D/E/F/H	16X16A	17X17A	17X17 B/C
3	1.666	45,940	41,011	41,372	34,270	32,340	37,239	36,657	33,358
4	1.666	62,330	55,413	55,337	47,673	45,294	51,350	51,491	46,834
5	1.666	74,792	66,288	65,727	57,748	54,871	62,039	62,914	57,084
6	1.666	75,000	74,319	73,354	65,267	62,062	69,975	71,558	64,721
7	1.666	75,000	75,000	75,000	71,011	67,495	75,000	75,000	70,451
8	1.666	75,000	75,000	75,000	75,000	71,738	75,000	75,000	75,000
9	1.666	75,000	75,000	75,000	75,000	75,000	75,000	75,000	75,000

1. Linear interpolation between points is permitted.

2. Decay heat limits include all sources of decay heat (i.e., fuel and NON-FUEL HARDWARE).

2.4.1 *Uniform Fuel Loading (cont'd)*

Table 2.4-2
Maximum Allowable Decay Heat and Burnup for the MPC-32/32F - Uniform Loading

Minimum Cooling Time (yrs)	Maximum Decay Heat per Assembly (kW)	Maximum Burnup per Assembly (MWD/MTU) (By Array/Class)							
		14x14A	14X14B	14X14C	15X15 A/B/C	15X15 D/E/F/H	16X16A	17X17A	17X17 B/C
3	1.250	33,672	29,927	30,083	24,944	23,549	27,135	26,800	24,353
4	1.250	47,316	42,131	41,984	36,302	34,559	39,101	39,203	35,740
5	1.250	57,703	51,203	50,690	44,671	42,553	48,025	48,688	44,241
6	1.250	65,404	57,850	57,023	50,832	48,538	54,499	55,781	50,493
7	1.250	71,165	62,847	61,715	55,538	52,932	59,398	61,079	55,221
8	1.250	75,000	66,759	65,454	59,103	56,331	63,234	65,178	58,832
9	1.250	75,000	69,958	68,488	62,040	59,078	66,313	68,496	61,786
10	1.250	75,000	72,667	71,132	64,514	61,352	68,925	71,371	64,241
11	1.250	75,000	75,000	73,427	66,700	63,360	71,247	73,778	66,418
12	1.250	75,000	75,000	75,000	68,642	65,169	73,279	75,000	68,325
13	1.250	75,000	75,000	75,000	70,491	66,870	75,000	75,000	70,122
14	1.250	75,000	75,000	75,000	72,251	68,409	75,000	75,000	71,862
15	1.250	75,000	75,000	75,000	73,844	69,924	75,000	75,000	73,429
16	1.250	75,000	75,000	75,000	75,000	71,425	75,000	75,000	75,000
17	1.250	75,000	75,000	75,000	75,000	72,793	75,000	75,000	75,000
18	1.250	75,000	75,000	75,000	75,000	74,249	75,000	75,000	75,000
19	1.250	75,000	75,000	75,000	75,000	75,000	75,000	75,000	75,000

1. *Linear interpolation between points is permitted.*
2. *Decay heat limits include all sources of decay heat (i.e., fuel and NON-FUEL HARDWARE).*

2.4.1 *Uniform Fuel Loading (cont'd)*

Table 2.4-3 (page 1 of 2)
Maximum Allowable Decay Heat and Burnup for the MPC-68/68FF - Uniform Loading

Minimum Cooling Time (yrs)	Maximum Decay Heat per Assembly (kW)	Maximum Burnup per Assembly (MWD/MTU) (By Array/Class)				
		7x7B	8x8B	8x8C/D/E	9x9A	9x9B
3	0.588	31,037	32,499	33,688	34,179	35,567
4	0.588	43,297	45,195	46,857	47,477	49,477
5	0.588	52,283	54,440	56,456	57,337	59,835
6	0 588	58,838	61,287	63,556	64,612	67,531
7	0.588	63,857	66,403	68,974	70,000	70,000
8	0.588	67,800	70,000	70,000	70,000	70,000
9	0.588	70,000	70,000	70,000	70,000	70,000

1. *Linear interpolation between points is permitted.*

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

2.4.1 Uniform Fuel Loading (cont'd)

Table 2.4-3 (page 2 of 2)
Maximum Allowable Decay Heat and Burnup for the MPC-68/68FF - Uniform Loading

		Maximum Burnup per Assembly (MWD/MTU) (By Array/Class)				
Minimum Cooling Time (yrs)	Maximum Decay Heat per Assembly (kW)	9x9C/D	9x9E/F	9x9G	10x10A/B	10x10C
3	0 588	35,123	33,299	38,533	32,790	35,614
4	0.588	48,925	46,157	53,602	45,498	49,475
5	0.588	59,133	55,559	65,015	54,784	59,784
6	0.588	66,681	62,479	70,000	61,652	67,423
7	0.588	70,000	67,792	70,000	66,796	70,000
8	0.588	70,000	70,000	70,000	70,000	70,000

1. *Linear interpolation between points is permitted.*

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

2.4.2 Regionalized Fuel Loading

The allowable maximum decay heat and burnup per assembly as a function of mimimum cooling time for regionalized fuel loading shall be calculated as follows (Regions are defined in Figures 2.1-1 through 2.1-4) :

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2.4.2.5 Linear interpolation is permitted between points after the values for fuel assembly decay heat in Region 1 ($q_{Region\ 1}$) and burnup (B) are calculated.

2.4.2.6 Decay heat limits per assembly are applicable to all sources of decay heat (i.e., fuel and NON-FUEL HARDWARE).

2.4.2 *Regionalized Fuel Loading (cont'd)*

Table 2.4-4 summarizes the sources for the generic values for the coefficients that are based on the thermal analysis, the values input by the cask user, and the values that are calculated to determine the allowable maximum decay heat and burnup per assembly for Regions 1 and 2 as a function of cooling time.

Table 2.4-4
Source of Values for Regionalized Storage Computations

VARIABLE	SOURCE
$N_{Region\ 1}$	Provided in Table 2.4-5
$N_{Region\ 2}$	Provided in Table 2.4-5
Q	Provided in Table 2.4-6
A_0	Provided in Table 2.4-6
A_1	Provided in Table 2.4-6
A_2	Provided in Table 2.4-6
D_0	Provided in Tables 2.4-7 and 8
D_1	Provided in Tables 2.4-7 and 8
$q_{Region\ 2}$	Input by user
$q_{Region\ 1}$	Calculated
B	Calculated

Table 2.4-5 provides the values for $N_{Region\ 1}$, $N_{Region\ 2}$, and $q_{Region\ 2}$ for the various MPC models.

Table 2.4-5
Regionalized Storage Non Cooling Time-Dependent Inputs

MPC MODEL	$N_{Region\ 1}$ (Maximum)	$N_{Region\ 2}$ (Maximum)	$q_{Region\ 2}$ (Minimum) (kW)	$q_{Region\ 2}$ (Maximum) (kW)
24/24E/24EF	4	20	0.900	1.666
32/32F	12	20	0.600	1.250
68/68FF	32	36	0.275	0.588

2.4.2 Regionalized Fuel Loading (cont'd)

Table 2.4-6 provides the values for Q, A_0, A_1, and A_2 for the various MPC models.

Table 2.4-6
Additional Regionalized Storage Non-Cooling Time-Dependent Inputs

MPC Model	Q (kW)	A_0	$A_1 \times 10^3$	$A_2 \times 10^4$
24/24E/24EF	40	0.48861	9.8125	1.6594
32/32F	40	0.77891	5.8735	1.1880
68/68FF	40	0.83760	-3.9413	5.4811

2.4.2 Regionalized Fuel Loading (cont'd)

Tables 2.4-7 and 2.4-8 provide the cooling time-dependent values used to calculate the maximum allowable fuel assembly decay heat and burnups for regionalized fuel loading, using the preceding equations.

Table 2.4-7 (page 1 of 2)
PWR Fuel Cooling Time-Dependent Inputs

	Array/Class 14x14A		Array/Class 14X14B		Array/Class 14X14C		Array/Class 15X15A/B/C	
Minimum Cooling Time (yrs)	D_0	D_1	D_0	D_1	D_0	D_1	D_0	D_1
3	-3,452	29,674	-3,295	26,580	-3,863	27,146	-2,932	22,311
4	2,313	36,003	2,318	31,851	1,958	32,021	1,651	27,707
5	6,476	40,982	5,986	36,174	5,615	36,060	5,473	31,359
6	8,932	45,178	8,480	39,496	8,068	39,164	7,561	34,617
7	10,277	48,711	9,744	42,483	9,435	41,824	9,157	37,1[illegible]
8	11,126	51,701	10,619	44,912	10,174	44,224	9,583	39,616
9	11,727	54,243	11,153	47,044	10,697	46,233	10,167	41,499
10	12,211	56,418	11,641	48,821	10,745	48,310	10,616	43,119
11	12,131	58,857	11,596	50,830	11,105	49,858	10,577	44,899
12	12,501	60,571	11,908	52,306	10,964	51,688	10,901	46,193
13	12,252	62,835	11,781	54,081	10,809	53,443	10,756	47,778
14	12,762	64,097	11,672	55,734	11,126	54,610	10,559	49,354
15	12,591	66,102	12,049	56,822	10,945	56,252	10,864	50,384
16	12,794	67,586	11,853	58,482	11,199	57,397	10,739	51,781
17	12,668	69,452	11,607	60,201	11,122	58,856	10,524	53,262
18	13,122	70,568	11,946	61,239	10,970	60,382	10,822	54,203
19	12,924	72,471	11,736	62,865	11,129	61,592	10,660	55,587
20	12,802	74,333	12,182	63,735	10,884	63,240	10,428	57,098

2.4.2 *Regionalized Fuel Loading (cont'd)*

Table 2.4-7 (page 2 of 2)
PWR Fuel Cooling Time-Dependent Inputs

	Array/Class 15x15D/E/F/H		Array/Class 16x16A		Array/Class 17x17A		Array/Class 17x17B/C	
Minimum Cooling Time (yrs)	D_0	D_1	D_0	D_1	D_0	D_1	D_0	D_1
3	-2,723	21,032	-3,003	24,124	-2,597	23,533	-2,564	21,544
4	1,847	26,151	1,834	29,832	2,080	29,706	1,690	27,222
5	5,627	29,541	6,017	33,607	6,043	34,116	5,741	30,800
6	7,997	32,433	8,111	37,111	8,489	37,834	7,842	34,121
7	9,276	34,925	9,833	39,652	9,748	41,065	9,569	36,522
8	10,146	36,948	10,129	42,484	10,551	43,702	9,889	39,155
9	10,666	38,730	10,761	44,442	11,083	45,931	10,349	41,150
10	10,690	40,530	11,129	46,237	11,019	48,282	10,853	42,711
11	11,123	41,790	11,069	48,143	11,423	49,884	10,751	44,534
12	11,044	43,300	11,472	49,446	11,245	51,855	11,113	45,770
13	10,855	44,812	11,349	51,109	11,564	53,193	10,939	47,347
14	11,148	45,809	11,119	52,806	11,388	54,940	10,779	48,867
15	11,011	47,131	11,439	53,890	11,116	56,717	11,063	49,893
16	10,833	48,474	11,246	55,447	11,441	57,828	10,924	51,276
17	11,167	49,301	11,082	56,956	11,238	59,488	10,788	52,643
18	10,957	50,634	11,342	58,006	11,060	61,099	10,949	53,709
19	10,804	51,915	11,183	59,494	11,353	62,178	10,819	55,048
20	10,663	53,171	11,035	60,989	11,043	63,938	10,626	56,469

2.4.2 *Regionalized Fuel Loading (cont'd)*

Table 2.4-8 (page 1 of 2)
BWR Fuel Cooling Time-Dependent Inputs

	Array/Class 7x7B		Array/Class 8x8B		Array/Class 8x8C/D/E		Array/Class 9x9A		Array/Class 9x9B	
Minimum Cooling Time (yrs)	D_0	D_1	D_0	D_1	D_0	D_1	D_0	D_1	D_0	D_1
3	-3,108	58,070	-3,485	61,198	-3,271	62,856	-3,250	63,655	-3,718	66,812
4	725	72,403	518	75,982	668	78,553	830	79,333	719	82,923
5	4,237	81,711	4,187	85,465	4,593	88,204	4,539	89,794	4,486	94,131
6	6,483	89,040	6,375	93,374	6,825	96,482	6,737	98,427	6,718	103,424
7	7,565	95,735	7,920	99,461	8,114	103,505	7,989	105,694	7,957	111,296
8	8,311	101,172	8,220	106,093	8,637	109,994	8,625	112,217	8,520	118,470
9	8,700	106,055	8,614	111,207	9,135	115,182	8,958	117,846	8,847	124,675
10	8,775	110,760	9,050	115,329	9,037	120,884	9,405	122,419	8,897	130,461
11	8,984	114,609	8,909	120,252	9,333	124,937	9,277	127,541	9,147	135,087
12	8,752	119,084	9,058	123,931	9,348	129,214	9,403	131,835	9,243	139,505
13	8,874	122,457	8,877	128,204	9,439	132,999	9,519	135,466	9,319	143,799
14	8,979	125,614	9,102	131,306	9,413	136,918	9,635	139,196	9,872	146,416
15	8,874	129,275	8,920	135,292	9,666	139,956	9,487	143,368	9,920	150,231
16	8,896	132,463	8,880	138,838	9,442	144,090	9,373	147,315	9,516	155,447
17	8,967	135,457	9,099	141,468	9,321	147,823	9,711	150,043	9,755	158,439
18	8,673	139,379	8,981	145,106	9,305	151,561	9,583	154,061	9,582	162,837
19	8,856	142,079	8,688	149,308	9,511	154,395	10,041	156,191	10,329	164,705
20	8,789	145,330	9,189	151,243	9,328	158,504	9,735	160,688	9,888	169,600

2.4.2 *Regionalized Fuel Loading (cont'd)*

Table 2.4-8 (page 2of 2)
BWR Fuel Cooling Time-Dependent Inputs

	Array/Class 9x9C/D		Array/Class 9x9E/F		Array/Class 9x9G		Array/Class 10x10A/B		Array/Class 10x10C	
Minimum Cooling Time (yrs)	D_0	D_1	D_0	D_1	D_0	D_1	D_0	D_1	D_0	D_1
3	-3,421	65,552	-3,036	61,795	-4,038	72,401	-3,270	61,327	-3,507	66,533
4	773	81,892	977	76,838	864	89,692	774	76,062	933	82,556
5	4,396	93,091	4,923	86,117	4,574	102,791	4,693	85,190	4,809	93,495
6	6,634	102,121	7,198	94,016	6,900	113,239	6,623	93,587	7,062	102,655
7	8,052	109,433	8,342	101,107	8,059	122,558	8,185	99,680	8,247	110,658
8	8,693	116,189	9,041	106,990	8,708	130,518	8,767	105,770	8,833	117,680
9	8,868	122,554	9,406	112,370	9,152	137,270	8,984	111,290	9,205	123,746
10	9,129	127,725	9,469	117,473	9,828	142,025	9,297	115,715	9,302	129.505
11	9,318	132,394	9,767	121,401	10,537	145,909	9,158	120,625	9,730	133,637
12	9,305	137,044	9,572	126,067	10,263	151,870	9,331	124,390	9,570	138,712
13	9,723	140,106	9,903	129,164	10,502	156,166	9,390	128,008	9,938	142,149
14	9,465	144,861	9,741	133,237	10,448	160,869	9,404	131,729	10,379	145,109
15	9,940	147,584	9,602	137,141	10,975	163,765	9,383	135,289	10,281	149,291
16	9,671	152,067	9,981	139,608	10,640	169,223	9,161	139,248	9,897	154,252
17	9,759	155,614	9,658	143,909	10,914	172,628	9,434	141,834	10,273	157,172
18	9,818	159,043	10,169	145,848	11,131	175,965	9,150	146,012	10,259	160,778
19	9,783	162,824	10,061	149,594	12,795	174,974	9,501	148,390	10,818	163,033
20	10,646	163,852	9,997	153,051	12,250	180,885	9,552	151,491	10,149	168,851

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 *and MPC-32F*

1. Fuel cell pitch· ≥ 9.158 inches

2. ^{10}B loading in the ~~Boral~~ neutron absorbers: ≥ 0 0372 g/cm^2

3 2 6 Fuel spacers shall be sized to ensure that the active fuel region of intact fuel assemblies remains within the ~~Boral~~ neutron poison region of the MPC basket with water in the MPC

DESIGN FEATURES

3.3 Codes and Standards

The American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code), 1995 Edition with Addenda through 1997, is the governing Code for the HI-STORM 100 System, as clarified in Specification 3.3.1 below, *except for Code Sections V and IX. The latest effective editions of ASME Code Sections V and IX, including addenda, shall be used for activities governed by those sections*. American Concrete Institute (ACI) 349-85 is the governing Code for plain concrete as clarified in Appendix 1.D of the Final Safety Analysis Report for the HI-STORM 100 Cask System.

3 3.1 ~~Exceptions~~*Alternatives* to Codes, Standards, and Criteria

Table 3-1 lists approved ~~exceptions~~*alternatives* to the ASME Code for the design of the HI-STORM 100 Cask System.

3.3.2 Construction/Fabrication ~~Exceptions~~*Alternatives* to Codes, Standards, and Criteria

Proposed alternatives to the ASME Code, Section III, 1995 Edition with Addenda through 1997 including exceptions allowed by Specification 3.3.1 may be used *on a case-specific basis* when authorized by the Director of the Office of Nuclear Material Safety and Safeguards or designee. The request for such alternative should demonstrate that:

1. The proposed alternatives would provide an acceptable level of quality and safety, or

2 Compliance with the specified requirements of the ASME Code, Section III, 1995 Edition with Addenda through 1997, would result in hardship or unusual difficulty without a compensating increase in the level of quality and safety.

Requests for ~~exceptions~~*alternatives* shall be submitted in accordance with 10 CFR 72.4

(continued)

DESIGN FEATURES

Table 3-1 (page 1 of ~~5~~9)
LIST OF ASME CODE ~~EXCEPTIONS~~*ALTERNATIVES* FOR HI-STORM 100 CASK SYSTEM

Component	Reference ASME Code Section/Article	Code Requirement	~~Exception~~*Alternative*, Justification & Compensatory Measures
MPC, MPC basket assembly, HI-STORM OVERPACK steel structure, and HI-TRAC TRANSFER CASK steel structure	*Subsection NCA*	*General Requirements. Requires preparation of a Design Specification, Design Report, Overpressure Protection Report, Certification of Construction Report, Data Report, and other administrative controls for an ASME Code stamped vessel.*	*Because the MPC, OVERPACK, and TRANSFER CASK are not ASME Code stamped vessels, none of the specifications, reports, certificates, or other general requirements specified by NCA are required. In lieu of a Design Specification and Design Report, the HI-STORM FSAR includes the design criteria, service conditions, and load combinations for the design and operation of the HI-STORM 100 System as well as the results of the stress analyses to demonstrate that applicable Code stress limits are met. Additionally, the fabricator is not required to have an ASME-certified QA program. All important-to-safety activities are governed by the NRC-approved Holtec QA program.* *Because the cask components are not certified to the Code, the terms "Certificate Holder" and "Inspector" are not germane to the manufacturing of NRC-certified cask components. To eliminate ambiguity, the responsibilities assigned to the Certificate Holder in the various articles of Subsections NB, NG, and NF of the Code, as applicable, shall be interpreted to apply to the NRC Certificate of Compliance (CoC) holder (and by extension, to the component fabricator) if the requirement must be fulfilled The Code term "Inspector" means the QA/QC personnel of the CoC holder and its vendors assigned to oversee and inspect the manufacturing process.*
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.

Table 3-1 (page 2 of 59)
LIST OF ASME CODE ~~EXCEPTIONS~~*ALTERNATIVES* FOR HI-STORM 100 CASK SYSTEM

Component	Reference ASME Code Section/Article	Code Requirement	~~Exception~~*Alternative*, Justification & Compensatory Measures
MPC basket supports and lift lugs	*NB-1130*	*NB-1132.2(d) requires that the first connecting weld of a nonpressure-retaining structural attachment to a component shall be considered part of the component unless the weld is more than 2t from the pressure-retaining portion of the component, where t is the nominal thickness of the pressure-retaining material.* *NB-1132.2(e) requires that the first connecting weld of a welded nonstructural attachment to a component shall conform to NB-4430 if the connecting weld is within 2t from the pressure-retaining portion of the component*	*The MPC basket supports (nonpressure-retaining structural attachment)and lift lugs (nonstructural attachments used exclusively for lifting an empty MPC) are welded to the inside of the pressure-retaining MPC shell, but are not designed in accordance with Subsection NB. The basket supports and associated attachment welds are designed to satisfy the stress limits of Subsection NG and the lift lugs and associated attachment welds are designed to satisfy the stress limits of Subsection NF, as a minimum. These attachments and their welds are shown by analysis to meet the respective stress limits for their service conditions. Likewise, non-structural items, such as shield plugs, spacers, etc. if used, can be attached to pressure-retaining parts in the same manner*

Table 3-1 (page 3 of ~~5~~9)
LIST OF ASME CODE ~~EXCEPTIONS~~*ALTERNATIVES* FOR HI-STORM 100 CASK SYSTEM

Component	Reference ASME Code Section/Article	Code Requirement	~~Exception~~*Alternative*, Justification & Compensatory Measures
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, MPC basket assembly, HI-STORM OVERPACK and HI-TRAC TRANSFER CASK	*NB-3100 NG-3100 NF-3100*	*Provides requirements for determining design loading conditions, such as pressure, temperature, and mechanical loads.*	*These requirements are not applicable. The HI-STORM FSAR, serving as the Design Specification, establishes the service conditions and load combinations for the storage system.*
MPC	*NB-3350*	*NB-3352.3 requires, for Category C joints, that the minimum dimensions of the welds and throat thickness shall be as shown in Figure NB-4243-1.*	*Due to MPC basket-to-shell interface requirements, the MPC shell-to-baseplate weld joint design (designated Category C) does not include a reinforcing fillet weld or a bevel in the MPC baseplate, which makes it different than any of the representative configurations depicted in Figure NB-4243-1. The transverse thickness of this weld is equal to the thickness of the adjoining shell (1/2 inch). The weld is designed as a full penetration weld that receives VT and RT or UT, as well as final surface PT examinations. Because the MPC shell design thickness is considerably larger than the minimum thickness required by the Code, a reinforcing fillet weld that would intrude into the MPC cavity space is not included. Not including this fillet weld provides for a higher quality radiographic examination of the full penetration weld.* *From the standpoint of stress analysis, the fillet weld serves to reduce the local bending stress (secondary stress) produced by the gross structural discontinuity defined by the flat plate/shell junction. In the MPC design, the shell and baseplate thicknesses are well beyond that required to meet their respective membrane stress intensity limits.*

Table 3-1 (page 4 of ~~5~~9)
LIST OF ASME CODE ~~EXCEPTIONS~~*ALTERNATIVES* FOR HI-STORM 100 CASK SYSTEM

Component	Reference ASME Code Section/Article	Code Requirement	~~Exception~~*Alternative*, Justification & Compensatory Measures
MPC, MPC Basket Assembly, HI-STORM OVERPACK steel structure, and HI-TRAC TRANSFER CASK steel structure	*NB-4120* *NG-4120* *NF-4120*	*NB-4121.2, NG-4121.2, and NF-4121.2 provide requirements for repetition of tensile or impact tests for material subjected to heat treatment during fabrication or installation.*	*In-shop operations of short duration that apply heat to a component, such as plasma cutting of plate stock, welding, machining, coating, and pouring of lead are not, unless explicitly stated by the Code, defined as heat treatment operations* *For the steel parts in the HI-STORM 100 System components, the duration for which a part exceeds the off-normal temperature limit defined in Chapter 2 of the FSAR shall be limited to 24 hours in a particular manufacturing process (such as the HI-TRAC lead pouring process).*
MPC, MPC basket assembly, HI-STORM OVERPACK steel structure, and HI-TRAC TRANSFER CASK steel structure	*NB-4220* *NF-4220*	*Requires certain forming tolerances to be met for cylindrical, conical, or spherical shells of a vessel*	*The cylindricity measurements on the rolled shells are not specifically recorded in the shop travelers, as would be the case for a Code-stamped pressure vessel. Rather, the requirements on inter-component clearances (such as the MPC-to-TRANSFER CASK) are guaranteed through fixture-controlled manufacturing. The fabrication specification and shop procedures ensure that all dimensional design objectives, including inter-component annular clearances are satisfied. The dimensions required to be met in fabrication are chosen to meet the functional requirements of the dry storage components Thus, although the post-forming Code cylindricity requirements are not evaluated for compliance directly, they are indirectly satisfied (actually exceeded) in the final manufactured components.*
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.

Table 3-1 (page 5 of ~~5~~9)
LIST OF ASME CODE ~~EXCEPTIONS~~*ALTERNATIVES* FOR HI-STORM 100 CASK SYSTEM

Component	Reference ASME Code Section/Article	Code Requirement	~~Exception~~*Alternative*, Justification & Compensatory Measures
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.
MPC Enclosure Vessel and Lid	NB-6111	All completed pressure retaining systems shall be pressure tested	The MPC enclosure vessel is seal welded in the field following fuel assembly loading The MPC enclosure vessel shall then be hydrostatically tested as defined in Chapter 9. Accessibility for leakage inspections preclude a Code compliant hydrostatic test All MPC enclosure vessel welds (except closure ring and vent/drain cover plate) are inspected by volumetric examination, except the MPC lid-to-shell weld shall be verified by volumetric or multi-layer PT examination. If PT alone is used, at a minimum, it must include the root and final layers and each approximately 3/8 inch of weld depth. For either UT or PT, the maximum undetectable flaw size must be demonstrated to be less than the critical flaw size. The critical flaw size must be determined in accordance with ASME Section XI methods The critical flaw size shall not cause the primary stress limits of NB-3000 to be exceeded. The inspection ~~process~~ *results*, including *relevant* findings (indications), shall be made a permanent part of the user's records by video, photographic, or other means which provide an equivalent retrievable record of weld integrity. The video or photographic records should be taken during the final interpretation period described in ASME Section V, Article 6, T-676. The vent/drain cover plate weld is confirmed by leakage testing and liquid penetrant examination and the closure ring weld is confirmed by liquid penetrant examination. The inspection of the weld must be performed by qualified personnel and shall meet the acceptance requirements of ASME Code Section III, NB-5350 for PT or NB-5332 for UT.

Table 3-1 (page 6 of ~~5~~9)
LIST OF ASME CODE ~~EXCEPTIONS~~*ALTERNATIVES* FOR HI-STORM 100 CASK SYSTEM

Component	Reference ASME Code Section/Article	Code Requirement	~~Exception~~*Alternative*, Justification & Compensatory Measures
MPC Enclosure Vessel	NB-7000	Vessels are required to have overpressure protection	No overpressure protection is provided. The function of the MPC enclosure vessel is to contain the radioactive contents under normal, off-normal, and accident conditions. The MPC vessel is designed to withstand maximum internal pressure considering 100% fuel rod failure and maximum accident temperatures.
MPC Enclosure Vessel	NB-8000	States requirements for nameplates, stamping and reports per NCA-8000.	The HI-STORM100 System is to be marked and identified in accordance with 10CFR71 and 10CFR72 requirements. Code stamping is not required QA data package to be in accordance with Holtec approved QA program.
MPC Basket Assembly	NG-2000	Requires materials to be supplied by ASME-approved material supplier.	Materials will be supplied by Holtec-approved supplier with CMTRs in accordance with NG-2000 requirements.

Table 3-1 (page 7 of ~~59~~) LIST OF ASME CODE ~~EXCEPTIONS~~ALTERNATIVES FOR HI-STORM 100 CASK SYSTEM			
Component	**Reference ASME Code Section/Article**	**Code Requirement**	**~~Exception~~*Alternative*, Justification & Compensatory Measures**
MPC basket assembly	*NG-4420*	*NG-4427(a) requires a fillet weld in any single continuous weld may be less than the specified fillet weld dimension by not more than 1/16 inch, provided that the total undersize portion of the weld does not exceed 10 percent of the length of the weld. Individual undersize weld portions shall not exceed 2 inches in length.*	*Modify the Code requirement (intended for core support structures) with the following text prepared to accord with the geometry and stress analysis imperatives for the fuel basket: For the longitudinal MPC basket fillet welds, the following criteria apply. 1) The specified fillet weld throat dimension must be maintained over at least 92 percent of the total weld length. All regions of undersized weld must be less than 3 inches long and separated from each other by at least 9 inches. 2) Areas of undercuts and porosity beyond that allowed by the applicable ASME Code shall not exceed 1/2 inch in weld length. The total length of undercut and porosity over any 1-foot length shall not exceed 2 inches. 3) The total weld length in which items (1) and (2) apply shall not exceed a total of 10 percent of the overall weld length. The limited access of the MPC basket panel longitudinal fillet welds makes it difficult to perform effective repairs of these welds and creates the potential for causing additional damage to the basket assembly (e g., to the neutron absorber and its sheathing) if repairs are attempted. The acceptance criteria provided in the foregoing have been established to comport with the objectives of the basket design and preserve the margins demonstrated in the supporting stress analysis. From the structural standpoint, the weld acceptance criteria are established to ensure that any departure from the ideal, continuous fillet weld seam would not alter the primary bending stresses on which the design of the fuel baskets is predicated. Stated differently, the permitted weld discontinuities are limited in size to ensure that they remain classifiable as local stress elevators ("peak stress", F, in the ASME Code for which specific stress intensity limits do not apply).*
MPC Basket Assembly	NG-8000	States requirements for nameplates, stamping and reports per NCA-8000	The HI-STORM100 System is to be marked and identified in accordance with 10CFR71 and 10CFR72 requirements. Code stamping is not required. The MPC basket data package to be in accordance with Holtec approved QA program.
OVERPACK Steel Structure	NF-2000	Requires materials to be supplied by ASME-approved material supplier.	Materials will be supplied by Holtec-approved supplier with CMTRs in accordance with NF-2000 requirements

Table 3-1 (page 8 of ~~5~~9) LIST OF ASME CODE ~~EXCEPTIONS~~*ALTERNATIVES* FOR HI-STORM 100 CASK SYSTEM			
Component	**Reference ASME Code Section/Article**	**Code Requirement**	**~~Exception~~*Alternative*, Justification & Compensatory Measures**
TRANSFER CASK Steel Structure	NF-2000	Requires materials to be supplied by ASME-approved material supplier.	Materials will be supplied by Holtec-approved supplier with CMTRs in accordance with NF-2000 requirements.
OVERPACK Baseplate and Lid Top Plate	NF-4441	Requires special examinations or requirements for welds where a primary member of thickness 1 inch or greater is loaded to transmit loads in the through thickness direction.	The ~~large~~ margins of safety in these welds under loads experienced during lifting operations or accident conditions are quite large. The OVERPACK baseplate welds to the inner shell, pedestal shell, and radial plates are only loaded during lifting conditions and have ~~a minimum~~ *large* safety factors ~~of > 12~~ during lifting Likewise, ~~T~~the top lid plate to lid shell weld has a *large structural margin under the inertia loads imposed during a non-mechanistic tipover event.* ~~safety factor > 6 under a deceleration of 45 g's.~~
OVERPACK Steel Structure	NF-3256 *NF-3266*	Provides requirements for welded joints.	Welds for which no structural credit is taken are identified as "Non-NF" welds in the design drawings by an "*". These non-structural welds are specified in accordance with the pre-qualified welds of AWS D1.1. These welds shall be made by welders and weld procedures qualified in accordance with AWS D1.1 or ASME Section IX. *Welds for which structural credit is taken in the safety analyses shall meet the stress limits for NF-3256.2, but are not required to meet the joint configuration requirements specified in these Code articles. The geometry of the joint designs in the cask structures are based on the fabricability and accessibility of the joint, not generally contemplated by this Code section governing supports.*

Table 3-1 (page 9 of 59)
LIST OF ASME CODE ~~EXCEPTIONS~~*ALTERNATIVES* FOR HI-STORM 100 CASK SYSTEM

Component	Reference ASME Code Section/Article	Code Requirement	~~Exception~~*Alternative*, Justification & Compensatory Measures
HI-STORM OVERPACK and HI-TRAC TRANSFER CASK	*NF-3320* *NF-4720*	*NF-3324.6 and NF-4720 provide requirements for bolting*	*These Code requirements are applicable to linear structures wherein bolted joints carry axial, shear, as well as rotational (torsional) loads. The OVERPACK and TRANSFER CASK bolted connections in the structural load path are qualified by design based on the design loadings defined in the FSAR. Bolted joints in these components see no shear or torsional loads under normal storage conditions. Larger clearances between bolts and holes may be necessary to ensure shear interfaces located elsewhere in the structure engage prior to the bolts experiencing shear loadings (which occur only during side impact scenarios).* *Bolted joints that are subject to shear loads in accident conditions are qualified by appropriate stress analysis. Larger bolt-to-hole clearances help ensure more efficient operations in making these bolted connections, thereby minimizing time spent by operations personnel in a radiation area. Additionally, larger bolt-to-hole clearances allow interchangeability of the lids from one particular fabricated cask to another.*

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)

b. For those ISFSI sites with design basis seismic acceleration values higher than those allowed for free-standing casks, the HI-STORM 100 System shall be anchored to the ISFSI pad. The site seismic characteristics and the anchorage system shall meet the following requirements:

i. The site acceleration response spectra at the top of the ISFSI pad shall have ZPAs that meet the following inequalities:

$$G_H \leq 2.12$$

AND

$$G_V \leq 1.5$$

Where:

G_H is the vectorial sum of the two horizontal ZPAs at a three-dimensional seismic site (or the horizontal ZPA at a two-dimensional site) and G_V is the vertical ZPA.

ii. Each HI-STORM 100 dry storage cask shall be anchored with twenty-eight (28), 2-inch diameter studs and compatible nuts of material suitable for the expected ISFSI environment. The studs shall meet the following requirements:

Yield Strength at Ambient Temperature: ≥ 80 ksi

Ultimate Strength at Ambient Temperature: ≥ 125 ksi

Initial Tensile Pre-Stress: ≥ 55 ksi AND ≤ 65 ksi

NOTE: The above anchorage specifications are required for the seismic spectra defined in item 3.4.3.b.i. Users may use fewer studs or those of different diameter to account for site-specific seismic spectra less severe than those specified above. The embedment design shall comply with Appendix B of ACI-349-97. A later edition of this Code may be used, provided a written reconciliation is performed.

iii. Embedment Concrete Compressive Strength: ≥ 4,000 psi at 28 days

(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. a. For free-standing casks, the ISFSI pad shall be verified by analysis to limit cask deceleration during design basis drop and non-mechanistic tip-over events 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 100 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.

   b. For anchored casks, the ISFSI pad shall be designed to meet the embedment requirements of the anchorage design. A cask tip-over event for an anchored cask is not credible. The ISFSI pad shall be verified by analysis to limit cask deceleration during a design basis drop event to ≤ 45 g's at the top of the MPC fuel basket, except as provided for in this paragraph below. Analyses shall be performed using methodologies consistent with those described in the HI-STORM 100 FSAR. A lift height above the ISFSI pad is not required to be established if the cask is lifted with a device design in accordance with ANSI N14.6 and having redundant drop protection features.

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.

(continued)

DESIGN FEATURES

3.4 Site-Specific Parameters and Analyses (continued)

8. LOADING OPERATIONS, TRANSPORT OPERATIONS, and UNLOADING OPERATIONS shall only be conducted with working area ambient temperatures ≥ 0° F.

9. For those users whose site-specific design basis includes an event or events (e.g., flood) that result in the blockage of any OVERPACK inlet or outlet air ducts for an extended period of time (i.e, longer than the total Completion Time of LCO 3.1.2), an analysis or evaluation may be performed to demonstrate adequate heat removal is available for the duration of the event. Adequate heat removal is defined as fuel cladding temperatures remaining below the short term temperature limit. If the analysis or evaluation is not performed, or if fuel cladding temperature limits are unable to be demonstrated by analysis or evaluation to remain below the short term temperature limit for the duration of the event, provisions shall be established to provide alternate means of cooling to accomplish this objective.

10. *For on-site TRANSPORT OPERATIONS with a loaded TRANSFER CASK, the requirements in Table 3-2a apply.*

Table 3-2a
TRANSFER CASK Operating Limits (Note 1)

Fuel Burnup (MWD/MTU)	**MPC Heat Load (kW)**	**Maximum Time with TRANSFER CASK in Vertical Orientation (Notes 2 and 3) (hrs)**	**Maximum Time with TRANSFER CASK in Horizontal Orientation (Notes 2 and 3) (hrs)**
Any authorized burnup	≤ 20	Unlimited	Unlimited
All Assemblies ≤ 45,000	> 20 and ≤ 30	Unlimited	60 OR unlimited, if fuel cladding hoop stress is shown by analysis to be ≤ 90 MPa
All Assemblies ≤ 45,000	> 30 and ≤ 40	60 OR unlimited, if fuel cladding hoop stress is shown by analysis to be ≤ 90 MPa	60 OR unlimited, if fuel cladding hoop stress is shown by analysis to be ≤ 90 MPa
One or more assemblies > 45,000	> 20 and ≤ 30	Unlimited	60
One or more assemblies > 45,000	> 30 and ≤ 40	60	60

Notes:

1. If the limits in two or more rows apply, the user may choose one set of limits to implement.
2. Time is measured from completion of the installation of the TRANSFER CASK top lid to when MPC transfer begins (for vertical orientation) and from the completion of TRANSFER CASK downending to the completion of upending (for horizontal orientation).
3. Fuel cladding hoop stress calculations may be performed using "best estimate" inputs.

(continued)

DESIGN FEATURES

3.5 Cask Transfer Facility (CTF)

3.5.1 TRANSFER CASK and MPC Lifters

Lifting of a loaded TRANSFER CASK and MPC ~~outside~~ *using devices that are not integral to* ~~of~~ structures governed by 10 CFR Part 50 shall be performed with a CTF that is designed, operated, fabricated, tested, inspected, and maintained in accordance with the guidelines of NUREG-0612, "Control of Heavy Loads at Nuclear Power Plants" and the below clarifications. The CTF Structure requirements below do not apply to heavy loads bounded by the regulations of 10 CFR Part 50.

3.5.2 CTF Structure Requirements

3.5.2.1 Cask Transfer Station and Stationary Lifting Devices

1. The metal weldment structure of the CTF structure shall be designed to comply with the stress limits of ASME Section III, Subsection NF, Class 3 for linear structures. The applicable loads, load combinations, and associated service condition definitions are provided in Table 3-3. All compression loaded members shall satisfy the buckling criteria of ASME Section III, Subsection NF.

2. If a portion of the CTF structure is constructed of reinforced concrete, then the factored load combinations set forth in ACI-318 (89) for the loads defined in Table 3-3 shall apply.

3. The TRANSFER CASK and MPC lifting device used with the CTF shall be designed, fabricated, operated, tested, inspected and maintained in accordance with NUREG-0612, Section 5.1.

4. The CTF shall be designed, constructed, and evaluated to ensure that if the MPC is dropped during inter-cask transfer operations, its confinement boundary would not be breached. This requirements applies to CTFs with either stationary or mobile lifting devices.

(continued)

DESIGN FEATURES

3.5.2.2 Mobile Lift Devices

If a mobile lifting device is used as the lifting device, in lieu of a stationary lifting device, is shall meet the guidelines of NUREG- 0612, Section 5.1, with the following clarifications.

1. Mobile lifting devices shall have a minimum safety factor of two over the allowable load table for the lifting device in accordance with the guidance of NUREG-0612, Section 5.1.6(1)(a) and shall be capable of stopping and holding the load during a Design Basis Earthquake (DBE) event.

2. Mobile lifting devices shall conform to meet the requirements of ANSI B30.5, "Mobile and Locomotive Cranes," in lieu of the requirements of ANSI B30.2, "Overhead and Gantry Cranes."

3 Mobile cranes are not required to meet the requirements of NUREG-0612, Section 5.1.6(2) for new cranes.

4. Horizontal movements of the TRANSFER CASK and MPC using a mobile crane are prohibited.

(continued)

DESIGN FEATURES

Table 3-3

Load Combinations and Service Condition Definitions for the CTF Structure (Note 1)

Load Combination	ASME III Service Condition for Definition of Allowable Stress	Comment
D* D + S	Level A	All primary load bearing members must satisfy Level A stress limits
D + M + W' (Note 2) D + F D + E D + Y	Level D	Factor of safety against overturning shall be ≥ 1.1

D = Dead load
D* = Apparent dead load
S = Snow and ice load for the CTF site
M = Tornado missile load for the CTF site
W' = Tornado wind load for the CTF site
F = Flood load for the CTF site
E = Seismic load for the CTF site
Y = Tsunami load for the CTF site

Notes:

1. The reinforced concrete portion of the CTF structure shall also meet the factored combinations of loads set forth in ACI-318(89).

2. Tornado missile load may be reduced or eliminated based on a PRA for the CTF site.

DESIGN FEATURES

3.6 Forced Helium Dehydration System

3.6.1 System Description

Use of the Forced Helium Dehydration (FHD) system, (a closed-loop system) is an alternative to vacuum drying the MPC for moderate burnup fuel (≤ 45,000 MWD/MTU) and mandatory for drying MPCs containing one or more high burnup fuel assemblies. The FHD system shall be designed for normal operation (i.e., excluding startup and shutdown ramps) in accordance with the criteria in Section 3.6.2.

3.6.2 Design Criteria

3.6.2.1 The temperature of the helium gas in the MPC shall be at least 15°F higher than the saturation temperature at coincident pressure.

3.6.2.2 The pressure in the MPC cavity space shall be ≤ 60.3 psig (75 psia).

3.6.2.3 The hourly recirculation rate of helium shall be ≥ 10 times the nominal helium mass backfilled into the MPC for fuel storage operations.

3.6.2.4 The partial pressure of the water vapor in the MPC cavity will not exceed 3 torr if the helium temperature at the demoisturer outlet is ≤ 21°F for a period of 30 minutes.

3.6.2.5 The condensing module shall be designed to de-vaporize the recirculating helium gas to a dew point ≤ 120°F.

3.6.2.6 The demoisturizing module shall be configured to be introduced into its helium conditioning function after the condensing module has been operated for the required length of time to assure that the bulk moisture vaporization in the MPC (defined as Phase 1 in FSAR Appendix 2.B) has been completed.

3.6.2.7 The helium circulator shall be sized to effect the minimum flow rate of circulation required by these design criteria.

3 6.2.8 The pre-heater module shall be engineered to ensure that the temperature of the helium gas in the MPC meets these design criteria

(continued)

DESIGN FEATURES

3.6 Forced Helium Dehydration System (continued)

3.6.3 Fuel Cladding Temperature

A steady-state thermal analysis of the MPC under the forced helium flow scenario shall be performed using the methodology described in HI-STORM 100 FSAR Subsections 4.4.1.1.1 through 4.4.1.1.4, with due recognition of the forced convection process during FHD system operation. This analysis shall demonstrate that the peak temperature of the fuel cladding under the most adverse condition of FHD system operation, is below the peak cladding temperature limit for normal conditions of storage for the applicable fuel type (PWR or BWR) and cooling time at the start of dry storage.