HOLTEC INTERNATIONAL

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May 18, 2007

U.S. Nuclear Regulatory Commission
ATTN: Document Control Desk
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Reference:
1. USNRC Docket No. 72-1014 (HI-STORM 100), TAC L23996
2. Holtec Project 5014
3. Letter from C. Regan (NRC) to S. Anton (Holtec), dated 30 March 2007
4. Holtec Letter 5014600, dated 23 June 2006
5. Holtec Letter 5014613, dated 19 January 2007

Subject: Response to RAI-2 on License Amendment Request #5 to HI-STORM 100 CoC

Dear Sir:

Via letter (Reference 3), the SFST requested that we provide additional information on our proposed amendment (Reference 4) to the HI-STORM 100 Certificate of Compliance (CoC). We herein respond to the SFST's request.

The responses to the request for additional information (RAI-2) are provided in Attachment 1. As a result of the RAI, changes are made to the proposed HI-STORM CoC and Appendix B. These are provided in Attachment 2. As requested in Reference 3, changes made to the Final Safety Analysis Report (FSAR) text are provided in Attachment 3. Justifications for these changes are provided in the response to the RAI that initiated the change. Holtec drawing 4724 is also submitted in Attachment 4.

To aid the SFST review effort, new or revised material in the proposed HI-STORM 100 FSAR, relative to our June 2006 submittal (Reference 4) and previous RAI response (Reference 5), is presented in a different font (Arial).

The following attachments all are provided in electronic format:

Attachment 1: Written Responses to NRC Request for Additional Information.

Attachment 2: Revised Proposed CoC Changes.

Attachment 3: List of Effective Pages and Proposed Revised FSAR Sections (marked as Rev. 4C in footer.)

Document ID: 5014623

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Attachment 4: Holtec Drawing 4724, "HI-TRAC 100D Version IP1 Assembly", Revision 0

Sincerely,

Tammy Morin
Project Manager, LAR 1014-5
Docket No. 72-1014

Distribution:
Mr. Christopher Regan, NRC

E-Mail Distribution (Letter Only):
Holtec Groups 1, 2 and 4
HUG Main and Licensing Committees

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Holtec Letter 5014623 (Response to RAI-2 on LAR 1014-5)

Attachment 1

Written Responses to NRC Request for Additional Information

(7 pages plus this cover sheet)

Attachment 1 to Holtec Letter 5014623
Holtec Responses to NRC RAI

General:

G-1. Propose language in the Certificate of Compliance (CoC) that specifically describes the HI-TRAC model for Indian Point Unit 1 (IPI) as the 75 ton HI-TRAC Version IPI, providing nominal shielding dimensions (thickness of shielding materials), and clarifies that this HI-TRAC model is only for use at IP1 while restoring the currently approved CoC language regarding the 100 ton and 125 ton HI-TRAC models.

The proposed wording, in response to the previous RAI (G-I) suggests that the IP1 HI-TRAC would have a maximum weight limit of 100 tons. This is not accurate, since the cask is designed for a maximum limit of 75 tons. This transfer cask is being designed for one specific site that can only handle a 75 ton maximum load. In addition to the large difference in maximum weight, the IP1 transfer cask has reduced radial shielding due to reduction in radial thickness of the lead and the steel outer shell. A comparison of the 100 ton HI-TRAC 100 and HI-TRAC 100D shows that, while there are some differences between the two, both are truly for a maximum weight limit of 100 tons and have the same radial shielding material thicknesses; the only difference between the 100 and 100D transfer casks is the number of radial ribs in the water (neutron shield) jacket.

This information is necessary to determine compliance with 10 CFR 72.236.

Holtec Response:

In Attachment 2 we provide the proposed change to the CoC Section 1.b as follows:

"Generically, two 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 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. Additionally, there is a HI-TRAC 100D Version IP1 transfer cask that has a maximum weight of 75 tons. This transfer cask will only be used at Indian Point Unit 1. It is shorter with the same cavity diameter as the other HI-TRAC models. The outside diameter differs due to the annular lead shielding thickness of 2.5 inches, and inner and outer shells thicknesses of 0.75 inches each."

Due to the clarification on the HI-TRAC 100D Version IP1, FSAR Chapter 1 and the title of drawing 4724 have been changed. Revision 0 of the drawing is provided in Attachment 4.

G-2. Remove the proposed explicit reference to the HI-STORM 100S Version B overpack from the CoC language. In response to the previous RAI (G-2) the applicant proposed adding explicit reference to the Version B overpack in the CoC language. However, this overpack was added to the HI-STORM system as allowed by 10 CFR 72.48 and was not reviewed and approved by the NRC. Only those proposed changes that have been reviewed and approved by the NRC may be included in the CoC. The previous RAI sought clarification that some variants of the HI-STORM overpacks cannot be used in the anchored configuration. This clarification can be achieved without explicit reference to the Version B (and any other) overpack or other design changes incorporated as allowed by 10 CFR 72.48. The proposed language should read similar to the following:

"The HI-STORM 100A applies to both the 100 and 100S, *with the exception of some overpack variants,* that are classified as..."

This information is necessary to determine compliance with 10 CFR 72.236.

Holtec Response:

Holtec has provided the suggested language in the proposed CoC in Attachment 2.

G-3. Explain the need to have the 14x14E assembly class/array listed as allowable contents for the Multi-Purpose Canister (MPC) 24, 24E, and 24EF.

Since the 14x14E assembly class/array is only comprised of IP1 fuel assemblies and these assemblies are all to be loaded into MPC-32(F)s, it appears there is no longer a practical need for listing this assembly class/array as part of the allowed contents for other MPCs containing pressurized water reactor (PWR) fuel. The applicant should consider the necessity of maintaining the assembly class/array as allowed contents for the other PWR MPCs and/or the removal of it from the MPCs' allowed contents.

This information is necessary to determine compliance with 10 CFR 72.236(a).

Holtec Response:

The proposed CoC in Attachment 2 and Chapter 2 of the FSAR have been modified to exclude loading of the 14x14E into the MPC-24s. Note that this *class/array has been analyzed throughout this FSAR in all PWR MPCs, however it is only to be loaded into the MPC-32 and MPC-32F.*

Shielding:

SH-1 Provide language in the Technical Specifications (TS) that clarifies the definition of non-fuel hardware (NFH) and the limitations regarding the types and locations of non-fuel hardware that may be stored with the different assembly classes/arrays in the different MPCs.

In response to the previous RAI (SH-1) the applicant proposed language to indicate that NFH is not permitted to be stored with the IP1 assembly class/array (i.e., 14xI4E). The proposed language states that, besides Antimony-Beryllium neutron sources, "Other NON-FUEL HARDWARE" is not permitted, which indicates that the neutron sources are classified as NFH. However, the current definition of NFH does not include neutron sources. Thus, the definition of NFH should be modified to include neutron sources. Additionally, the TS should be modified to preclude the loading of neutron sources with assemblies and/or in MPCs and MPC basket locations for which loading of neutron sources is not supported by the applicant's shielding evaluation.

This information is necessary to determine compliance with 10 CFR 72.236(a).

Holtec Response:

Table 2.1-1 Section VIII.C. and Table 2.1-1 Section V.D. of the TS were modified to delete the word "Other" when referring to the NFH for IP1. NFH is not permitted to be stored in an MPC-32 with the 14x14E fuel assembly class/array. It is not Holtec's intent to modify the definition of NFH in the proposed TS at this time to include neutron sources. However, once LAR 1014-3 progresses into rulemaking, we intend to reconcile the CoC proposed here with the changes in LAR 1014-3 and LAR 1014-4.

SH-2 Provide either a shielding analysis of the IP1 fuel as damaged fuel or provide greater justification for why this analysis is not necessary. In response to the previous RAI (SH-2) the applicant states that an analysis is not needed. To support this position, the applicant relies on assembly design, results of assembly inspections, and shielding analyses done for the other MPCs that allow limited loading of damaged fuel assemblies. The staff does not consider this justification sufficient.

Although the assembly design includes a shroud and the cladding is stainless steel, the applicant does not make it clear as to how these assembly characteristics make damage to IP1 assemblies and, under any accident condition, geometric reconfiguration of damaged IP1 assemblies much less likely than for standard PWR assemblies. Also, while inspections of the assemblies have not identified any damage, the inspections were limited in their ability to determine the condition of the entire assembly; they appear to have been visual inspections, which enable inspection of only a small fraction of an assembly. Thus, there are large portions of each assembly that have not been inspected. The applicant notes that it is the inability to fully inspect the assemblies and a lack of sufficient records that, in accordance with the damaged fuel assembly and intact fuel assembly TS definitions, requires classifying the assemblies as damaged. The shielding analyses for damaged fuel in the other MPCs (Final Safety Analysis Report (FSAR) Section 5.4.2.2) is based upon the limitation of damaged fuel to certain locations. The analysis for the MPC-68 appears to be the most relevant; however, damaged fuel is limited to 16 of the outermost basket storage locations. Thus, not all the outermost locations are analyzed for damaged fuel. Also, while other shielding analysis show that assemblies in the basket's periphery locations dominate the dose rates, the contributions from the remaining assemblies are not insignificant, particularly for the neutron dose rate, which is a significant fraction of the overall dose rate in the analyzed accident condition. Thus, the dose rate decrease at the cask midplane may not be as significant for the IP1 MPC-32 as it is for the MPC-68 while the dose rate increase may be more significant in areas such as around the lower parts of the cask, where personnel are likely to be.

Therefore, the applicant should either provide a shielding analysis for damaged fuel loaded in the MPC-32 in the configuration proposed in the amendment (providing dose rates for all three radial locations - locations 1, 2, and 3) or provide greater justification, quantitative as well as qualitative, to support the argument that the analysis is not needed. This justification should include information that details how the design (the presence of the shroud and the stainless steel cladding and shroud) results in a much lower likelihood of assembly damage and geometric reconfiguration, including information regarding the mechanisms for damage to stainless steel versus the cladding material of standard PWR assemblies, the operating environment to which the assemblies were exposed and whether or not the conditions of that environment were conducive to mechanisms for damaging stainless steel, whether any failures occurred during in-core operations, and so forth. Staff recognizes that stainless steels are not susceptible to the hydride concerns that affect zirconium-based alloys; however, staff notes that stainless steels that have been exposed to a sufficient neutron fluence do experience embrittlement.

In addition, while staff understands the purpose of the amendment shielding analysis is to show that the dose rates from the HI-STORM 100 system designed for IP1 are bounded by the dose rates from the standard HI-STORM 100 system and that the applicant's current analysis indicates the presence of a large margin, a true demonstration of the bounding nature of the standard system versus the IP1 system will compare the bounding condition of the standard system with the bounding condition of the IP1 system. It is not clear that the currently proposed supplemental shielding analysis is making such a comparison. Also, with regard to analyses for damaged fuel, as stated in Section 5.4.2.2 of the FSAR, while under normal conditions damaged fuel resembles intact fuel, damaged fuel assemblies cannot be guaranteed to stay intact under accident conditions and may begin to resemble fuel debris in its post-accident configuration. If the applicant selects to justify not analyzing IP1 fuel as damaged fuel by providing a shielding analysis, the staff notes that the review and potential finding of acceptability will be limited to only IP1 fuel loaded in the design configuration presented in the amendment application; in other words, this justification, which then forms a part of the analysis method for this amendment, cannot be extended to other fuel contents or other design configurations.

This information is necessary to determine compliance with 10 CFR 72.236(a) and 72.236 (d).

Holtec Response:

Analysis was performed assuming the IP1 fuel as damaged fuel consistent with the calculations discussed in FSAR Section 5.4.2.2. The analysis consisted of modeling the fuel assemblies in all locations in the MPC-32 with a fuel density that was twice the normal fuel density and correspondingly increasing the source term by a factor of two. The discussion of the methodology and results of the analysis was added to 5.II.1.3 and dose rates are provided in Table 5.II.3.

SH-3 Provide either an analysis that includes the contribution of the Antimony-Beryllium (Sb-Be) sources in the IP1 assemblies or further justification for why this analysis is not necessary.

In response to the previous RAI (SH-3) the applicant references the license amendment request (LAR) 1014-4 that is currently in rulemaking, wherein a few neutron source types were requested for inclusion as allowable contents. The NRC's approval of that change in the contents is based on a restriction of the number of sources (one) and the allowable location (the interior basket region). The more active neutron sources (Plutonium-Beryllium (Pu-Be) and Americium-Beryllium (Am-Be) sources) are described as the reason for imposing these restrictions. Since different limitations were not proposed for the other neutron sources requested in LAR 1014-4, these other neutron sources (including the Sb-Be sources) were only reviewed to ensure they were bounded by the Pu-Be and Am-Be sources. The applicant also states that the source term calculation done for the Dresden Antimony-Beryllium sources is "extremely conservative." However, the basis for this statement is not provided nor does this statement justify why it is acceptable to omit the Antimony-Beryllium sources' contribution. The applicant should justify (quantitatively as well as qualitatively) why the inclusion of the Antimony-Beryllium sources in the analysis is not necessary. This justification should include: 1) a more realistic calculation of the contribution from the IP1 neutron sources with adequate justification of how the calculated source term is realistic and still bounding, and 2) a comparison of the source strength versus an assembly and an individual fuel rod (since the neutron source takes the place of a fuel rod in the assembly). The comparison should show that replacing a fuel rod does not (noticeably) alter the source term. Or, the applicant should include the contribution from the IP1 neutron sources in the IP1 analysis. Again, as stated in the preceding RAI question, while it is recognized that there is large margin in this case, use of an appropriate comparison of bounding conditions is important.

This information is necessary to determine compliance with 10 CFR 72.236(a) and 72.236(d).

Holtec Response:

A more detailed analysis of neutrons being generated in the Be through Be's gamma-n reaction and the gamma radiation from the fuel has been performed. The secondary neutron generation from a source rod would represent less than 0.5% of the neutron source strength of the assembly, and is in fact similar to the source strength of the rod that is replaced by the secondary source. Therefore, it is not necessary to explicitly consider the sources in the dose rate analyses. The discussion of the secondary source analysis and results for IP1 has been added to FSAR Section 5.II.2.1 in regards to Antimony-Beryllium source contribution.

SH-4 Provide further justification for the cobalt impurity level used in the steel in the IP1 analysis. In response to the previous RAI (SH-8 and SH-9) the applicant references ORNL/TM-6501, which uses a Cobalt impurity in steel of 0.8g/kg. Given that the document is from 1978, the impurity level seems appropriate for use with IP1 fuels. The authors of that document assume this impurity level based on a personal communication with a fuel vendor representative, who explained that to achieve this level of cobalt contamination required a judicious selection of the heats from which the nickel-containing metals are taken. Such a statement implies that 0.8g/kg is not a bounding impurity level, but rather a minimum level for steels at the time. The applicant's FSAR also references PNL-6906, Vol. 1, which indicates that fuels with steel manufactured during the same time frame could have as much as 2.2g/kg impurity in the steel. Thus, it would seem that the steel of the IP1 assemblies may have a higher impurity level and therefore a larger contribution to the dose rates. The shielding analysis should include an appropriate level of Cobalt impurity in the assembly's steel components with an appropriate justification (quantitative as well as qualitative) for the assumed impurity level. Such justification should include, if available, any information regarding the steels, such as established impurity limits or composition measurements, used by the IP1 fuel assembly manufacturer(s). As stated in the preceding RAI questions, it is important that an appropriate comparison of the bounding conditions be provided in the shielding analysis.

This information is necessary to determine compliance with 10 CFR 72.236(d).

Holtec Response:

Dose rates for IP1 have been recalculated assuming a bounding cobalt impurity value of 2.2 g/kg for all stainless steel parts and FSAR Section 5.II.2 has been modified to included further discussion.

SH-5 Modify Section 11.II.1.5 of the proposed FSAR to be consistent with the restriction on the transport orientation of the IP1 version of the HI-TRAC. In response to the previous RAI (SH-13) the applicant stated that the HI-TRAC version for IP1 is precluded from horizontal transport. There still remains text in the proposed FSAR that appears to be inconsistent with the restriction. The staff notes, for example, that Section 11.II.1.5 describes upending and downending operations for the IP1 HI-TRAC. The information in the FSAR should be consistent regarding the restriction on the transfer cask's transport orientation.

This information is necessary to determine compliance with 10 CFR 72.1 26(a) and 72.236(d).

Holtec Response:

The text in FSAR Section 11.II.1.5 has been modified to remove the statement pertaining to upending and downending a HI-TRAC 100D Version IP1 in the off-normal handling condition.

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Holtec Letter 5014623 (Response to RAI-2 on LAR 1014-5)

Attachment 2

Revised Proposed CoC Sections

(87 pages plus this cover sheet)

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 No.	Amendment No.	Amendment Effective Date	Package Identification No.
1014	05/31/00	06/01/20	72-1014	~~2~~5	~~06/07/05~~*TBD*	USA/72-1014

Issued To: (Name/Address)

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

Safety Analysis Report Title

Holtec International
Final Safety Analysis Report for the
HI-STORM 100 Cask System

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), 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 MPC, the HI-TRAC transfer cask, and the HI-STORM 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 aluminum heat conduction elements (AHCEs), which are installed in some early-vintage MPCs. 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 neutron absorbers, provides criticality control.

CERTIFICATE OF COMPLIANCE
FOR SPENT FUEL STORAGE CASKS
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1. b. Description (continued)

There are 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. All eight MPC models have the same external diameter.

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. *Generically, t*~~T~~wo 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 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. *Additionally, there is a HI-TRAC 100D Version IP1 transfer cask that has a maximum weight of 75 tons. This transfer cask will only be used at Indian Point Unit 1. It is shorter with the same cavity diameter as the other HI-TRAC models. The outside diameter differs due to the annular lead shielding thickness of 2.5 inches, and inner and outer shells thicknesses of 0.75 inches each.*

The HI-STORM 100 or 100S storage overpack provides shielding and structural protection of the MPC during storage. The HI-STORM 100S is a variation of the HI-STORM 100 overpack design that includes a modified lid 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 four air inlets at the bottom and 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. *With the exception of some overpack variants,* ~~T~~the HI-STORM 100A applies to both the HI-STORM 100 and 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 any HI-STORM 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 regulatory 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.

Each first time user of a HI-STORM 100 Cask System Supplemental Cooling System (SCS) that uses components or a system that is not essentially identical to components or a system that has been previously tested, shall measure and record coolant temperatures for the inlet and outlet of cooling provided to the annulus between the HI-TRAC and MPC and the coolant flow rate. The user shall also record the MPC operating pressure and decay heat. An analysis shall be performed, using this information, that validates the thermal methods described in the FSAR which were used to determine the type and amount of supplemental cooling necessary.

9. SPECIAL REQUIREMENTS FOR FIRST SYSTEMS IN PLACE (continued)

Letter reports summarizing the results of each thermal validation test and SCS validation test and analysis 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 *or cask loading 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 *or cask loading pool*.

f. MPC welding, NDE inspections, pressure testing, draining, moisture removal (by vacuum drying or forced helium dehydration, as applicable), and helium backfilling. . (A mockup may be used for this dry-run exercise.)

g. Operation of the Supplemental Cooling System *if applicable.*

h. Transfer cask upending/downending on the horizontal transfer trailer or other transfer device, as applicable to the site's cask handling arrangement.

I. Transfer of the MPC from the transfer cask to the overpack.

j. Placement of the HI-STORM 100 Cask System at the ISFSI, for aboveground systems only.

k. 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. When the Supplemental Cooling System is in operation to provide for decay heat removal in accordance with Section 3.1.4 of Appendix A the licensee is exempt from the requirements of 10 CR 72.236(f).

CERTIFICATE OF COMPLIANCE FOR SPENT FUEL STORAGE CASKS

Supplemental Sheet

12. 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. The HI-STORM 100 Cask System may be fabricated and used in accordance with any approved amendment to CoC No. 1014 listed in 10 CFR 72.214. Each of the licensed HI-STORM 100 System components (i.e., the MPC, overpack, and transfer cask), if fabricated in accordance with any of the approved CoC Amendments, may be used with one another provided an assessment is performed by the CoC holder that demonstrates design compatibility.

FOR THE U. S. NUCLEAR REGULATORY COMMISSION

Robert Nelson, Chief
Licensing Section
Spent Fuel Project Office
Office of Nuclear Material Safety and Safeguards
Washington, DC 20555

Dated, ~~June 7, 2005~~*TBD*

Attachments:
1. Appendix A
2. Appendix B

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 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.

a. For MPCs partially loaded with array/class 6x6A, 6x6B, 6x6C, 7x7A, or 8x8A fuel assemblies, all remaining 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.

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

2.1.2 Uniform Fuel Loading

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

(continued)

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 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. 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.

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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: (hatched)

REGION 2: (open)

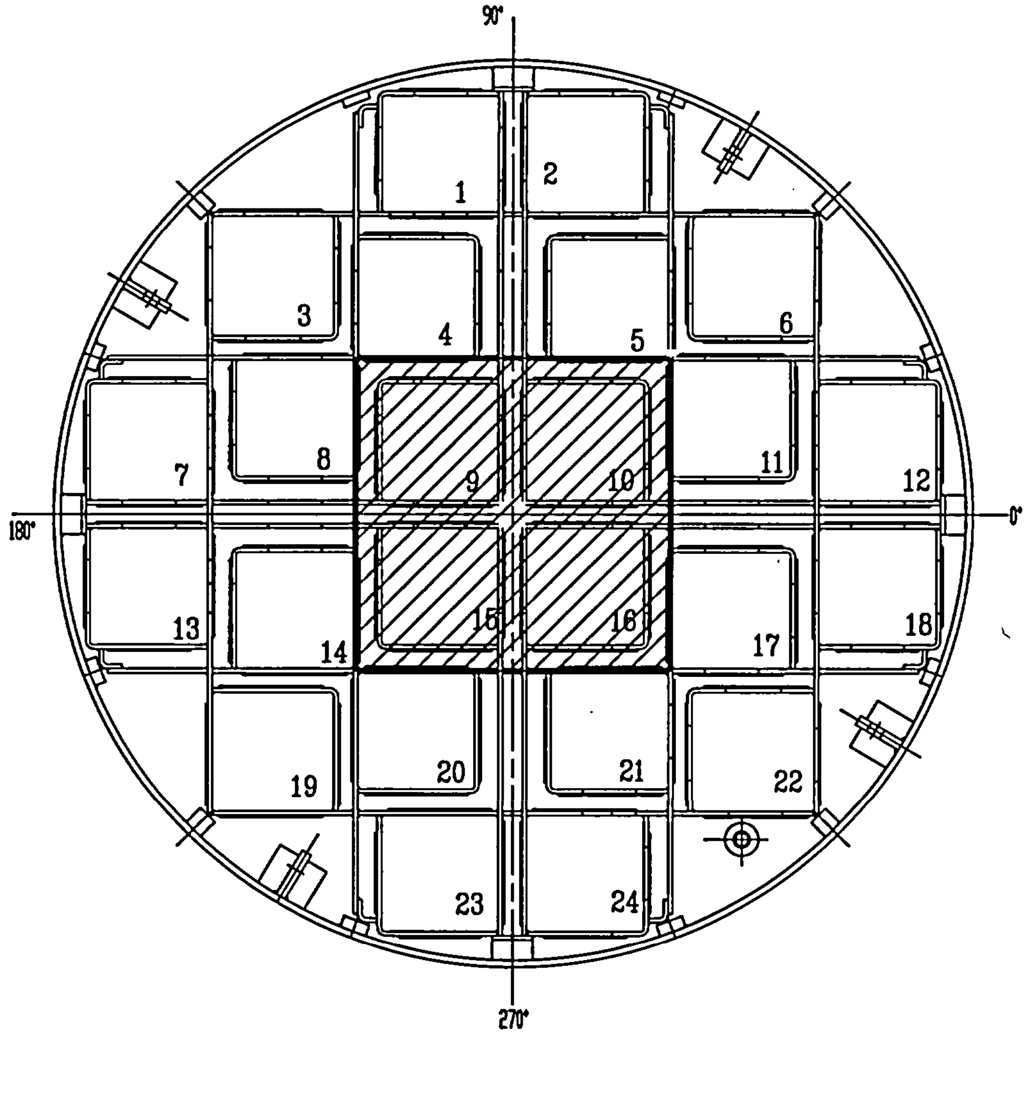

FIGURE 2.1–1
FUEL LOADING REGIONS – MPC–24

LEGEND:

REGION 1: (hatched)

REGION 2: (blank)

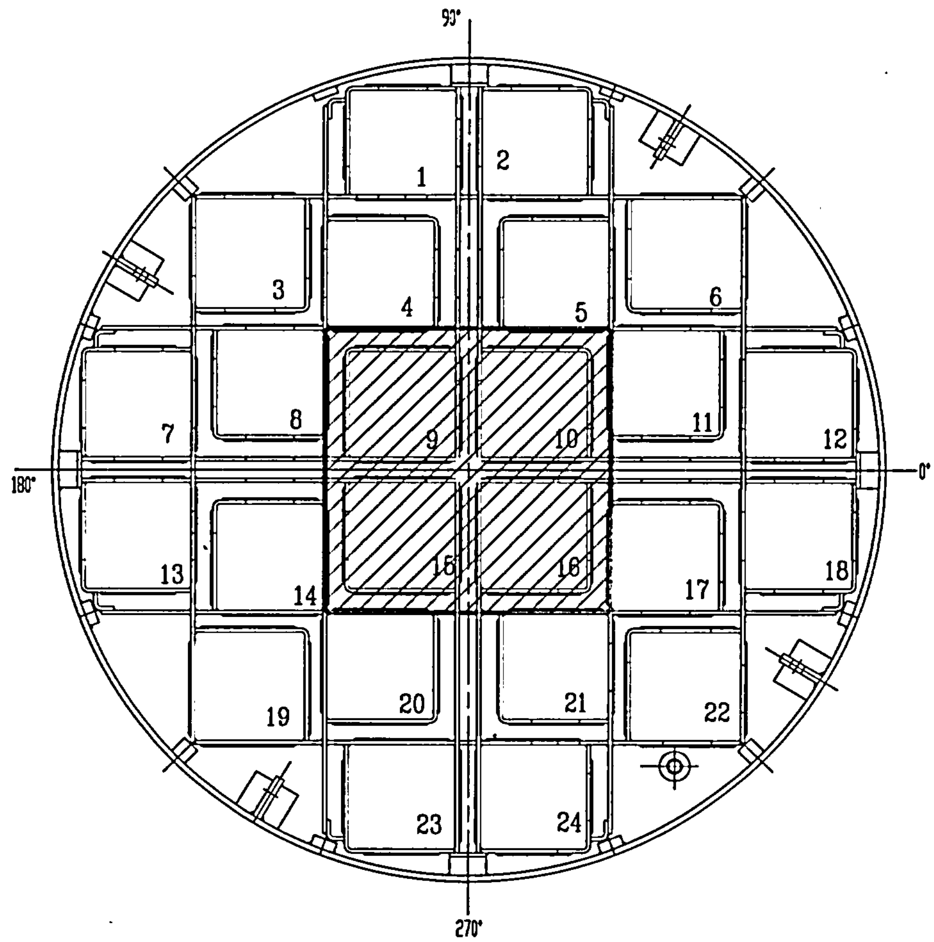

FIGURE 2.1-2

FUEL LOADING REGIONS - MPC-24E/24EF

LEGEND:

REGION 1: (hatched)

REGION 2: (blank)

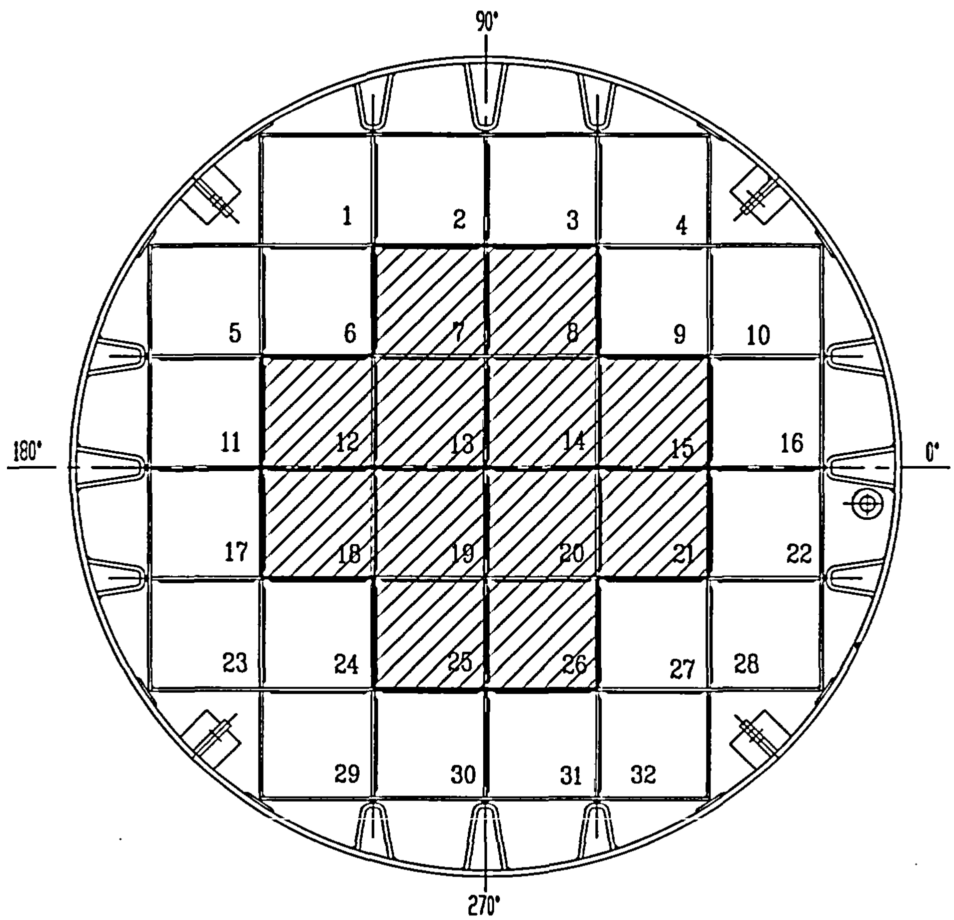

FIGURE 2.1-3

FUEL LOADING REGIONS - MPC-32/32F

APPROVED CONTENTS
2.0

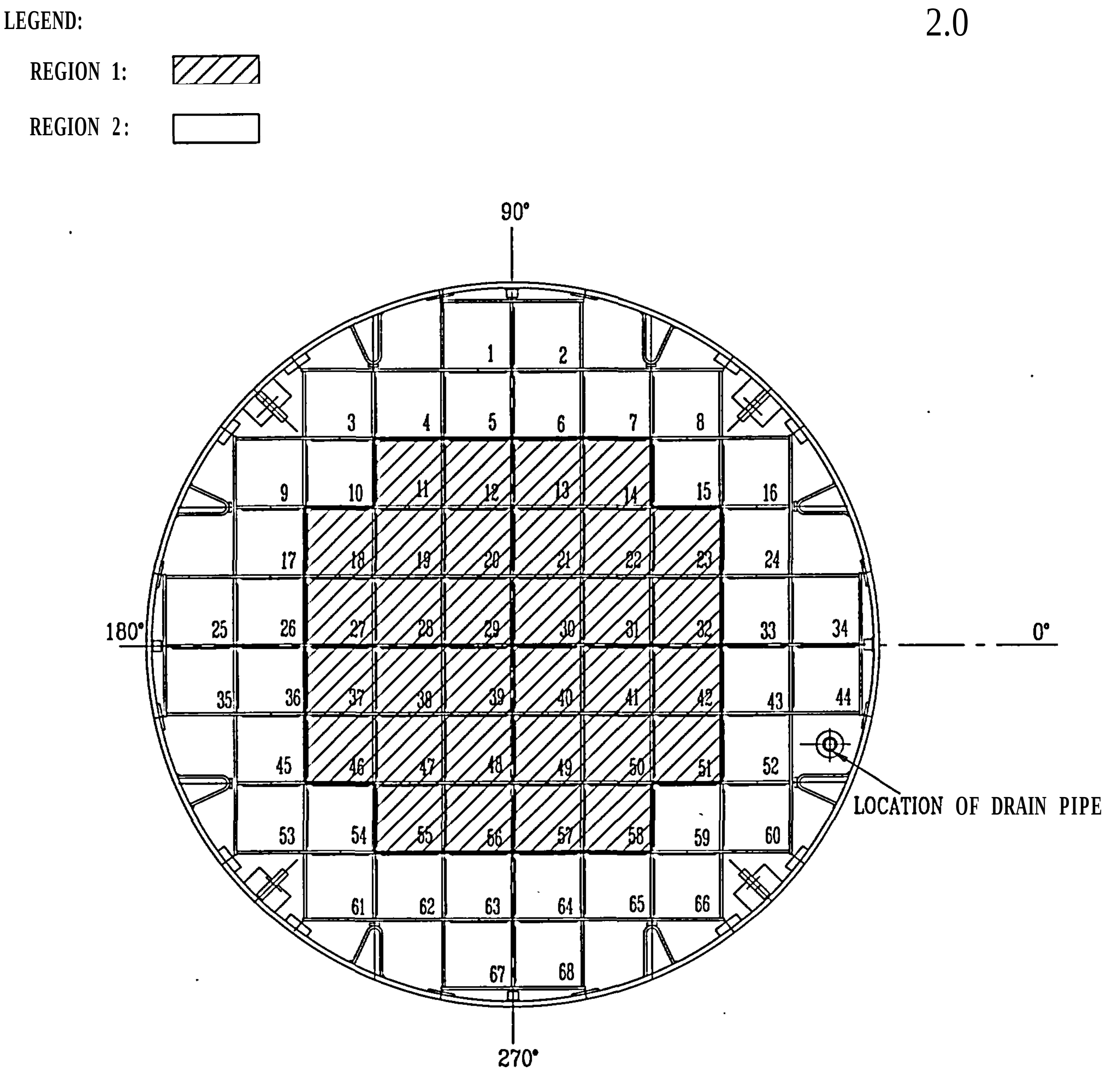

FIGURE 2.1-4

FUEL LOADING REGIONS - MPC-68/68FF

Table 2.1-1 (page 1 of 39)
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:	ZR or Stainless Steel (SS) as specified in Table 2.1-2 for the applicable fuel assembly array/class.
b. Initial Enrichment:	As specified in Table 2.1-2 for the applicable fuel assembly array/class.
c. Post-irradiation Cooling Time and Average Burnup Per Assembly:	
i. Array/Classes 14x14D,~~14x14E~~, and 15x15G	Cooling time $\geq$ 8 years and an average burnup $\leq$ 40,000 MWD/MTU.
ii. All Other Array/Classes	Cooling time and average burnup as specified in Section 2.4.
iii. NON-FUEL HARDWARE	As specified in Table 2.1-8.

Table 2.1-1 (page 2 of 39)
Fuel Assembly Limits

I. MPC MODEL: MPC-24 (continued)

A. Allowable Contents (continued)

d. Decay Heat Per Fuel Storage Location:	
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)

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

C. Deleted.

D. Neutron sources and 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, orifice rod assemblies, or vibration suppressor inserts may be stored in any fuel storage 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 39)
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:	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.

Table 2.1-1 (page 4 of 39)
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	$\leq$ 115 Watts
ii.	Array/Class 8x8F	$\leq$ 183.5 Watts.
iii.	Array/Classes 10x10D and 10x10E	$\leq$ 95 Watts
iv.	All Other Array/Classes	As specified in Section 2.4.

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

Table 2.1-1 (page 5 of 39)
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:	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.

Table 2.1-1 (page 6 of 39)
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.

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 39)
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:	ZR
b. Maximum PLANAR-AVERAGE INITIAL ENRICHMENT:	As specified in Table 2.1-3 for fuel assembly array/class 6x6B.
c. Initial Maximum Rod Enrichment:	As specified in Table 2.1-3 for fuel assembly array/class 6x6B.
d. Post-irradiation Cooling Time and Average Burnup Per Assembly:	Cooling time $\geq$ 18 years and an average burnup $\leq$ 30,000 MWD/MTIHM.
e. Decay Heat Per Assembly:	$\leq$ 115 Watts
f. Fuel Assembly Length:	$\leq$ 135.0 inches (nominal design)
g. Fuel Assembly Width:	$\leq$ 4.70 inches (nominal design)
h. Fuel Assembly Weight:	$\leq$ 400 lbs, including channels

Table 2.1-1 (page 8 of 39)
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:	ZR
b. Maximum PLANAR-AVERAGE INITIAL ENRICHMENT:	As specified in Table 2.1-3 for array/class 6x6B.
c. Initial Maximum Rod Enrichment:	As specified in Table 2.1-3 for array/class 6x6B.
d. Post-irradiation Cooling Time and Average Burnup Per Assembly:	Cooling time $\geq$ 18 years and an average burnup $\leq$ 30,000 MWD/MTIHM.
e. Decay Heat Per Assembly:	$\leq$ 115 Watts
f. Fuel Assembly Length:	$\leq$ 135.0 inches (nominal design)
g. Fuel Assembly Width:	$\leq$ 4.70 inches (nominal design)
h. Fuel Assembly Weight:	$\leq$ 550 lbs, including channels and DFC

Table 2.1-1 (page 9 of 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:	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 39)
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 39)
Fuel Assembly Limits

III. MPC MODEL: MPC-68F

A. Allowable Contents

1. Uranium oxide, BWR INTACT FUEL ASSEMBLIES, with or without 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:	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 39)
Fuel Assembly Limits

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

A. Allowable Contents (continued)

2. Uranium oxide, BWR DAMAGED FUEL ASSEMBLIES, with or without 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:	ZR
b. Maximum PLANAR-AVERAGE INITIAL ENRICHMENT:	As specified in Table 2.1-3 for the applicable fuel assembly array/class.
c. Initial Maximum Rod Enrichment:	As specified in Table 2.1-3 for the applicable fuel assembly array/class.
d. Post-irradiation Cooling Time and Average Burnup Per Assembly:	Cooling time $\geq$ 18 years and an average burnup $\leq$ 30,000 MWD/MTU.
e. Decay Heat Per Assembly:	$\leq$ 115 Watts
f. Fuel Assembly Length:	$\leq$ 135.0 inches (nominal design)
g. Fuel Assembly Width:	$\leq$ 4.70 inches (nominal design)
h. Fuel Assembly Weight:	$\leq$ 550 lbs, including channels and DFC

Table 2.1-1 (page 13 of 39)
Fuel Assembly Limits

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

A. Allowable Contents (continued)

3. Uranium oxide, BWR FUEL DEBRIS, with or without 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:	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 39)
Fuel Assembly Limits

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

A. Allowable Contents (continued)

4. Mixed oxide (MOX), BWR INTACT FUEL ASSEMBLIES, with or without 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:	ZR
b. Maximum PLANAR-AVERAGE INITIAL ENRICHMENT:	As specified in Table 2.1-3 for fuel assembly array/class 6x6B.
c. Initial Maximum Rod Enrichment:	As specified in Table 2.1-3 for fuel assembly array/class 6x6B.
d. Post-irradiation Cooling Time and Average Burnup Per Assembly:	Cooling time $\geq$ 18 years and an average burnup $\leq$ 30,000 MWD/MTIHM.
e. Decay Heat Per Assembly	$\leq$ 115 Watts
f. Fuel Assembly Length:	$\leq$ 135.0 inches (nominal design)
g. Fuel Assembly Width:	$\leq$ 4.70 inches (nominal design)
h. Fuel Assembly Weight:	$\leq$ 400 lbs, including channels

Table 2.1-1 (page 15 of 39)
Fuel Assembly Limits

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

A. Allowable Contents (continued)

5. Mixed oxide (MOX), BWR DAMAGED FUEL ASSEMBLIES, with or without 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:	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 39)
Fuel Assembly Limits

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

A. Allowable Contents (continued)

6. Mixed Oxide (MOX), BWR FUEL DEBRIS, with or without 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:	ZR
b. Maximum PLANAR-AVERAGE INITIAL ENRICHMENT:	As specified in Table 2.1-3 for original fuel assembly array/class 6x6B.
c. Initial Maximum Rod Enrichment:	As specified in Table 2.1-3 for original fuel assembly array/class 6x6B.
d. Post-irradiation Cooling Time and Average Burnup Per Assembly:	Cooling time $\geq$ 18 years and an average burnup $\leq$ 30,000 MWD/MTIHM for the original fuel assembly.
e. Decay Heat Per Assembly	$\leq$ 115 Watts
f. Original Fuel Assembly Length:	$\leq$ 135.0 inches (nominal design)
g. Original Fuel Assembly Width:	$\leq$ 4.70 inches (nominal design)
h. Fuel Debris Weight:	$\leq$ 550 lbs, including channels and DFC

Table 2.1-1 (page 17 of 39)
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:	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 39)
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 39)
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:	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 20 of 39)
Fuel Assembly Limits

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

A. Allowable Contents (continued)

d. Decay Heat Per Fuel Storage Location:

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)

Table 2.1-1 (page 21 of 39)
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:	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 22 of 39)
Fuel Assembly Limits

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

A. Allowable Contents (continued)

d. Decay Heat Per Fuel Storage Location:

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 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. Neutron sources and FUEL DEBRIS are not authorized for loading in the MPC-24E.

Note 1: Fuel assemblies containing BPRAs, TPDs, WABAs, water displacement guide tube plugs, orifice rod assemblies, or vibration supressor inserts may be stored in any fuel storage 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 39)
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:	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. *Array/Class 14x14E*	*Cooling time ≥ 30 years.* *An average burnup ≤ 30,000 MWD/MTU with an enrichment ≥ 3.5 wt. % ^{235}U or, an average burnup ≤ 10,000 MWD/MTU with an enrichment ≥ 2.7 wt. % ^{235}U and < 3.5 wt. % ^{235}U.*
iii. All Other Array/Classes	As specified in Section 2.4.
iv. NON-FUEL HARDWARE	As specified in Table 2.1-8.

Table 2.1-1 (page 24 of 39)
Fuel Assembly Limits

V. MPC MODEL: MPC-32 (continued)

A. Allowable Contents (continued)

d. Decay Heat Per Fuel Storage Location:

i. Array/Classes 14x14D, ~~14x14E~~, and 15x15G	$\leq$ 500 Watts
ii. *Array/Class 14x14E*	*$\leq$ 250 Watts*
*i*ii. All Other Array/Classes	As specified in Section 2.4.
e. Fuel Assembly Length	$\leq$ 176.8 inches (nominal design)
f. Fuel Assembly Width	$\leq$ 8.54 inches (nominal design)
g. Fuel Assembly Weight	$\leq$ 1,680 lbs (including NON-FUEL HARDWARE)

Table 2.1-1 (page 25 of 39)
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 $\geq$ 9 years and an average burnup $\leq$ 30,000 MWD/MTU or cooling time $\geq$ 20 years and an average burnup $\leq$ 40,000 MWD/MTU.
ii. *Array/Class 14x14E*	*Cooling time $\geq$ 30 years.* *An average burnup $\leq$ 30,000 MWD/MTU with an enrichment $\geq$ 3.5 wt. % ^{235}U or, an average burnup $\leq$ 10,000 MWD/MTU with an enrichment $\geq$ 2.7 wt. % ^{235}U and < 3.5 wt. % ^{235}U.*
*i*ii. All Other Array/Classes	As specified in Section 2.4.
i*v*~~ii~~. NON-FUEL HARDWARE	As specified in Table 2.1-8.

Table 2.1-1 (page 26 of 39)
Fuel Assembly Limits

V. MPC MODEL: MPC-32 (continued)

A. Allowable Contents (continued)

d. Decay Heat Per Fuel Storage Location:

i. Array/Classes 14x14D~~,~~ ~~14x14E,~~ and 15x15G	< 500 Watts.
ii. *Array/Class 14x14E*	*< 250 Watts.*
*i*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: *With the exception of array/class 14x14E, u*~~U~~p 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. *For array/class 14x14E, up to 32 INTACT FUEL ASSEMBLIES, and/or DAMAGED FUEL ASSEMBLIES stored in DAMAGED FUEL CONTAINERS.*

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

D. *Array/class 14x14E, Indian Point Unit 1, fuel assemblies may contain an antimony-beryllium secondary source assembly. Up to 32 Indian Point Unit 1 secondary source assemblies may be stored in the MPC-32. Neutron sources are not authorized for loading in the MPC-32 with arrays/classes other than 14x14E.* ~~*Other*~~ *NON-FUEL HARDWARE is not permitted to be stored with array/class 14x14E.*

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 storage 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 27 of 39)
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:	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.
e. 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.

Table 2.1-1 (page 28 of 39)
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.

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
ii. All Other Array/Classes	≤ 700 lbs, including channels

Table 2.1-1 (page 29 of 39)
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:	ZR or Stainless Steel (SS) in accordance with Table 2.1-3 for the applicable fuel assembly array/class.
b. Maximum PLANAR-AVERAGE INITIAL ENRICHMENT:	
i. Array/Classes 6x6A, 6x6B, 6x6C, 7x7A, and 8x8A.	As specified in Table 2.1-3 for the applicable fuel assembly array/class.
ii. All Other Array Classes	$\leq$ 4.0 wt.% ^{235}U.
c. Initial Maximum Rod Enrichment	As specified in Table 2.1-3 for the applicable fuel assembly array/class.
d. Post-irradiation Cooling Time and Average Burnup Per Assembly:	
i. Array/Class 6x6A, 6x6B, 6x6C, 7x7A, or 8x8A	Cooling time $\geq$ 18 years and an average burnup $\leq$ 30,000 MWD/MTU (or MWD/MTIHM).
ii. Array/Class 8x8F	Cooling time $\geq$ 10 years and an average burnup $\leq$ 27,500 MWD/MTU.
iii. Array/Class 10x10D and 10x10E	Cooling time $\geq$ 10 years and an average burnup $\leq$ 22,500 MWD/MTU.
iv. All Other Array/Classes	As specified in Section 2.4.

Table 2.1-1 (page 30 of 39)
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.

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 31 of 39)
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 32 of 39)
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:	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 33 of 39)
Fuel Assembly Limits

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

A. Allowable Contents (continued)

d. Decay Heat Per Fuel Storage Location:	
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)

Table 2.1-1 (page 34 of 39)
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:	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 35 of 39)
Fuel Assembly Limits

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

A. Allowable Contents (continued)

d. Decay Heat Per Fuel Storage Location:

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 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-24EF fuel storage locations may be filled with PWR INTACT FUEL ASSEMBLIES meeting the applicable specifications.

C. Neutron sources are not permitted for loading in the MPC-24EF.

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 storage 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 $\geq$ 9 years and an average burnup $\leq$ 30,000 MWD/MTU or cooling time $\geq$ 20 years and an average burnup $\leq$ 40,000 MWD/MTU.
ii. *Array/Class 14x14E*	*Cooling time $\geq$ 30 years.* *An average burnup $\leq$ 30,000 MWD/MTU with an enrichment $\geq$ 3.5 wt. % ^{235}U or, an average burnup $\leq$ 10,000 MWD/MTU with an enrichment $\geq$ 2.7 wt. % ^{235}U and < 3.5 wt. % ^{235}U.*
iii. All Other Array/Classes	As specified in Section 2.4.
iv. 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 Fuel Storage Location:

i. Array/Classes 14x14D~~,~~ ~~14x14E,~~ and 15x15G	< 500 Watts.
ii. *Array/Class 14x14E*	*< 250 Watts.*
*i*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)

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 $\geq$ 9 years and an average burnup $\leq$ 30,000 MWD/MTU or cooling time $\geq$ 20 years and an average burnup $\leq$ 40,000 MWD/MTU.
ii. *Array/Class 14x14E*	*Cooling time $\geq$ 30 years.* *An average burnup $\leq$ 30,000 MWD/MTU with an enrichment $\geq$ 3.5 wt. % ^{235}U or,* *an average burnup $\leq$ 10,000 MWD/MTU with an enrichment $\geq$ 2.7 wt. % ^{235}U and* *< 3.5 wt. % ^{235}U.*
*i*ii. All Other Array/Classes	As specified in Section 2.4.
i*v*~~ii~~. 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 Fuel Storage Location:

i. Array/Classes 14x14D~~,~~ ~~14x14E,~~ and 15x15G	≤ 500 Watts.
ii. Array/Class 14x14E	*≤ 250 Watts.*
*i*ii. All Other Array/Classes	As specified in Section 2.3.

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: *With the exception of array/class 14x14E, u*~~U~~p 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-32F fuel storage locations may be filled with PWR INTACT FUEL ASSEMBLIES meeting the applicable specifications. *For array/class 14x14E, up to 32 INTACT FUEL ASSEMBLIES and/or DAMAGED FUEL ASSEMBLIES stored in DAMAGED FUEL CONTAINERS. FUEL DEBRIS from array/class 14x14E is not authorized for loading in the MPC-32F.*

C. *Array/class 14x14E, Indian Point Unit 1, fuel assemblies may contain an antimony-beryllium secondary source assembly. Up to 32 Indian Point Unit 1 secondary source assemblies may be stored in the MPC-32*. Neutron sources are not permitted for loading in the MPC-32F *with arrays/classes other than 14x14E*. ~~Other~~ *NON-FUEL HARDWARE is not permitted to be stored with array/class 14x14E.*

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 storage 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	ZR	ZR	ZR	SS	SS
Design Initial U (kg/assy.) (Note 3)	≤ 365	≤ 412	≤ 438	≤ 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)	*N/A* ~~≤ 5.0 *4.5* (24)~~ *N/A* ~~≤ 5.0 *4.5* (24E/24EF)~~
Initial Enrichment (MPC-24, 24E, 24EF, 32, or 32F with soluble boron credit - see Note 5) (wt % ^{235}U)	≤ 5.0	≤ 5.0	≤ 5.0	≤ 5.0	≤ ~~5.0~~ *4.5 (MPC-32/32F only, Note 8)*
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	ZR	ZR	ZR	ZR	ZR	ZR
Design Initial U (kg/assy.) (Note 3)	$\leq$ 473	$\leq$ 473	$\leq$ 473	$\leq$ 495	$\leq$ 495	$\leq$ 495
Initial Enrichment (MPC-24, 24E and 24EF without soluble boron credit) (wt % ^{235}U) (Note 7)	$\leq$ 4.1 (24) $\leq$ 4.5 (24E/24EF)	$\leq$ 4.1 (24) $\leq$ 4.5 (24E/24EF)	$\leq$ 4.1 (24) $\leq$ 4.5 (24E/24EF)	$\leq$ 4.1 (24) $\leq$ 4.5 (24E/24EF)	$\leq$ 4.1 (24) $\leq$ 4.5 (24E/24EF)	$\leq$ 4.1 (24) $\leq$ 4.5 (24E/24EF)
Initial Enrichment (MPC-24, 24E, 24EF, 32, or 32F with soluble boron credit - see Note 5) (wt % ^{235}U)	$\leq$ 5.0	$\leq$ 5.0	$\leq$ 5.0	$\leq$ 5.0	$\leq$ 5.0	$\leq$ 5.0
No. of Fuel Rod Locations	204	204	204	208	208	208
Fuel Rod Clad O.D. (in.)	$\geq$ 0.418	$\geq$ 0.420	$\geq$ 0.417	$\geq$ 0.430	$\geq$ 0.428	$\geq$ 0.428
Fuel Rod Clad I.D. (in.)	$\leq$ 0.3660	$\leq$ 0.3736	$\leq$ 0.3640	$\leq$ 0.3800	$\leq$ 0.3790	$\leq$ 0.3820
Fuel Pellet Dia. (in.)	$\leq$ 0.3580	$\leq$ 0.3671	$\leq$ 0.3570	$\leq$ 0.3735	$\leq$ 0.3707	$\leq$ 0.3742
Fuel Rod Pitch (in.)	$\leq$ 0.550	$\leq$ 0.563	$\leq$ 0.563	$\leq$ 0.568	$\leq$ 0.568	$\leq$ 0.568
Active Fuel Length (in.)	$\leq$ 150	$\leq$ 150	$\leq$ 150	$\leq$ 150	$\leq$ 150	$\leq$ 150
No. of Guide and/or Instrument Tubes	21	21	21	17	17	17
Guide/Instrument Tube Thickness (in.)	$\geq$ 0.0165	$\geq$ 0.015	$\geq$ 0.0165	$\geq$ 0.0150	$\geq$ 0.0140	$\geq$ 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	SS	ZR	ZR	ZR	ZR	ZR
Design Initial U (kg/assy.) (Note 3)	≤ 420	≤ 495	≤ 448	≤ 433	≤ 474	≤ 480
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 5) (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. 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.

8. *This fuel assembly array/class includes only the Indian Point Unit 1 fuel assembly. This assembly class has been analyzed in all PWR MPCs, however, it is only to be loaded in the MPC-32/32F.*

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	ZR	ZR	ZR	ZR	ZR	ZR
Design Initial U (kg/assy.) (Note 3)	$\leq$ 110	$\leq$ 110	$\leq$ 110	$\leq$ 100	$\leq$ 198	$\leq$ 120
Maximum PLANAR-AVERAGE INITIAL ENRICHMENT (wt.% ^{235}U) (Note 14)	$\leq$ 2.7	$\leq$ 2.7 for the UO_2 rods. See Note 4 for MOX rods	$\leq$ 2.7	$\leq$ 2.7	$\leq$ 4.2	$\leq$ 2.7
Initial Maximum Rod Enrichment (wt.% ^{235}U)	$\leq$ 4.0	$\leq$ 4.0	$\leq$ 4.0	$\leq$ 5.5	$\leq$ 5.0	$\leq$ 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.)	$\geq$ 0.5550	$\geq$ 0.5625	$\geq$ 0.5630	$\geq$ 0.4860	$\geq$ 0.5630	$\geq$ 0.4120
Fuel Rod Clad I.D. (in.)	$\leq$ 0.5105	$\leq$ 0.4945	$\leq$ 0.4990	$\leq$ 0.4204	$\leq$ 0.4990	$\leq$ 0.3620
Fuel Pellet Dia. (in.)	$\leq$ 0.4980	$\leq$ 0.4820	$\leq$ 0.4880	$\leq$ 0.4110	$\leq$ 0.4910	$\leq$ 0.3580
Fuel Rod Pitch (in.)	$\leq$ 0.710	$\leq$ 0.710	$\leq$ 0.740	$\leq$ 0.631	$\leq$ 0.738	$\leq$ 0.523
Active Fuel Length (in.)	$\leq$ 120	$\leq$ 120	$\leq$ 77.5	$\leq$ 80	$\leq$ 150	$\leq$ 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	$\geq$ 0
Channel Thickness (in.)	$\leq$ 0.060	$\leq$ 0.060	$\leq$ 0.060	$\leq$ 0.060	$\leq$ 0.120	$\leq$ 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	ZR	ZR	ZR	ZR	ZR	ZR
Design Initial U (kg/assy.) (Note 3)	≤ 192	≤ 190	≤ 190	< 190	≤ 191	≤ 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	ZR	ZR	ZR	ZR	ZR	ZR
Design Initial U (kg/assy.) (Note 3)	≤ 180	≤ 182	≤ 182	≤ 183	≤ 183	≤ 164
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	ZR	ZR	ZR	SS	SS
Design Initial U (kg/assy.) (Note 3)	≤ 188	≤ 188	≤ 179	≤ 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. 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

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Table 2.1-5

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Table 2.1-6 (page 1 of 2)

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Table 2.1-6 (page 2 of 2)

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Table 2.1-7 (page 1 of 2)

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Table 2.1-7 (page 2 of 2)

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Table 2.1-8
NON-FUEL HARDWARE COOLING AND AVERAGE BURNUP (Notes 1, 2, and 3)

Post-irradiation Cooling Time (years)	**INSERTS** (Note 4) **BURNUP (MWD/MTU)**	**GUIDE TUBE HARDWARE** (Note 5) **BURNUP (MWD/MTU)**	**CONTROL COMPONENT** (Note 6) **BURNUP (MWD/MTU)**	**APSR BURNUP (MWD/MTU)**
≥ 3	≤ 24,635	NA (Note 7)	NA	NA
≥ 4	≤ 30,000	≤ 20,000	NA	NA
≥ 5	≤ 36,748	≤ 25,000	≤ 630,000	≤ 45,000
≥ 6	≤ 44,102	≤ 30,000	-	≤ 54,500
≥ 7	≤ 52,900	≤ 40,000	-	≤ 68,000
≥ 8	≤ 60,000	≤ 45,000	-	≤ 83,000
≥ 9	-	≤ 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. Burnups for NON-FUEL HARDWARE are to be determined based on the burnup and uranium mass of the fuel assemblies in which the component was inserted during reactor operation.
2. 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.
3. Applicable to uniform loading and regionalized loading.
4. Includes Burnable Poison Rod Assemblies (BPRAs), Wet Annular Burnable Absorbers (WABAs), and vibration suppressor inserts..
5. Includes Thimble Plug Devices (TPDs), water displacement guide tube plugs, and orifice rod assemblies.
6. Includes Control Rod Assemblies (CRAs), Control Element Assemblies (CEAs), and Rod Cluster Control Assemblies (RCCAs).
7. NA means not authorized for loading at this cooling time.

2.4 Decay Heat, Burnup, and Cooling Time Limits for ZR-Clad Fuel

This section provides the limits on ZR-clad fuel assembly decay heat, burnup, and cooling time for storage in the HI-STORM 100 System. A detailed discussion of how to calculate the limits and verify compliance, including examples, is provided in Chapter 12 of the HI-STORM 100 FSAR.

2.4.1 Uniform Fuel Loading Decay Heat Limits for ZR-clad fuel

Table 2.4-1 provides the maximum allowable decay heat per fuel storage location for ZR-clad fuel in uniform fuel loading for each MPC model.

Table 2.4-1

Maximum Allowable Decay Heat per Fuel Storage Location
(Uniform Loading, ZR-Clad)

MPC Model	**Decay Heat per Fuel Storage Location** (kW)
Intact Fuel Assemblies	
MPC-24	≤ 1.157
MPC-24E/24EF	≤ 1.173
MPC-32/32F	≤ 0.898
MPC-68/68FF	≤ 0.414
Damaged Fuel Assemblies and Fuel Debris	
MPC-24	≤ 1.099
MPC-24E/24EF	≤ 1.114
MPC-32/32F	≤ 0.718
MPC-68/68FF	≤ 0.393

2.4.2 Regionalized Fuel Loading Decay Heat Limits for ZR-Clad Fuel

Table 2.4-2 provides the maximum allowable decay heat per fuel storage location for ZR-clad fuel in regionalized loading for each MPC model.

2.4.2 Regionalized Fuel Loading Decay Heat Limits for ZR-Clad Fuel (cont'd)

Table 2.4-2

Fuel Storage Regions and Maximum Decay Heat per MPC

MPC Model	Number of Fuel Storage Locations in Inner and Outer Regions	Inner Region Maximum Decay Heat per Assembly (kW)	Outer Region Maximum Decay Heat per Assembly (kW)
MPC-24	4 and 20	1.470	0.900
MPC-24E/24EF	4 and 20	1.540	0.900
MPC-32/32F	12 and 20	1.131	0.600
MPC-68/68FF	32 and 36	0.500	0.275

2.4.3 Burnup Limits as a Function of Cooling Time for ZR-Clad Fuel

The maximum allowable fuel assembly average burnup varies with the following parameters:

- Minimum fuel assembly cooling time
- Maximum fuel assembly decay heat
- Minimum fuel assembly average enrichment

The maximum allowable ZR-clad fuel assembly average burnup for a given MINIMUM ENRICHMENT is calculated as described below for minimum cooling times between 3 and 20 years using the maximum permissible decay heat determined in Section 2.4.1 or 2.4.2. Different fuel assembly average burnup limits may be calculated for different minimum enrichments (by individual fuel assembly) for use in choosing the fuel assemblies to be loaded into a given MPC.

2.4.3.1 Choose a fuel assembly minimum enrichment, E_{235}.

2.4.3.2 Calculate the maximum allowable fuel assembly average burnup for a minimum cooling time between 3 and 20 years using the equation below.

$$Bu = (A \times q) + (B \times q^2) + (C \times q^3) + [D \times (E_{235})^2] + (E \times q \times E_{235}) + (F \times q^2 \times E_{235}) + G$$

Equation 2.4.3

Where:

Bu = Maximum allowable average burnup per fuel assembly (MWD/MTU)

2.4.3 Burnup Limits as a Function of Cooling Time for ZR-Clad Fuel (cont'd)

q = Maximum allowable decay heat per fuel storage location determined in Section 2.4.1 or 2.4.2 (kW)

E_{235} = Minimum fuel assembly average enrichment (wt. % ^{235}U) (e.g., for 4.05 wt.%, use 4.05)

A through G = Coefficients from Tables 2.4-3 and 2.4-4 for the applicable fuel assembly array/class and minimum cooling time

2.4.3.3 Calculated burnup limits shall be rounded down to the nearest integer.

2.4.3.4 Calculated burnup limits greater than 68,200 MWD/MTU for PWR fuel and 65,000 MWD/MTU for BWR must be reduced to be equal to these values.

2.4.3.5 Linear interpolation of calculated burnups between cooling times for a given fuel assembly maximum decay heat and minimum enrichment is permitted. For example, the allowable burnup for a cooling time of 4.5 years may be interpolated between those burnups calculated for 4 year and 5 years.

2.4.3.6 Each ZR-clad fuel assembly to be stored must have a MINIMUM ENRICHMENT greater than or equal to the value used in Step 2.4.3.2.

2.4.4 When complying with the maximum fuel storage location decay heat limits, users must account for the decay heat from both the fuel assembly and any NON-FUEL HARDWARE, as applicable for the particular fuel storage location, to ensure the decay heat emitted by all contents in a storage location does not exceed the limit.

Table 2.4-3 (Page 1 of 8)

PWR Fuel Assembly Cooling Time-Dependent Coefficients
(ZR-Clad Fuel)

Cooling Time (years)	Array/Class 14x14A						
	A	B	C	D	E	F	G
≥ 3	20277.1	303.592	-68.329	-139.41	2993.67	-498.159	-615.411
≥ 4	35560.1	-6034.67	985.415	-132.734	3578.92	-723.721	-609.84
≥ 5	48917.9	-14499.5	2976.09	-150.707	4072.55	-892.691	-54.8362
≥ 6	59110.3	-22507	5255.61	-177.017	4517.03	-1024.01	613.36
≥ 7	67595.6	-30158.1	7746.6	-200.128	4898.71	-1123.21	716.004
≥ 8	74424.9	-36871.1	10169.4	-218.676	5203.64	-1190.24	741.163
≥ 9	81405.8	-44093.1	12910.8	-227.916	5405.34	-1223.27	250.224
≥ 10	86184.3	-49211.7	15063.4	-237.641	5607.96	-1266.21	134.435
≥ 11	92024.9	-55666.8	17779.6	-240.973	5732.25	-1282.12	-401.456
≥ 12	94775.8	-58559.7	19249.9	-246.369	5896.27	-1345.42	-295.435
≥ 13	100163	-64813.8	22045.1	-242.572	5861.86	-1261.66	-842.159
≥ 14	103971	-69171	24207	-242.651	5933.96	-1277.48	-1108.99
≥ 15	108919	-75171.1	27152.4	-243.154	6000.2	-1301.19	-1620.63
≥ 16	110622	-76715.2	28210.2	-240.235	6028.33	-1307.74	-1425.5
≥ 17	115582	-82929.7	31411.9	-235.234	5982.3	-1244.11	-1948.05
≥ 18	119195	-87323.5	33881.4	-233.28	6002.43	-1245.95	-2199.41
≥ 19	121882	-90270.6	35713.7	-231.873	6044.42	-1284.55	-2264.05
≥ 20	124649	-93573.5	37853.1	-230.22	6075.82	-1306.57	-2319.63

Table 2.4-3 (Page 2 of 8)

PWR Fuel Assembly Cooling Time-Dependent Coefficients
(ZR-Clad Fuel)

Cooling Time (years)	Array/Class 14x14B						
	A	B	C	D	E	F	G
≥ 3	18937.9	70.2997	-28.6224	-130.732	2572.36	-383.393	-858.17
≥ 4	32058.7	-4960.63	745.224	-125.978	3048.98	-551.656	-549.108
≥ 5	42626.3	-10804.1	1965.09	-139.722	3433.49	-676.643	321.88
≥ 6	51209.6	-16782.3	3490.45	-158.929	3751.01	-761.524	847.282
≥ 7	57829.9	-21982	5009.12	-180.026	4066.65	-846.272	1200.45
≥ 8	62758	-26055.3	6330.88	-196.804	4340.18	-928.336	1413.17
≥ 9	68161.4	-30827.6	7943.87	-204.454	4500.52	-966.347	1084.69
≥ 10	71996.8	-34224.3	9197.25	-210.433	4638.94	-1001.83	1016.38
≥ 11	75567.3	-37486.1	10466.9	-214.95	4759.55	-1040.85	848.169
≥ 12	79296.7	-40900.3	11799.6	-212.898	4794.13	-1040.51	576.242
≥ 13	82257.3	-43594	12935	-212.8	4845.81	-1056.01	410.807
≥ 14	83941.2	-44915.2	13641	-215.389	4953.19	-1121.71	552.724
≥ 15	87228.5	-48130	15056.9	-212.545	4951.12	-1112.5	260.194
≥ 16	90321.7	-50918.3	16285.5	-206.094	4923.36	-1106.35	-38.7487
≥ 17	92836.2	-53314.5	17481.7	-203.139	4924.61	-1109.32	-159.673
≥ 18	93872.8	-53721.4	17865.1	-202.573	4956.21	-1136.9	30.0594
≥ 19	96361.6	-56019.1	19075.9	-199.068	4954.59	-1156.07	-125.917
≥ 20	98647.5	-57795.1	19961.8	-191.502	4869.59	-1108.74	-217.603

Table 2.4-3 (Page 3 of 8)

PWR Fuel Assembly Cooling Time-Dependent Coefficients
(ZR-Clad Fuel)

Cooling Time (years)	Array/Class 14x14C						
	A	B	C	D	E	F	G
≥ 3	19176.9	192.012	-66.7595	-138.112	2666.73	-407.664	-1372.41
≥ 4	32040.3	-4731.4	651.014	-124.944	3012.63	-530.456	-890.059
≥ 5	43276.7	-11292.8	2009.76	-142.172	3313.91	-594.917	-200.195
≥ 6	51315.5	-16920.5	3414.76	-164.287	3610.77	-652.118	463.041
≥ 7	57594.7	-21897.6	4848.49	-189.606	3940.67	-729.367	781.46
≥ 8	63252.3	-26562.8	6273.01	-199.974	4088.41	-732.054	693.879
≥ 9	67657.5	-30350.9	7533.4	-211.77	4283.39	-772.916	588.456
≥ 10	71834.4	-34113.7	8857.32	-216.408	4383.45	-774.982	380.243
≥ 11	75464.1	-37382.1	10063	-218.813	4460.69	-776.665	160.668
≥ 12	77811.1	-39425.1	10934.3	-225.193	4604.68	-833.459	182.463
≥ 13	81438.3	-42785.4	12239.9	-220.943	4597.28	-803.32	-191.636
≥ 14	84222.1	-45291.6	13287.9	-218.366	4608.13	-791.655	-354.59
≥ 15	86700.1	-47582.6	14331.2	-218.206	4655.34	-807.366	-487.316
≥ 16	88104.7	-48601.1	14927.9	-219.498	4729.97	-849.446	-373.196
≥ 17	91103.3	-51332.5	16129	-212.138	4679.91	-822.896	-654.296
≥ 18	93850.4	-53915.8	17336.9	-207.666	4652.65	-799.697	-866.307
≥ 19	96192.9	-55955.8	18359.3	-203.462	4642.65	-800.315	-1007.75
≥ 20	97790.4	-57058.1	19027.7	-200.963	4635.88	-799.721	-951.122

Table 2.4-3 (Page 4 of 8)

PWR Fuel Assembly Cooling Time-Dependent Coefficients
(ZR-Clad Fuel)

Cooling Time (years)	Array/Class 15x15A/B/C						
	A	B	C	D	E	F	G
$\geq$ 3	15789.2	119.829	-21.8071	-127.422	2152.53	-267.717	-580.768
$\geq$ 4	26803.8	-3312.93	415.027	-116.279	2550.15	-386.33	-367.168
$\geq$ 5	36403.6	-7831.93	1219.66	-126.065	2858.32	-471.785	326.863
$\geq$ 6	44046.1	-12375.9	2213.52	-145.727	3153.45	-539.715	851.971
$\geq$ 7	49753.5	-16172.6	3163.61	-166.946	3428.38	-603.598	1186.31
$\geq$ 8	55095.4	-20182.5	4287.03	-183.047	3650.42	-652.92	1052.4
$\geq$ 9	58974.4	-23071.6	5156.53	-191.718	3805.41	-687.18	1025
$\geq$ 10	62591.8	-25800.8	5995.95	-195.105	3884.14	-690.659	868.556
$\geq$ 11	65133.1	-27747.4	6689	-203.095	4036.91	-744.034	894.607
$\geq$ 12	68448.4	-30456	7624.9	-202.201	4083.52	-753.391	577.914
$\geq$ 13	71084.4	-32536.4	8381.78	-201.624	4117.93	-757.16	379.105
$\geq$ 14	73459.5	-34352.3	9068.86	-197.988	4113.16	-747.015	266.536
$\geq$ 15	75950.7	-36469.4	9920.52	-199.791	4184.91	-779.222	57.9429
$\geq$ 16	76929.1	-36845.6	10171.3	-197.88	4206.24	-794.541	256.099
$\geq$ 17	79730	-39134.8	11069.4	-190.865	4160.42	-773.448	-42.6853
$\geq$ 18	81649.2	-40583	11736.1	-187.604	4163.36	-785.838	-113.614
$\geq$ 19	83459	-41771.8	12265.9	-181.461	4107.51	-758.496	-193.442
$\geq$ 20	86165.4	-44208.8	13361.2	-178.89	4107.62	-768.671	-479.778

Table 2.4-3 (Page 5 of 8)

PWR Fuel Assembly Cooling Time-Dependent Coefficients
(ZR-Clad Fuel)

Cooling Time (years)	Array/Class 15x15D/E/F/H						
	A	B	C	D	E	F	G
≥ 3	15192.5	50.5722	-12.3042	-126.906	2009.71	-235.879	-561.574
≥ 4	25782.5	-3096.5	369.096	-113.289	2357.75	-334.695	-254.964
≥ 5	35026.5	-7299.87	1091.93	-124.619	2664	-414.527	470.916
≥ 6	42234.9	-11438.4	1967.63	-145.948	2945.81	-474.981	1016.84
≥ 7	47818.4	-15047	2839.22	-167.273	3208.95	-531.296	1321.12
≥ 8	52730.7	-18387.2	3702.43	-175.057	3335.58	-543.232	1223.61
≥ 9	56254.6	-20999.9	4485.93	-190.489	3547.98	-600.64	1261.55
≥ 10	59874.6	-23706.5	5303.88	-193.807	3633.01	-611.892	1028.63
≥ 11	62811	-25848.4	5979.64	-194.997	3694.14	-618.968	862.738
≥ 12	65557.6	-27952.4	6686.74	-198.224	3767.28	-635.126	645.139
≥ 13	67379.4	-29239.2	7197.49	-200.164	3858.53	-677.958	652.601
≥ 14	69599.2	-30823.8	7768.51	-196.788	3868.2	-679.88	504.443
≥ 15	71806.7	-32425	8360.38	-191.935	3851.65	-669.917	321.146
≥ 16	73662.6	-33703.5	8870.78	-187.366	3831.59	-658.419	232.335
≥ 17	76219.8	-35898.1	9754.72	-189.111	3892.07	-694.244	-46.924
≥ 18	76594.4	-35518.2	9719.78	-185.11	3897.04	-712.82	236.047
≥ 19	78592.7	-36920.8	10316.5	-179.54	3865.84	-709.551	82.478
≥ 20	80770.5	-38599.9	11051.3	-175.106	3858.67	-723.211	-116.014

Table 2.4-3 (Page 6 of 8)

PWR Fuel Assembly Cooling Time-Dependent Coefficients
(ZR-Clad Fuel)

Cooling Time (years)	Array/Class 16X16A						
	A	B	C	D	E	F	G
≥ 3	17038.2	158.445	-37.6008	-136.707	2368.1	-321.58	-700.033
≥ 4	29166.3	-3919.95	508.439	-125.131	2782.53	-455.722	-344.199
≥ 5	40285	-9762.36	1629.72	-139.652	3111.83	-539.804	139.67
≥ 6	48335.7	-15002.6	2864.09	-164.702	3444.97	-614.756	851.706
≥ 7	55274.9	-20190	4258.03	-185.909	3728.11	-670.841	920.035
≥ 8	60646.6	-24402.4	5483.54	-199.014	3903.29	-682.26	944.913
≥ 9	64663.2	-27753.1	6588.21	-215.318	4145.34	-746.822	967.914
≥ 10	69306.9	-31739.1	7892.13	-218.898	4237.04	-746.815	589.277
≥ 11	72725.8	-34676.6	8942.26	-220.836	4312.93	-750.85	407.133
≥ 12	76573.8	-38238.7	10248.1	-224.934	4395.85	-757.914	23.7549
≥ 13	78569	-39794.3	10914.9	-224.584	4457	-776.876	69.428
≥ 14	81559.4	-42453.6	11969.6	-222.704	4485.28	-778.427	-203.031
≥ 15	84108.6	-44680.4	12897.8	-218.387	4460	-746.756	-329.078
≥ 16	86512.2	-46766.8	13822.8	-216.278	4487.79	-759.882	-479.729
≥ 17	87526.7	-47326.2	14221	-218.894	4567.68	-805.659	-273.692
≥ 18	90340.3	-49888.6	15349.8	-212.139	4506.29	-762.236	-513.316
≥ 19	93218.2	-52436.7	16482.4	-207.653	4504.12	-776.489	-837.1
≥ 20	95533.9	-54474.1	17484.2	-203.094	4476.21	-760.482	-955.662

Table 2.4-3 (Page 7 of 8)

PWR Fuel Assembly Cooling Time-Dependent Coefficients
(ZR-Clad Fuel)

Cooling Time (years)	Array/Class 17x17A						
	A	B	C	D	E	F	G
≥ 3	16784.4	3.90244	-10.476	-128.835	2256.98	-287.108	-263.081
≥ 4	28859	-3824.72	491.016	-120.108	2737.65	-432.361	-113.457
≥ 5	40315.9	-9724	1622.89	-140.459	3170.28	-547.749	425.136
≥ 6	49378.5	-15653.1	3029.25	-164.712	3532.55	-628.93	842.73
≥ 7	56759.5	-21320.4	4598.78	-190.58	3873.21	-698.143	975.46
≥ 8	63153.4	-26463.8	6102.47	-201.262	4021.84	-685.431	848.497
≥ 9	67874.9	-30519.2	7442.84	-218.184	4287.23	-754.597	723.305
≥ 10	72676.8	-34855.2	8928.27	-222.423	4382.07	-741.243	387.877
≥ 11	75623	-37457.1	9927.65	-232.962	4564.55	-792.051	388.402
≥ 12	80141.8	-41736.5	11509.8	-232.944	4624.72	-787.134	-164.727
≥ 13	83587.5	-45016.4	12800.9	-230.643	4623.2	-745.177	-428.635
≥ 14	86311.3	-47443.4	13815.2	-228.162	4638.89	-729.425	-561.758
≥ 15	87839.2	-48704.1	14500.3	-231.979	4747.67	-775.801	-441.959
≥ 16	91190.5	-51877.4	15813.2	-225.768	4692.45	-719.311	-756.537
≥ 17	94512	-55201.2	17306.1	-224.328	4740.86	-747.11	-1129.15
≥ 18	96959	-57459.9	18403.8	-220.038	4721.02	-726.928	-1272.47
≥ 19	99061.1	-59172.1	19253.1	-214.045	4663.37	-679.362	-1309.88
≥ 20	100305	-59997.5	19841.1	-216.112	4721.71	-705.463	-1148.45

Table 2.4-3 (Page 8 of 8)

PWR Fuel Assembly Cooling Time-Dependent Coefficients
(ZR-Clad Fuel)

Cooling Time (years)	Array/Class 17x17B/C						
	A	B	C	D	E	F	G
≥ 3	15526.8	18.0364	-9.36581	-128.415	2050.81	-243.915	-426.07
≥ 4	26595.4	-3345.47	409.264	-115.394	2429.48	-350.883	-243.477
≥ 5	36190.4	-7783.2	1186.37	-130.008	2769.53	-438.716	519.95
≥ 6	44159	-12517.5	2209.54	-150.234	3042.25	-489.858	924.151
≥ 7	50399.6	-16780.6	3277.26	-173.223	3336.58	-555.743	1129.66
≥ 8	55453.9	-20420	4259.68	-189.355	3531.65	-581.917	1105.62
≥ 9	59469.3	-23459.8	5176.62	-199.63	3709.99	-626.667	1028.74
≥ 10	63200.5	-26319.6	6047.8	-203.233	3783.02	-619.949	805.311
≥ 11	65636.3	-28258.3	6757.23	-214.247	3972.8	-688.56	843.457
≥ 12	68989.7	-30904.4	7626.53	-212.539	3995.62	-678.037	495.032
≥ 13	71616.6	-32962.2	8360.45	-210.386	4009.11	-666.542	317.009
≥ 14	73923.9	-34748	9037.75	-207.668	4020.13	-662.692	183.086
≥ 15	76131.8	-36422.3	9692.32	-203.428	4014.55	-655.981	47.5234
≥ 16	77376.5	-37224.7	10111.4	-207.581	4110.76	-703.37	161.128
≥ 17	80294.9	-39675.9	11065.9	-201.194	4079.24	-691.636	-173.782
≥ 18	82219.8	-41064.8	11672.1	-195.431	4043.83	-675.432	-286.059
≥ 19	84168.9	-42503.6	12309.4	-190.602	4008.19	-656.192	-372.411
≥ 20	86074.2	-43854.4	12935.9	-185.767	3985.57	-656.72	-475.953

Table 2.4-4 (Page 1 of 10)

BWR Fuel Assembly Cooling Time-Dependent Coefficients
(ZR-Clad Fuel)

Cooling Time (years)	Array/Class 7x7B						
	A	B	C	D	E	F	G
≥ 3	26409.1	28347.5	-16858	-147.076	5636.32	-1606.75	1177.88
≥ 4	61967.8	-6618.31	-4131.96	-113.949	6122.77	-2042.85	-96.7439
≥ 5	91601.1	-49298.3	17826.5	-132.045	6823.14	-2418.49	-185.189
≥ 6	111369	-80890.1	35713.8	-150.262	7288.51	-2471.1	86.6363
≥ 7	126904	-108669	53338.1	-167.764	7650.57	-2340.78	150.403
≥ 8	139181	-132294	69852.5	-187.317	8098.66	-2336.13	97.5285
≥ 9	150334	-154490	86148.1	-193.899	8232.84	-2040.37	-123.029
≥ 10	159897	-173614	100819	-194.156	8254.99	-1708.32	-373.605
≥ 11	166931	-186860	111502	-193.776	8251.55	-1393.91	-543.677
≥ 12	173691	-201687	125166	-202.578	8626.84	-1642.3	-650.814
≥ 13	180312	-215406	137518	-201.041	8642.19	-1469.45	-810.024
≥ 14	185927	-227005	148721	-197.938	8607.6	-1225.95	-892.876
≥ 15	191151	-236120	156781	-191.625	8451.86	-846.27	-1019.4
≥ 16	195761	-244598	165372	-187.043	8359.19	-572.561	-1068.19
≥ 17	200791	-256573	179816	-197.26	8914.28	-1393.37	-1218.63
≥ 18	206068	-266136	188841	-187.191	8569.56	-730.898	-1363.79
≥ 19	210187	-273609	197794	-182.151	8488.23	-584.727	-1335.59
≥ 20	213731	-278120	203074	-175.864	8395.63	-457.304	-1364.38

Table 2.4-4 (Page 2 of 10)

BWR Fuel Assembly Cooling Time-Dependent Coefficients
(ZR-Clad Fuel)

Cooling Time (years)	Array/Class 8x8B						
	A	B	C	D	E	F	G
≥ 3	28219.6	28963.7	-17616.2	-147.68	5887.41	-1730.96	1048.21
≥ 4	66061.8	-10742.4	-1961.82	-123.066	6565.54	-2356.05	-298.005
≥ 5	95790.7	-53401.7	19836.7	-134.584	7145.41	-2637.09	-298.858
≥ 6	117477	-90055.9	41383.9	-154.758	7613.43	-2612.69	-64.9921
≥ 7	134090	-120643	60983	-168.675	7809	-2183.3	-40.8885
≥ 8	148186	-149181	81418.7	-185.726	8190.07	-2040.31	-260.773
≥ 9	159082	-172081	99175.2	-197.185	8450.86	-1792.04	-381.705
≥ 10	168816	-191389	113810	-195.613	8359.87	-1244.22	-613.594
≥ 11	177221	-210599	131099	-208.3	8810	-1466.49	-819.773
≥ 12	183929	-224384	143405	-207.497	8841.33	-1227.71	-929.708
≥ 13	191093	-240384	158327	-204.95	8760.17	-811.708	-1154.76
≥ 14	196787	-252211	169664	-204.574	8810.95	-610.928	-1208.97
≥ 15	203345	-267656	186057	-208.962	9078.41	-828.954	-1383.76
≥ 16	207973	-276838	196071	-204.592	9024.17	-640.808	-1436.43
≥ 17	213891	-290411	211145	-202.169	9024.19	-482.1	-1595.28
≥ 18	217483	-294066	214600	-194.243	8859.35	-244.684	-1529.61
≥ 19	220504	-297897	219704	-190.161	8794.97	-10.9863	-1433.86
≥ 20	227821	-318395	245322	-194.682	9060.96	-350.308	-1741.16

Table 2.4-4 (Page 3 of 10)

BWR Fuel Assembly Cooling Time-Dependent Coefficients
(ZR-Clad Fuel)

Cooling Time (years)	Array/Class 8x8C/D/E						
	A	B	C	D	E	F	G
≥ 3	28592.7	28691.5	-17773.6	-149.418	5969.45	-1746.07	1063.62
≥ 4	66720.8	-12115.7	-1154	-128.444	6787.16	-2529.99	-302.155
≥ 5	96929.1	-55827.5	21140.3	-136.228	7259.19	-2685.06	-334.328
≥ 6	118190	-92000.2	42602.5	-162.204	7907.46	-2853.42	-47.5465
≥ 7	135120	-123437	62827.1	-172.397	8059.72	-2385.81	-75.0053
≥ 8	149162	-152986	84543.1	-195.458	8559.11	-2306.54	-183.595
≥ 9	161041	-177511	103020	-200.087	8632.84	-1864.4	-433.081
≥ 10	171754	-201468	122929	-209.799	8952.06	-1802.86	-755.742
≥ 11	179364	-217723	137000	-215.803	9142.37	-1664.82	-847.268
≥ 12	186090	-232150	150255	-216.033	9218.36	-1441.92	-975.817
≥ 13	193571	-249160	165997	-213.204	9146.99	-1011.13	-1119.47
≥ 14	200034	-263671	180359	-210.559	9107.54	-694.626	-1312.55
≥ 15	205581	-275904	193585	-216.242	9446.57	-1040.65	-1428.13
≥ 16	212015	-290101	207594	-210.036	9212.93	-428.321	-1590.7
≥ 17	216775	-299399	218278	-204.611	9187.86	-398.353	-1657.6
≥ 18	220653	-306719	227133	-202.498	9186.34	-181.672	-1611.86
≥ 19	224859	-314004	235956	-193.902	8990.14	145.151	-1604.71
≥ 20	228541	-320787	245449	-200.727	9310.87	-230.252	-1570.18

Table 2.4-4 (Page 4 of 10)

BWR Fuel Assembly Cooling Time-Dependent Coefficients
(ZR-Clad Fuel)

Cooling Time (years)	Array/Class 9x9A						
	A	B	C	D	E	F	G
≥ 3	30538.7	28463.2	-18105.5	-150.039	6226.92	-1876.69	1034.06
≥ 4	71040.1	-16692.2	1164.15	-128.241	7105.27	-2728.58	-414.09
≥ 5	100888	-60277.7	24150.1	-142.541	7896.11	-3272.86	-232.197
≥ 6	124846	-102954	50350.8	-161.849	8350.16	-3163.44	-91.1396
≥ 7	143516	-140615	76456.5	-185.538	8833.04	-2949.38	-104.802
≥ 8	158218	-171718	99788.2	-196.315	9048.88	-2529.26	-259.929
≥ 9	172226	-204312	126620	-214.214	9511.56	-2459.19	-624.954
≥ 10	182700	-227938	146736	-215.793	9555.41	-1959.92	-830.943
≥ 11	190734	-246174	163557	-218.071	9649.43	-1647.5	-935.021
≥ 12	199997	-269577	186406	-223.975	9884.92	-1534.34	-1235.27
≥ 13	207414	-287446	204723	-228.808	10131.7	-1614.49	-1358.61
≥ 14	215263	-306131	223440	-220.919	9928.27	-988.276	-1638.05
≥ 15	221920	-321612	239503	-217.949	9839.02	-554.709	-1784.04
≥ 16	226532	-331778	252234	-216.189	9893.43	-442.149	-1754.72
≥ 17	232959	-348593	272609	-219.907	10126.3	-663.84	-1915.3
≥ 18	240810	-369085	296809	-219.729	10294.6	-859.302	-2218.87
≥ 19	244637	-375057	304456	-210.997	10077.8	-425.446	-2127.83
≥ 20	248112	-379262	309391	-204.191	9863.67	100.27	-2059.39

Table 2.4-4 (Page 5 of 10)

BWR Fuel Assembly Cooling Time-Dependent Coefficients
(ZR-Clad Fuel)

Cooling Time (years)	Array/Class 9x9B						
	A	B	C	D	E	F	G
≥ 3	30613.2	28985.3	-18371	-151.117	6321.55	-1881.28	988.92
≥ 4	71346.6	-15922.9	631.132	-128.876	7232.47	-2810.64	-471.737
≥ 5	102131	-60654.1	23762.7	-140.748	7881.6	-3156.38	-417.979
≥ 6	127187	-105842	51525.2	-162.228	8307.4	-2913.08	-342.13
≥ 7	146853	-145834	79146.5	-185.192	8718.74	-2529.57	-484.885
≥ 8	162013	-178244	103205	-197.825	8896.39	-1921.58	-584.013
≥ 9	176764	-212856	131577	-215.41	9328.18	-1737.12	-1041.11
≥ 10	186900	-235819	151238	-218.98	9388.08	-1179.87	-1202.83
≥ 11	196178	-257688	171031	-220.323	9408.47	-638.53	-1385.16
≥ 12	205366	-280266	192775	-223.715	9592.12	-472.261	-1661.6
≥ 13	215012	-306103	218866	-231.821	9853.37	-361.449	-1985.56
≥ 14	222368	-324558	238655	-228.062	9834.57	3.47358	-2178.84
≥ 15	226705	-332738	247316	-224.659	9696.59	632.172	-2090.75
≥ 16	233846	-349835	265676	-221.533	9649.93	913.747	-2243.34
≥ 17	243979	-379622	300077	-222.351	9792.17	1011.04	-2753.36
≥ 18	247774	-386203	308873	-220.306	9791.37	1164.58	-2612.25
≥ 19	254041	-401906	327901	-213.96	9645.47	1664.94	-2786.2
≥ 20	256003	-402034	330566	-215.242	9850.42	1359.46	-2550.06

Table 2.4-4 (Page 6 of 10)

BWR Fuel Assembly Cooling Time-Dependent Coefficients
(ZR-Clad Fuel)

Cooling Time (years)	Array/Class 9x9C/D						
	A	B	C	D	E	F	G
≥ 3	30051.6	29548.7	-18614.2	-148.276	6148.44	-1810.34	1006
≥ 4	70472.7	-14696.6	-233.567	-127.728	7008.69	-2634.22	-444.373
≥ 5	101298	-59638.9	23065.2	-138.523	7627.57	-2958.03	-377.965
≥ 6	125546	-102740	49217.4	-160.811	8096.34	-2798.88	-259.767
≥ 7	143887	-139261	74100.4	-184.302	8550.86	-2517.19	-275.151
≥ 8	159633	-172741	98641.4	-194.351	8636.89	-1838.81	-486.731
≥ 9	173517	-204709	124803	-212.604	9151.98	-1853.27	-887.137
≥ 10	182895	-225481	142362	-218.251	9262.59	-1408.25	-978.356
≥ 11	192530	-247839	162173	-217.381	9213.58	-818.676	-1222.12
≥ 12	201127	-268201	181030	-215.552	9147.44	-232.221	-1481.55
≥ 13	209538	-289761	203291	-225.092	9588.12	-574.227	-1749.35
≥ 14	216798	-306958	220468	-222.578	9518.22	-69.9307	-1919.71
≥ 15	223515	-323254	237933	-217.398	9366.52	475.506	-2012.93
≥ 16	228796	-334529	250541	-215.004	9369.33	662.325	-2122.75
≥ 17	237256	-356311	273419	-206.483	9029.55	1551.3	-2367.96
≥ 18	242778	-369493	290354	-215.557	9600.71	659.297	-2589.32
≥ 19	246704	-377971	302630	-210.768	9509.41	1025.34	-2476.06
≥ 20	249944	-382059	308281	-205.495	9362.63	1389.71	-2350.49

Table 2.4-4 (Page 7 of 10)

BWR Fuel Assembly Cooling Time-Dependent Coefficients
(ZR-Clad Fuel)

Cooling Time (years)	Array/Class 9x9E/F						
	A	B	C	D	E	F	G
≥ 3	30284.3	26949.5	-16926.4	-147.914	6017.02	-1854.81	1026.15
≥ 4	69727.4	-17117.2	1982.33	-127.983	6874.68	-2673.01	-359.962
≥ 5	98438.9	-58492	23382.2	-138.712	7513.55	-3038.23	-112.641
≥ 6	119765	-95024.1	45261	-159.669	8074.25	-3129.49	221.182
≥ 7	136740	-128219	67940.1	-182.439	8595.68	-3098.17	315.544
≥ 8	150745	-156607	88691.5	-193.941	8908.73	-2947.64	142.072
≥ 9	162915	-182667	109134	-198.37	8999.11	-2531	-93.4908
≥ 10	174000	-208668	131543	-210.777	9365.52	-2511.74	-445.876
≥ 11	181524	-224252	145280	-212.407	9489.67	-2387.49	-544.123
≥ 12	188946	-240952	160787	-210.65	9478.1	-2029.94	-652.339
≥ 13	193762	-250900	171363	-215.798	9742.31	-2179.24	-608.636
≥ 14	203288	-275191	196115	-218.113	9992.5	-2437.71	-1065.92
≥ 15	208108	-284395	205221	-213.956	9857.25	-1970.65	-1082.94
≥ 16	215093	-301828	224757	-209.736	9789.58	-1718.37	-1303.35
≥ 17	220056	-310906	234180	-201.494	9541.73	-1230.42	-1284.15
≥ 18	224545	-320969	247724	-206.807	9892.97	-1790.61	-1381.9
≥ 19	226901	-322168	250395	-204.073	9902.14	-1748.78	-1253.22
≥ 20	235561	-345414	276856	-198.306	9720.78	-1284.14	-1569.18

Table 2.4-4 (Page 8 of 10)

BWR Fuel Assembly Cooling Time-Dependent Coefficients
(ZR-Clad Fuel)

Cooling Time (years)	Array/Class 9x9G						
	A	B	C	D	E	F	G
≥ 3	35158.5	26918.5	-17976.7	-149.915	6787.19	-2154.29	836.894
≥ 4	77137.2	-19760.1	2371.28	-130.934	8015.43	-3512.38	-455.424
≥ 5	113405	-77931.2	35511.2	-150.637	8932.55	-4099.48	-629.806
≥ 6	139938	-128700	68698.3	-173.799	9451.22	-3847.83	-455.905
≥ 7	164267	-183309	109526	-193.952	9737.91	-3046.84	-737.992
≥ 8	182646	-227630	146275	-210.936	10092.3	-2489.3	-1066.96
≥ 9	199309	-270496	184230	-218.617	10124.3	-1453.81	-1381.41
≥ 10	213186	-308612	221699	-235.828	10703.2	-1483.31	-1821.73
≥ 11	225587	-342892	256242	-236.112	10658.5	-612.076	-2134.65
≥ 12	235725	-370471	285195	-234.378	10604.9	118.591	-2417.89
≥ 13	247043	-404028	323049	-245.79	11158.2	-281.813	-2869.82
≥ 14	253649	-421134	342682	-243.142	11082.3	400.019	-2903.88
≥ 15	262750	-448593	376340	-245.435	11241.2	581.355	-3125.07
≥ 16	270816	-470846	402249	-236.294	10845.4	1791.46	-3293.07
≥ 17	279840	-500272	441964	-241.324	11222.6	1455.84	-3528.25
≥ 18	284533	-511287	458538	-240.905	11367.2	1459.68	-3520.94
≥ 19	295787	-545885	501824	-235.685	11188.2	2082.21	-3954.2
≥ 20	300209	-556936	519174	-229.539	10956	2942.09	-3872.87

Table 2.4-4 (Page 9 of 10)

BWR Fuel Assembly Cooling Time-Dependent Coefficients
(ZR-Clad Fuel)

Cooling Time (years)	Array/Class 10x10A/B						
	A	B	C	D	E	F	G
≥ 3	29285.4	27562.2	-16985	-148.415	5960.56	-1810.79	1001.45
≥ 4	67844.9	-14383	395.619	-127.723	6754.56	-2547.96	-369.267
≥ 5	96660.5	-55383.8	21180.4	-137.17	7296.6	-2793.58	-192.85
≥ 6	118098	-91995	42958	-162.985	7931.44	-2940.84	60.9197
≥ 7	135115	-123721	63588.9	-171.747	8060.23	-2485.59	73.6219
≥ 8	148721	-151690	84143.9	-190.26	8515.81	-2444.25	-63.4649
≥ 9	160770	-177397	104069	-197.534	8673.6	-2101.25	-331.046
≥ 10	170331	-198419	121817	-213.692	9178.33	-2351.54	-472.844
≥ 11	179130	-217799	138652	-209.75	9095.43	-1842.88	-705.254
≥ 12	186070	-232389	151792	-208.946	9104.52	-1565.11	-822.73
≥ 13	192407	-246005	164928	-209.696	9234.7	-1541.54	-979.245
≥ 14	200493	-265596	183851	-207.639	9159.83	-1095.72	-1240.61
≥ 15	205594	-276161	195760	-213.491	9564.23	-1672.22	-1333.64
≥ 16	209386	-282942	204110	-209.322	9515.83	-1506.86	-1286.82
≥ 17	214972	-295149	217095	-202.445	9292.34	-893.6	-1364.97
≥ 18	219312	-302748	225826	-198.667	9272.27	-878.536	-1379.58
≥ 19	223481	-310663	235908	-194.825	9252.9	-785.066	-1379.62
≥ 20	227628	-319115	247597	-199.194	9509.02	-1135.23	-1386.19

Table 2.4-4 (Page 10 of 10)

BWR Fuel Assembly Cooling Time-Dependent Coefficients
(ZR-Clad Fuel)

Cooling Time (years)	Array/Class 10x10C						
	A	B	C	D	E	F	G
$\geq$ 3	31425.3	27358.9	-17413.3	-152.096	6367.53	-1967.91	925.763
$\geq$ 4	71804	-16964.1	1000.4	-129.299	7227.18	-2806.44	-416.92
$\geq$ 5	102685	-62383.3	24971.2	-142.316	7961	-3290.98	-354.784
$\geq$ 6	126962	-105802	51444.6	-164.283	8421.44	-3104.21	-186.615
$\geq$ 7	146284	-145608	79275.5	-188.967	8927.23	-2859.08	-251.163
$\geq$ 8	162748	-181259	105859	-199.122	9052.91	-2206.31	-554.124
$\geq$ 9	176612	-214183	133261	-217.56	9492.17	-1999.28	-860.669
$\geq$ 10	187756	-239944	155315	-219.56	9532.45	-1470.9	-1113.42
$\geq$ 11	196580	-260941	174536	-222.457	9591.64	-944.473	-1225.79
$\geq$ 12	208017	-291492	204805	-233.488	10058.3	-1217.01	-1749.84
$\geq$ 13	214920	-307772	221158	-234.747	10137.1	-897.23	-1868.04
$\geq$ 14	222562	-326471	240234	-228.569	9929.34	-183.47	-2016.12
$\geq$ 15	228844	-342382	258347	-226.944	9936.76	117.061	-2106.05
$\geq$ 16	233907	-353008	270390	-223.179	9910.72	360.39	-2105.23
$\geq$ 17	244153	-383017	304819	-227.266	10103.2	380.393	-2633.23
$\geq$ 18	249240	-395456	321452	-226.989	10284.1	169.947	-2623.67
$\geq$ 19	254343	-406555	335240	-220.569	10070.5	764.689	-2640.2
$\geq$ 20	260202	-421069	354249	-216.255	10069.9	854.497	-2732.77

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Holtec Letter 5014623 (Response to RAI-2 on LAR 1014-5)

Attachment 3

List of Effective Pages and Proposed Revised FSAR Sections

(135 pages plus this cover sheet)

LIST OF EFFECTIVE PAGES FOR HISTORM 100 LAR 1014-5, RAI-2

Page(s)	Revision
1.0-1 through 1.0-31	***4C***
1.1-1 through 1.1-4	***4A***
Fig. 1.1.1	0
Fig. 1.1.1A	1
Fig. 1.1.1B	2
Fig. 1.1.2	0
Fig. 1.1.3 through 1.1.3A	1
Fig. 1.1.3B	2
Fig 1.1.4 through 1.1.5	1
1.2-1 through 1.2-40	***4A***
Fig. 1.2.1	1
Fig. 1.2.1A	2
Fig. 1.2.2	4
Fig. 1.2.3	3
Fig. 1.2.4	4
Fig. 1.2.5 through 1.2.6	0
Fig. 1.2.7	1
Fig. 1.2.8 through 1.2.8A	1
Fig. 1.2.9 through 1.2.9A	1
Fig. 1.2.10 through 1.2.12	0
Fig. 1.2.13 through 1.2.15	Deleted
Fig. 1.2.16a through 1.2.16f	0
Fig. 1.2.17a through 1.2.16d	0
Fig. 1.2.18	1
1.3-1	***4A***
1.4-1 through 1.4-3	1
Fig. 1.4.1 through 1.4.2	0
1.5-1 through 1.5-2	***4C***
Drawings	See Section 1.5
Bills-of-Material	See Section 1.5
1.6-1 through 1.6-2	3
1.A-1 through 1.A-7	4
Fig. 1.A.1 through 1.A.5	0
1.B-1 through 1.B-3	3
1.C-1	2
1.C-2 through 1.C-5	Deleted
1.D-1 through 1.D-7	4
1.II-1 through 1.II-4	***4A***

LIST OF EFFECTIVE PAGES FOR HISTORM 100 LAR 1014-5, RAI-2

Page(s)	Revision
2.0-1 through 2.0-36	4
2.1-1 through 2.1-68	***4C***
Fig. 2.1.1	3
Fig. 2.1.2 through 2.1.2C	1
Fig. 2.1.2D	3
Fig. 2.1.3 through 2.1.5	0
Fig. 2.1.6 through 2.1.8	Deleted
Fig. 2.1.9	3
2.2-1 through 2.2-58	4
2.3-1 through 2.3-22	4
Fig. 2.3.1 through 2.3.4	0
2.4-1 through 2.4-3	3
2.5-1	0
2.6-1 through 2.6-3	3
2A-1 through 2.A-5	2
Fig. 2.A.1	1
2.B-1 through 2.B-4	3
Fig. 2.B.1	3
2.C-1 through 2.C-2	3
Fig. 2.C.1	3
2.II-1 through 2.II-5	***4B***
Fig. 2.II.1	***4A***

Page(s)	Revision
3.0-1 through 3.0-10	4
3.1-1 through 3.1-46	4
Fig. 3.1.1 through 3.1.3	1
3.2-1 through 3.2-14	4
3.3-1 through 3.3-11	3
3.4-1 through 3.4-135	4
Fig. 3.4.1	0
Fig. 3.4.2	1
Fig. 3.4.3 through 3.4.4	0
Fig. 3.4.5	1
Fig. 3.4.6 through 3.4.9	0
Fig. 3.4.10	1
Fig. 3.4.11 through 3.4.29	0
Fig. 3.4.30 through 3.4.49	1
3.5-1 through 3.5-19	0
Fig. 3.5.1 through 3.5.9	0
3.6-1 through 3.6-9	3
3.7-1 through 3.7-8	3
3.8-1 through 3.8-2	2
3.A-1 through 3.A-15	1
Fig. 3.A.1 through 3.A.18	0
3.II-1 through 3.II-18	***4B***
Fig. 3.II.1	***4A***

LIST OF EFFECTIVE PAGES FOR HISTORM 100 LAR 1014-5, RAI-2

Page(s)	Revision
4.0-1 through 4.0-2	3
4.1-1 through 4.1-5	3
4.2-1 through 4.2-11	3
Fig. 4.2.1 through 4.2.2	Deleted
Fig. 4.2.3	3
4.3-1 through 4.3-2	3
4.4-1 through 4.4-65	3
Fig. 4.4.1 through 4.4.6	0
Fig. 4.4.7	2
Fig. 4.4.8	Deleted
Fig. 4.4.9	1
Fig. 4.4.10	0
Fig. 4.4.11	Deleted
Fig. 4.4.12 through 4.4.13	0
Fig. 4.4.14 through 4.4.15	Deleted
Fig. 4.4.16 through 4.4.17	1
Fig. 4.4.18	Deleted
Fig. 4.4.19 through 4.4.20	1
Fig. 4.4.21 through 4.4.23	Deleted
Fig. 4.4.24	0
Fig. 4.4.25 through 4.4.26	1
Fig. 4.4.27	3
4.5-1 through 4.5-26	4
Fig. 4.5.1	0
Fig. 4.5.2	1
4.6-1 through 4.6-2	3
4.7-1 through 4.7-3	3
4.A-1	Deleted
4.B-1 through 4.B-9	3
Fig. 4.B.1 through 4.B.8	1
4.II-1 through 4.II-8	***4A***

LIST OF EFFECTIVE PAGES FOR HISTORM 100 LAR 1014-5, RAI-2

Page(s)	Revision
5.0-1 through 5.0-3	4
5.1-1 through 5.1-26	4
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Fig. 5.1.2	0
Fig. 5.1.3	3
Fig. 5.1.4	0
Fig. 5.1.5	0
Fig. 5.1.6	0
Fig. 5.1.7	0
Fig. 5.1.8	0
Fig. 5.1.9	0
Fig. 5.1.10	0
Fig. 5.1.11	0
Fig. 5.1.12	1
Fig. 5.1.13	2
5.2-1 through 5.2-58	***4A***
5.3-1 through 5.3-15	4
Fig. 5.3.1	1
Fig. 5.3.2	0
Fig. 5.3.3	0
Fig. 5.3.4	2
Fig. 5.3.5	0
Fig. 5.3.6	0
Fig. 5.3.7	1
Fig. 5.3.8	0
Fig. 5.3.9	0
Fig. 5.3.10	3
Fig. 5.3.11	1
Fig. 5.3.12	4
Fig. 5.3.13	4
Fig. 5.3.14	4
Fig. 5.3.15	1
Fig. 5.3.16	1
Fig. 5.3.17	1
Fig. 5.3.18	1
Fig. 5.3.19	2
Fig. 5.3.20	1
Fig. 5.3.21	1
Fig. 5.3.22	2
5.4-1 through 5.4-35	4
5.5-1	0
5.6-1 through 5.6-3	3
5.A-1 through 5.A-4	0
5.B-1 through 5.B-7	0
5.C-1through 5.C-61	0
5.D-1 through 5.D-6	1
5.E-1 through 5.E-4	1
5.F-1 through 5.F-4	3
Fig. 5.F.1 through 5.F.7	3
5.II-1 through 5.II-10	***4C***

LIST OF EFFECTIVE PAGES FOR HISTORM 100 LAR 1014-5, RAI-2

Page(s)	Revision
6.1-1 through 6.1-19	***4A***
6.2-1 through 6.2-64	3
6.3-1 through 6.3-19	3
Fig. 6.3.1	1
Fig. 6.3.1A	1
Fig. 6.3.2	1
Fig. 6.3.3	0
Fig. 6.3.4	1
Fig. 6.3.5	1
Fig. 6.3.6	0
Fig. 6.3.7	1
6.4-1 through 6.4-34	***4A***
Fig. 6.4.1	Deleted
Fig. 6.4.2 through 6.4.15	1
Fig. 6.4.16	3
Fig. 6.4.17	3
6.5-1	0
6.6-1	0
6.7-1 and 6.7-2	1
6.A-1 through 6.A-20	2
Fig. 6.A.1 through 6.A.5	0
Fig. 6.A.6	2
6.B-1 and 6.B-2	0
6.C-1 through 6.C-16	***4A***
6.D-1 through 6.D-35	1
6.II-1 through 6.II-6	***4A***
Fig. 6.II.1	***4A***

Page(s)	Revision
7.0-1	3
7.1-1 through 7.1-8	4
Fig. 7.1.1	0
7.2-1	3
7.3-1	3
7.4-1	3
7.II-1	***4A***

LIST OF EFFECTIVE PAGES FOR HISTORM 100 LAR 1014-5, RAI-2

Page(s)	Revision
8.0-1 through 8.0-6	***4A***
8.1-1 through 8.1-42	4
Fig. 8.1.1	4
Fig. 8.1.2a and 8.1.2b	0
Fig. 8.1.2c	4
Fig. 8.1.2d	1
Fig. 8.1.2e and 8.1.2f	0
Fig. 8.1.3	0
Fig. 8.1.4	4
Fig. 8.1.5 and 8.1.6	0
Fig. 8.1.7	1
Fig. 8.1.8 through 8.1.11	0
Fig. 8.1.12 and 8.1.13	1
Fig. 8.1.14 through 8.1.17	0
Fig. 8.1.18 and 8.1.19	1
Fig. 8.1.20	4
Fig. 8.1.21	1
Fig. 8.1.22a and 8.1.22b	1
Fig. 8.1.23	1
Fig. 8.1.24	0
Fig. 8.1.25 through 8.1.28	1
Fig. 8.1.29a and 8.1.29b	1
Fig. 8.1.30	1
Fig. 8.1.31	4
Fig. 8.1.32	0
Fig. 8.1.33	4
Fig. 8.1.34a	1
Fig. 8.1.34b	Deleted
Fig. 8.1.35 and 8.1.36	0
Fig. 8.1.37	4
8.2-1	4
8.3-1 through 8.3-11	4
Fig. 8.3.1 and 8.3.2a	4
Fig. 8.3.2b through 8.3.2d	0
Fig. 8.3.3	0
Fig. 8.3.4	1
8.4-1 through 8.4-3	4
Fig. 8.4.1a	0
Fig. 8.4.1b	4
8.5-1 through 8.5-5	4
Fig. 8.5.1	0
8.6-1	1
8.II-1	***4A***

LIST OF EFFECTIVE PAGES FOR HISTORM 100 LAR 1014-5, RAI-2

Page(s)	Revision
9.0-1	4
9.1-1 through 9.1-29	4
9.2-1 through 9.2-4	4
9.3-1	1
9.4-1	0
9.II-1	***4A***

Page(s)	Revision
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10.2-1	3
10.3-1 through 10.3-42	4
Fig. 10.3.1a	4
Fig. 10.3.1b	4
Fig. 10.3.1c	4
Fig. 10.3.1d	4
Fig. 10.3.1e	4
10.4-1 through 10.4-5	3
10.5-1	0
10.II-1	***4A***

Page(s)	Revision
11.1-1 through 11.1-18	4
11.2-1 through 11.2-52	***4B***
Fig. 11.2.1	0
Fig. 11.2.2 through 11.2.5	Deleted
Fig. 11.2.6	1
11.3-1	1
11.II-1 through 11.II-14	***4C***

LIST OF EFFECTIVE PAGES FOR HISTORM 100 LAR 1014-5, RAI-2

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12.0-1	0
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12.4-1	0
12.5-1	0
Appendix 12.A Cover	3
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B 3.0-1 through B 3.0-9	0
B 3.1.1-1 through B 3.1.1-7	4
B 3.1.2-1 through B 3.1.2-7	3
B 3.1.3-1 through B 3.1.3-5	3
B 3.1.4-1 through B 3.1.4-3	3
B 3.2.1-1	3
B 3.2.2-1 through B 3.2.2-3	1
B 3.2.3-1	3
B 3.3.1-1 through B 3.3.1-5	3
Appendix 12.B	Deleted
12.II-1	***4A***

Page(s)	Revision
13.0-1 through 13.0-2	4
13.1-1	Deleted
13.2-1	Deleted
13.3-1	Deleted
13.4-1	Deleted
13.5-1	Deleted
13.6-1	4
13.II-1	***4A***

CHAPTER 1[†]: GENERAL DESCRIPTION

1.0 GENERAL INFORMATION

This Final Safety Analysis Report (FSAR) for Holtec International's HI-STORM 100 System is a compilation of information and analyses to support a United States Nuclear Regulatory Commission (NRC) licensing review as a spent nuclear fuel (SNF) dry storage cask under requirements specified in 10CFR72 [1.0.1]. This FSAR describes the basis for NRC approval and issuance of a Certificate of Compliance (C of C) for storage under provisions of 10CFR72, Subpart L, for the HI-STORM 100 System to safely store spent nuclear fuel (SNF) at an Independent Spent Fuel Storage Installation (ISFSI). This report has been prepared in the format and content suggested in NRC Regulatory Guide 3.61 [1.0.2] and NUREG-1536 Standard Review Plan for Dry Cask Storage Systems [1.0.3] to facilitate the NRC review process.

The purpose of this chapter is to provide a general description of the design features and storage capabilities of the HI-STORM 100 System, drawings of the structures, systems, and components important to safety, and the qualifications of the certificate holder. This report is also suitable for incorporation into a site-specific Safety Analysis Report, which may be submitted by an applicant for a site-specific 10 CFR 72 license to store SNF at an ISFSI or a facility similar in objective and scope. Table 1.0.1 contains a listing of the terminology and notation used in this FSAR.

To aid NRC review, additional tables and references have been added to facilitate the location of information requested by NUREG-1536. Table 1.0.2 provides a matrix of the topics in NUREG-1536 and Regulatory Guide 3.61, the corresponding 10CFR72 requirements, and a reference to the applicable FSAR section that addresses each topic.

The HI-STORM 100 FSAR is in full compliance with the intent of all regulatory requirements listed in Section III of each chapter of NUREG-1536. However, an exhaustive review of the provisions in NUREG-1536, particularly Section IV (Acceptance Criteria) and Section V (Review Procedures) has identified certain deviations from a verbatim compliance to all guidance. A list of all such items, along with a discussion of their intent and Holtec International's approach for compliance with the underlying intent is presented in Table 1.0.3 herein. Table 1.0.3 also contains the justification for the alternative method for compliance adopted in this FSAR. The justification may be in the form of a supporting analysis, established industry practice, or other NRC guidance documents. Each chapter in this FSAR provides a clear statement with respect to the extent of compliance to the NUREG-1536 provisions. Chapter 1 is in full compliance with NUREG-1536; no exceptions are taken.

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

The generic design basis and the corresponding safety analysis of the HI-STORM 100 System contained in this FSAR are intended to bound the SNF characteristics, design, conditions, and interfaces that exist in the vast majority of domestic power reactor sites and potential away-from-reactor storage sites in the contiguous United States. This FSAR also provides the basis for component fabrication and acceptance, and the requirements for safe operation and maintenance of the components, consistent with the design basis and safety analysis documented herein. In accordance with 10CFR72, Subpart K, site-specific implementation of the generically certified HI-STORM 100 System requires that the licensee perform a site-specific evaluation, as defined in 10CFR72.212. The HI-STORM 100 System FSAR identifies a limited number of conditions that are necessarily site-specific and are to be addressed in the licensee's 10CFR72.212 evaluation. These include:

- Siting of the ISFSI and design of the storage pad (including the embedment for anchored cask users) and security system. Site-specific demonstration of compliance with regulatory dose limits. Implementation of a site-specific ALARA program.

- An evaluation of site-specific hazards and design conditions that may exist at the ISFSI site or the transfer route between the plant's cask receiving bay and the ISFSI. These include, but are not limited to, explosion and fire hazards, flooding conditions, land slides, and lightning protection.

- Determination that the physical and nucleonic characteristics and the condition of the SNF assemblies to be dry stored meet the fuel acceptance requirements of the Certificate of Compliance.

- An evaluation of interface and design conditions that exist within the plant's fuel building in which canister fuel loading, canister closure, and canister transfer operations are to be conducted in accordance with the applicable 10CFR50 requirements and technical specifications for the plant.

- Detailed site-specific operating, maintenance, and inspection procedures prepared in accordance with the generic procedures and requirements provided in Chapters 8 and 9, and the technical specifications provided in the Certificate of Compliance.

- Performance of pre-operational testing.

- Implementation of a safeguards and accountability program in accordance with 10CFR73. Preparation of a physical security plan in accordance with 10CFR73.55.

- Review of the reactor emergency plan, quality assurance (QA) program, training program, and radiation protection program.

The generic safety analyses contained in the HI-STORM 100 FSAR may be used as input and for guidance by the licensee in performing a 10CFR72.212 evaluation.

Within this report, all figures, tables and references cited are identified by the double decimal system m.n.i, where m is the chapter number, n is the section number, and i is the sequential number. Thus, for example, Figure 1.2.3 is the third figure in Section 1.2 of Chapter 1.

Revisions to this document are made on a section level basis. Complete sections have been replaced if any material in the section changed. The specific changes are noted with revision bars in the right margin. Figures are revised individually. Drawings are controlled separately within the Holtec QA program and have individual revision numbers. Bills-of-Material (BOMs) are considered separate drawings and are not necessarily at the same revision level as the drawing(s) to which they apply. If a drawing or BOM was revised in support of the current FSAR revision, that drawing/BOM is included in Section 1.5 at its latest revision level. Drawings and BOMs appearing in this FSAR may be revised between formal updates to the FSAR. Therefore, the revisions of drawings/BOMs in Section 1.5 may not be current.

The HI-STORM 100 System has been expanded slightly to include options specific for Indian Point Unit 1. The affected components are the MPC enclosure vessel, MPC-32 and MPC-32F, HI-STORM 100S Version B and HI-TRAC 100D Version IP1. Information pertaining to these changes is generally contained in supplements to each chapter identified by a Roman numeral "II" (i.e. Chapter 1 and Supplement 1.II). Certain sections of the main FSAR are also affected and are appropriately modified for continuity with the "II" supplements. Unless superseded or specifically modified by information in the "II" supplements, the information in the main FSAR chapters is applicable to the HI-STORM 100 System at Indian Point Unit 1.

1.0.1 Engineering Change Orders

The changes authorized by the Holtec ECOs (with corresponding 10CFR72.48 evaluations, if applicable) listed in the following table are reflected in Revision 4 of this FSAR.

LIST OF ECO'S AND APPLICABLE 10CFR72.48 EVALUATIONS

Affected Item	ECO Number	72.48 Evaluation or Screening Number
MPC-68/68F/68FF Basket	1021-63	718
MPC-24/24E/24EF Basket	1022-59	718
MPC-32 Basket	1023-32	718
MPC Enclosure Vessel	1021-71	634
HI-STORM Overpack	1024-95, 1024-103, 1024-115	736, 776
HI-TRAC 100 and 100D Transfer Cask	1026-30, 1026-31, 1026-32, 1026-40	670, 760
HI-TRAC 125 and 125D Transfer Cask	-	-
General FSAR Changes	5014-112, 5014-114, 5014-121, 5014-122, 5014-123, 5014-124, 5014-125, 5014-126, 5014-129	763, 751, 756, 762

Table 1.0.1

TERMINOLOGY AND NOTATION

ALARA is an acronym for As Low As Reasonably Achievable.

Boral is a generic term to denote an aluminum-boron carbide cermet manufactured in accordance with U.S. Patent No. 4027377. The individual material supplier may use another trade name to refer to the same product.

Boral^{TM} means Boral manufactured by AAR Advanced Structures.

BWR is an acronym for boiling water reactor.

C.G. is an acronym for center of gravity.

Commercial Spent Fuel or CSF refers to nuclear fuel used to produce energy in a commercial nuclear power plant.

Confinement Boundary means the outline formed by the sealed, cylindrical enclosure of the Multi-Purpose Canister (MPC) shell welded to a solid baseplate, a lid welded around the top circumference of the shell wall, the port cover plates welded to the lid, and the closure ring welded to the lid and MPC shell providing the redundant sealing.

Confinement System means the Multi-Purpose Canister (MPC) which encloses and confines the spent nuclear fuel during storage.

Controlled Area means that area immediately surrounding an ISFSI for which the owner/user exercises authority over its use and within which operations are performed.

Cooling Time (or post-irradiation cooling time) for a spent fuel assembly is the time between reactor shutdown and the time the spent fuel assembly is loaded into the MPC.

DBE means Design Basis Earthquake.

DCSS is an acronym for Dry Cask Storage System.

Damaged Fuel Assembly is a fuel assembly with known or suspected cladding defects, as determined by review of records, greater than pinhole leaks or hairline cracks, empty fuel rod locations that are not replaced 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 (or Canister) means a specially designed enclosure for damaged fuel or fuel debris which permits gaseous and liquid media to escape while minimizing dispersal of gross particulates. The Damaged Fuel Container/Canister (DFC) features a lifting location which is suitable for remote handling of a loaded or unloaded DFC.

Design Heat Load is the computed heat rejection capacity of the HI-STORM system with a certified MPC loaded with CSF stored in *uniform storage* with the ambient at the normal temperature and the peak cladding temperature (PCT) at 400ºC. The Design Heat Load is less than the thermal capacity of the system by a suitable margin that reflects the conservatism in the system thermal analysis.

Design Life is the minimum duration for which the component is engineered to perform its intended function set forth in this FSAR, if operated and maintained in accordance with this FSAR.

Design Report is a document prepared, reviewed and QA validated in accordance with the provisions of 10CFR72 Subpart G. The Design Report shall demonstrate compliance with the requirements set forth in the Design Specification. A Design Report is mandatory for systems, structures, and components designated as Important to Safety. The FSAR serves as the Design Report for the HI-STORM 100 System.

Design Specification is a document prepared in accordance with the quality assurance requirements of 10CFR72 Subpart G to provide a complete set of design criteria and functional requirements for a system, structure, or component, designated as Important to Safety, intended to be used in the operation, implementation, or decommissioning of the HI-STORM 100 System. The FSAR serves as the Design Specification for the HI-STORM 100 System.

Enclosure Vessel (or MPC Enclosure Vessel) means the pressure vessel defined by the cylindrical shell, baseplate, port cover plates, lid, closure ring, and associated welds that provides confinement for the helium gas contained within the MPC. The Enclosure Vessel (EV) and the fuel basket together constitute the multi-purpose canister.

Fracture Toughness is a property which is a measure of the ability of a material to limit crack propagation under a suddenly applied load.

FSAR is an acronym for Final Safety Analysis Report (10CFR72).

Fuel Basket means a honeycombed structural weldment with square openings which can accept a fuel assembly of the type for which it is designed.

Fuel Debris refers to ruptured fuel rods, severed rods, loose fuel pellets, or fuel assemblies with known or suspected defects which cannot be handled by normal means due to fuel cladding damage.

High Burnup Fuel, or HBF is a commercial spent fuel assembly with an average burnup greater than 45,000 MWD/MTU.

HI-TRAC transfer cask or HI-TRAC means the transfer cask used to house the MPC during MPC fuel loading, unloading, drying, sealing, and on-site transfer operations to a HI-STORM storage overpack or HI-STAR storage/transportation overpack. The HI-TRAC shields the loaded MPC allowing loading operations to be performed while limiting radiation exposure to personnel. HI-TRAC is an acronym for **H**oltec **I**nternational **Tra**nsfer **C**ask. In this FSAR there are several HI-TRAC transfer casks, the 125-ton standard design HI-TRAC (HI-TRAC-125), the 125-ton dual-purpose lid design (HI-TRAC 125D), the 100-ton HI-TRAC (HI-TRAC-100), ~~and~~ the 100-ton dual purpose lid design (HI-TRAC 100D*), and the 75-ton dual purpose lid design for Indian Point 1 (HI-TRAC 100D Version IP1)*. The 100-ton HI-TRAC is provided for use at sites with a maximum crane capacity of less than 125 tons. The term HI-TRAC is used as a generic term to refer to all HI-TRAC transfer cask designs, unless the discussion requires distinguishing among the designs. The HI-TRAC is equipped with a pair of lifting trunnions and the HI-TRAC 100 and HI-TRAC 125 designs also include pocket trunnions. The trunnions are used to lift and downend/upend the HI-TRAC with a loaded MPC.

HI-STORM overpack or storage overpack means the cask that receives and contains the sealed multi-purpose canisters containing spent nuclear fuel. It provides the gamma and neutron shielding, ventilation passages, missile protection, and protection against natural phenomena and accidents for the MPC. The term "overpack" as used in this FSAR refers to all overpack designs, including the standard design (HI-STORM 100) and two alternate designs (HI-STORM 100S and HI-STORM 100S Version B). The term "overpack" also applies to those overpacks designed for high seismic deployment (HI-STORM 100A or HI-STORM 100SA), unless otherwise clarified.

HI-STORM 100 System consists of any loaded MPC model placed within any design variant of the HI-STORM overpack.

Holtite*™* is the trade name for all present and future neutron shielding materials formulated under Holtec International's R&D program dedicated to developing shielding materials for application in dry storage and transport systems. The Holtite development program is an ongoing experimentation effort to identify neutron shielding materials with enhanced shielding and temperature tolerance characteristics. Holtite-A™ is the first and only shielding material qualified under the Holtite R&D program. As such, the terms Holtite and Holtite-A may be used interchangeably throughout this FSAR.

Holtite™-A is a trademarked Holtec International neutron shield material.

Important to Safety (ITS) means a function or condition required to store spent nuclear fuel safely; to prevent damage to spent nuclear fuel during handling and storage, and to provide reasonable assurance that spent nuclear fuel can be received, handled, packaged, stored, and retrieved without undue risk to the health and safety of the public.

Independent Spent Fuel Storage Installation (ISFSI) means a facility designed, constructed, and licensed for the interim storage of spent nuclear fuel and other radioactive materials associated with spent fuel storage in accordance with 10CFR72.

Intact Fuel Assembly is defined as a fuel assembly without known or suspected cladding defects greater than pinhole leaks and hairline cracks, and which can be handled by normal means. 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).

License Life means the duration for which the system is authorized by virtue of its certification by the U.S. NRC.

Long-term Storage means the time beginning after on-site handling is complete and the loaded overpack is at rest in its designated storage location on the ISFSI pad and lasting up to the end of the licensed life of the HI-STORM 100 System (20 years).

Lowest Service Temperature (LST) is the minimum metal temperature of a part for the specified service condition.

Maximum Reactivity means the highest possible k-effective including bias, uncertainties, and calculational statistics evaluated for the worst-case combination of fuel basket manufacturing tolerances.

METAMIC® is a trade name for an aluminum/boron carbide composite neutron absorber material qualified for use in the MPCs.

METCON™ is a trade name for the HI-STORM overpack. The trademark is derived from the **met**al-**con**crete composition of the HI-STORM overpack.

MGDS is an acronym for Mined Geological Disposal System.

Minimum Enrichment is the minimum assembly average enrichment. Natural uranium blankets are not considered in determining minimum enrichment.

Moderate Burnup Fuel, or MBF is a commercial spent fuel assembly with an average burnup less than or equal to 45,000 MWD/MTU.

Multi-Purpose Canister (MPC) means the sealed canister consisting of a honeycombed fuel basket for spent nuclear fuel storage, contained in a cylindrical canister shell (the MPC Enclosure Vessel). There are different MPCs with different fuel basket geometries for storing PWR or BWR fuel, but all MPCs have identical exterior ~~dimensions~~*diameters*. The MPC is the confinement boundary for storage conditions.

NDT is an acronym for Nil Ductility Transition Temperature, which is defined as the temperature at which the fracture stress in a material with a small flaw is equal to the yield stress in the same material if it had no flaws.

Table 1.0.1 (continued)

TERMINOLOGY AND NOTATION

Neutron Absorber Material is a generic term used in this FSAR to indicate any neutron absorber material qualified for use in the HI-STORM 100 System MPCs.

Neutron Shielding means a material used to thermalize and capture neutrons emanating from the radioactive spent nuclear fuel.

Non-Fuel Hardware is defined as Burnable Poison Rod Assemblies (BPRAs), Thimble Plug Devices (TPDs), Control Rod Assemblies (CRAs), Axial Power Shaping Rods (APSRs), Wet Annular Burnable Absorbers (WABAs), Rod Cluster Control Assemblies (RCCAs), water displacement guide tube plugs, orifice rod assemblies, and vibration suppressor inserts.

Planar-Average Initial Enrichment is the average of the distributed fuel rod initial enrichments within a given axial plane of the assembly lattice.

Plain Concrete is concrete that is unreinforced and is of density specified in this FSAR .

Post-Core Decay Time (PCDT) is synonymous with cooling time.

PWR is an acronym for pressurized water reactor.

Reactivity is used synonymously with effective neutron multiplication factor or k-effective.

Regionalized Fuel Loading is a term used to describe an optional fuel loading strategy used in lieu of uniform fuel loading. Regionalized fuel loading allows high heat emitting fuel assemblies to be stored in fuel storage locations in the center of the fuel basket provided lower heat emitting fuel assemblies are stored in the peripheral fuel storage locations. Users choosing regionalized fuel loading must also consider other restrictions in the CoC such as those for non-fuel hardware and damaged fuel containers. Regionalized fuel loading does not apply to the MPC-68F model.

SAR is an acronym for Safety Analysis Report (10CFR71).

Service Life means the duration for which the component is reasonably expected to perform its intended function, if operated and maintained in accordance with the provisions of this FSAR. Service Life may be much longer than the Design Life because of the conservatism inherent in the codes, standards, and procedures used to design, fabricate, operate, and maintain the component.

Short-term Operations means those normal operational evolutions necessary to support fuel loading or fuel unloading operations. These include, but are not limited to MPC cavity drying, helium backfill, MPC transfer, and onsite handling of a loaded HI-TRAC transfer cask.

Table 1.0.1 (continued)

TERMINOLOGY AND NOTATION

Single Failure Proof means that the handling system is designed so that all directly loaded tension and compression members are engineered to satisfy the enhanced safety criteria of Paragraphs 5.1.6(1)(a) and (b) of NUREG-0612.

SNF is an acronym for spent nuclear fuel.

SSC is an acronym for Structures, Systems and Components.

STP is Standard Temperature and Pressure conditions.

Thermal Capacity of the HI-STORM system is defined as the amount of heat the storage system, containing an MPC loaded with CSF stored in *uniform storage,* will actually reject with the ambient environment at the normal temperature and the peak fuel cladding temperature (PCT) at 400ºC.

Thermosiphon is the term used to describe the buoyancy-driven natural convection circulation of helium within the MPC fuel basket maximum heat load during short-term operating conditions up to which no time limit or other restriction is imposed on the operating condition.

Uniform Fuel Loading is a fuel loading strategy where any authorized fuel assembly may be stored in any fuel storage location, subject to other restrictions in the CoC, such as those applicable to non-fuel hardware, and damaged fuel containers.

ZPA is an acronym for zero period acceleration.

ZR means any zirconium-based fuel cladding material authorized for use in a commercial nuclear power plant reactor. Any reference to Zircaloy fuel cladding in this FSAR applies to any zirconium-based fuel cladding material.

Table 1.0.2

HI-STORM 100 SYSTEM FSAR REGULATORY COMPLIANCE
CROSS REFERENCE MATRIX

Regulatory Guide 3.61 Section and Content	Associated NUREG-1536 Review Criteria	Applicable 10CFR72 or 10CFR20 Requirement	HI-STORM FSAR
1. General Description			
1.1 Introduction	1.III.1 General Description & Operational Features	10CFR72.24(b)	1.1
1.2 General Description	1.III.1 General Description & Operational Features	10CFR72.24(b)	1.2
1.2.1 Cask Character-istics	1.III.1 General Description & Operational Features	10CFR72.24(b)	1.2.1
1.2.2 Operational Features	1.III.1 General Description & Operational Features	10CFR72.24(b)	1.2.2
1.2.3 Cask Contents	1.III.3 DCSS Contents	10CFR72.2(a)(1) 10CFR72.236(a)	1.2.3
1.3 Identification of Agents & Contractors	1.III.4 Qualification of the Applicant	10CFR72.24(j) 10CFR72.28(a)	1.3
1.4 Generic Cask Arrays	1.III.1 General Description & Operational Features	10CFR72.24(c)(3)	1.4
1.5 Supplemental Data	1.III.2 Drawings	10CFR72.24(c)(3)	1.5
NA	1.III.6 Consideration of Transport Requirements	10CFR72.230(b) 10CFR72.236(m)	1.1
NA	1.III.5 Quality Assurance	10CFR72.24(n)	1.3
2. Principal Design Criteria			
2.1 Spent Fuel To Be Stored	2.III.2.a Spent Fuel Specifications	10CFR72.2(a)(1) 10CFR72.236(a)	2.1
2.2 Design Criteria for Environmental Conditions and Natural Phenomena	2.III.2.b External Conditions, 2.III.3.b Structural, 2.III.3.c Thermal	10CFR72.122(b)	2.2
		10CFR72.122(c)	2.2.3.3, 2.2.3.10
		10CFR72.122(b)(1)	2.2
		10CFR72.122(b)(2)	2.2.3.11
		10CFR72.122(h)(1)	2.0
2.2.1 Tornado and Wind Loading	2.III.2.b External Conditions	10CFR72.122(b)(2)	2.2.3.5

Table 1.0.2 (continued)

HI-STORM 100 SYSTEM FSAR REGULATORY COMPLIANCE
CROSS REFERENCE MATRIX

Regulatory Guide 3.61 Section and Content	**Associated NUREG-1536 Review Criteria**	**Applicable 10CFR72 or 10CFR20 Requirement**	**HI-STORM FSAR**
2.2.2 Water Level (Flood)	2.III.2.b External Conditions 2.III.3.b Structural	10CFR72.122(b)(2)	2.2.3.6
2.2.3 Seismic	2.III.3.b Structural	10CFR72.102(f) 10CFR72.122(b)(2)	2.2.3.7
2.2.4 Snow and Ice	2.III.2.b External Conditions 2.III.3.b Structural	10CFR72.122(b)	2.2.1.6
2.2.5 Combined Load	2.III.3.b Structural	10CFR72.24(d) 10CFR72.122(b)(2)(ii)	2.2.7
NA	2.III.1 Structures, Systems, and Components Important to Safety	10CFR72.122(a) 10CFR72.24(c)(3)	2.2.4
NA	2.III.2 Design Criteria for Safety Protection Systems	10CFR72.236(g) 10CFR72.24(c)(1) 10CFR72.24(c)(2) 10CFR72.24(c)(4) 10CFR72.120(a) 10CFR72.236(b)	2.0, 2.2
NA	2.III.3.c Thermal	10CFR72.128(a)(4)	2.3.2.2, 4.0
NA	2.III.3f Operating Procedures	10CFR72.24(f) 10CFR72.128(a)(5)	10.0, 8.0
		10CFR72.236(h)	8.0
		10CFR72.24(1)(2)	1.2.1, 1.2.2
		10CFR72.236(1)	2.3.2.1
		10CFR72.24(e) 10CFR72.104(b)	10.0, 8.0
	2.III.3.g Acceptance Tests & Maintenance	10CFR72.122(1) 10CFR72.236(g) 10CFR72.122(f) 10CFR72.128(a)(1)	9.0
2.3 Safety Protection Systems	--	--	2.3

Table 1.0.2 (continued)

HI-STORM 100 SYSTEM FSAR REGULATORY COMPLIANCE
CROSS REFERENCE MATRIX

Regulatory Guide 3.61 Section and Content	Associated NUREG-1536 Review Criteria	Applicable 10CFR72 or 10CFR20 Requirement	HI-STORM FSAR
2.3.1 General	--	--	2.3
2.3.2 Protection by Multiple Confinement Barriers and Systems	2.III.3.b Structural	10CFR72.236(1)	2.3.2.1
	2.III.3.c Thermal	10CFR72.236(f)	2.3.2.2
	2.III.3.d Shielding/ Confinement/ Radiation Protection	10CFR72.126(a) 10CFR72.128(a)(2)	2.3.5.2
		10CFR72.128(a)(3)	2.3.2.1
		10CFR72.236(d)	2.3.2.1, 2.3.5.2
		10CFR72.236(e)	2.3.2.1
2.3.3 Protection by Equipment & Instrument Selection 2.3.4 Nuclear Criticality Safety	2.III.3.d Shielding/ Confinement/ Radiation Protection 2.III.3.e Criticality	10CFR72.122(h)(4) 10CFR72.122(i) 10CFR72.128(a)(1) 10CFR72.124(a) 10CFR72.236(c) 10CFR72.124(b)	2.3.5 2.3.4, 6.0
2.3.5 Radiological Protection	2.III.3.d Shielding/ Confinement/ Radiation Protection	10CFR72.24(d) 10CFR72.104(a) 10CFR72.236(d)	10.4.1
		10CFR72.24(d) 10CFR72.106(b) 10CFR72.236(d)	10.4.2
		10CFR72.24(m)	2.3.2.1
2.3.6 Fire and Explosion Protection	2.III.3.b Structural	10CFR72.122(c)	2.3.6, 2.2.3.10
2.4 Decommissioning Considerations	2.III.3.h Decommissioning	10CFR72.24(f) 10CFR72.130 10CFR72.236(h)	2.4
	14.III.1 Design	10CFR72.130	2.4
	14.III.2 Cask Decontamination	10CFR72.236(i)	2.4

Table 1.0.2 (continued)

HI-STORM 100 SYSTEM FSAR REGULATORY COMPLIANCE CROSS REFERENCE MATRIX

Regulatory Guide 3.61 Section and Content	Associated NUREG-1536 Review Criteria	Applicable 10CFR72 or 10CFR20 Requirement	HI-STORM FSAR
	14.III.3 Financial Assurance & Record Keeping	10CFR72.30	(1)
	14.III.4 License Termination	10CFR72.54	(1)
3. Structural Evaluation			
3.1 Structural Design	3.III.1 SSC Important to Safety	10CFR72.24(c)(3) 10CFR72.24(c)(4)	3.1
	3.III.6 Concrete Structures	10CFR72.24(c)	3.1
3.2 Weights and Centers of Gravity	3.V.1.b.2 Structural Design Features	--	3.2
3.3 Mechanical Properties of Materials	3.V.1.c Structural Materials 3.V.2.c Structural Materials	10CFR72.24(c)(3)	3.3
NA	3.III.2 Radiation Shielding, Confinement, and Subcriticality	10CFR72.24(d) 10CFR72.124(a) 10CFR72.236(c) 10CFR72.236(d) 10CFR72.236(1)	3.4.4.3 3.4.7.3 3.4.10
NA	3.III.3 Ready Retrieval	10CFR72.122(f) 10CFR72.122(h) 10CFR72.122(l)	3.4.4.3
NA	3.III.4 Design-Basis Earthquake	10CFR72.24(c) 10CFR72.102(f)	3.4.7
NA	3.III.5 20 Year Minimum Design Length	10CFR72.24(c) 10CFR72.236(g)	3.4.11 3.4.12
3.4 General Standards for Casks	--	--	3.4
3.4.1 Chemical and Galvanic Reactions	3.V.1.b.2 Structural Design Features	--	3.4.1
3.4.2 Positive Closure	--	--	3.4.2
3.4.3 Lifting Devices	3.V.1.ii(4)(a) Trunnions --	--	3.4.3

HI-STORM 100 SYSTEM FSAR REGULATORY COMPLIANCE
CROSS REFERENCE MATRIX

Regulatory Guide 3.61 Section and Content	Associated NUREG-1536 Review Criteria	Applicable 10CFR72 or 10CFR20 Requirement	HI-STORM FSAR
3.4.4 Heat	3.V.1.d Structural Analysis	10CFR72.24(d) 10CFR72.122(b)	3.4.4
3.4.5 Cold	3.V.1.d Structural Analysis	10CFR72.24(d) 10CFR72.122(b)	3.4.5
3.5 Fuel Rods	--	10CFR72.122(h)(1)	3.5
4. Thermal Evaluation			
4.1 Discussion	4.III Regulatory Requirements	10CFR72.24(c)(3) 10CFR72.128(a)(4) 10CFR72.236(f) 10CFR72.236(h)	4.1
4.2 Summary of Thermal Properties of Materials	4.V.4.b Material Properties	--	4.2
4.3 Specifications for Components	4.IV Acceptance Criteria ISG-11, Revision 3	10CFR72.122(h)(1)	4.3
4.4 Thermal Evaluation for Normal Conditions of Storage	4.IV Acceptance Criteria ISG-11, Revision 3	10CFR72.24(d) 10CFR72.236(g)	4.4, 4.5
NA	4.IV Acceptance Criteria	10CFR72.24(d) 10CFR72.122(c)	11.1, 11.2
4.5 Supplemental Data	4.V.6 Supplemental Info.	--	--
5. Shielding Evaluation			
5.1 Discussion and Results	--	10CFR72.104(a) 10CFR72.106(b)	5.1
5.2 Source Specification	5.V.2 Radiation Source Definition	--	5.2
5.2.1 Gamma Source	5.V.2.a Gamma Source	--	5.2.1, 5.2.3

Table 1.0.2 (continued)

HI-STORM 100 SYSTEM FSAR REGULATORY COMPLIANCE
CROSS REFERENCE MATRIX

Regulatory Guide 3.61 Section and Content	Associated NUREG-1536 Review Criteria	Applicable 10CFR72 or 10CFR20 Requirement	HI-STORM FSAR
5.2.2 Neutron Source	5.V.2.b Neutron Source	--	5.2.2, 5.2.3
5.3 Model Specification	5.V.3 Shielding Model Specification	--	5.3
5.3.1 Description of the Radial and Axial Shielding Configura-tions	5.V.3.a Configuration of the Shielding and Source	10CFR72.24(c)(3)	5.3.1
5.3.2 Shield Regional Densities	5.V.3.b Material Properties	10CFR72.24(c)(3)	5.3.2
5.4 Shielding Evaluation	5.V.4 Shielding Analysis	10CFR72.24(d) 10CFR72.104(a) 10CFR72.106(b) 10CFR72.128(a)(2) 10CFR72.236(d)	5.4
5.5 Supplemental Data	5.V.5 Supplemental Info.	--	Appendices 5.A, 5.B, and 5.C
6. Criticality Evaluation			
6.1 Discussion and Results	--	--	6.1
6.2 Spent Fuel Loading	6.V.2 Fuel Specification	--	6.1, 6.2
6.3 Model Specifications	6.V.3 Model Specification	--	6.3
6.3.1 Description of Calcula-tional Model	6.V.3.a Configuration	-- 10CFR72.124(b) 10CFR72.24(c)(3)	6.3.1

Table 1.0.2 (continued)

HI-STORM 100 SYSTEM FSAR REGULATORY COMPLIANCE
CROSS REFERENCE MATRIX

Regulatory Guide 3.61 Section and Content	Associated NUREG-1536 Review Criteria	Applicable 10CFR72 or 10CFR20 Requirement	HI-STORM FSAR
6.3.2 Cask Regional Densities	6.V.3.b Material Properties	10CFR72.24(c)(3) 10CFR72.124(b) 10CFR72.236(g)	6.3.2
6.4 Criticality Calculations	6.V.4 Criticality Analysis	10CFR72.124	6.4
6.4.1 Calculational or Experimental Method	6.V.4.a Computer Programs and 6.V.4.b Multiplication Factor	10CFR72.124	6.4.1
6.4.2 Fuel Loading or Other Contents Loading Optimization	6.V.3.a Configuration	--	6.4.2, 6.3.3
6.4.3 Criticality Results	6.IV Acceptance Criteria	10CFR72.24(d) 10CFR72.124 10CFR72.236(c)	6.1, 6.2, 6.3.1, 6.3.2
6.5 Critical Benchmark Experiments	6.V.4.c Benchmark Comparisons	--	6.5, Appendix 6.A, 6.4.3
6.6 Supplemental Data	6.V.5 Supplemental Info.	--	Appendices 6.B,6.C, and 6.D
7. Confinement			
7.1 Confinement Boundary	7.III.1 Description of Structures, Systems and Components Important to Safety ISG-18	10CFR72.24(c)(3) 10CFR72.24(1)	7.0, 7.1
7.1.1 Confinement Vessel	7.III.2 Protection of Spent Fuel Cladding	10CFR72.122(h)(l)	7.1, 7.1.1
7.1.2 Confinement Penetrations	--	--	7.1.2
7.1.3 Seals and Welds	--	--	7.1.3

HI-STORM 100 SYSTEM FSAR REGULATORY COMPLIANCE
CROSS REFERENCE MATRIX

Regulatory Guide 3.61 Section and Content	Associated NUREG-1536 Review Criteria	Applicable 10CFR72 or 10CFR20 Requirement	HI-STORM FSAR
7.1.4 Closure	7.III.3 Redundant Sealing	10CFR72.236(e)	7.1.1, 7.1.4
7.2 Requirements for Normal Conditions of Storage	7.III.7 Evaluation of Confinement System ISG-18	10CFR72.24(d) 10CFR72.236(1)	7.1
7.2.1 Release of Radioactive Material	7.III.6 Release of Nuclides to the Environment	10CFR72.24(1)(1)	7. 1
	7.III.4 Monitoring of Confinement System	10CFR72.122(h)(4) 10CFR72.128(a)(l)	7.1.4
	7.III.5 Instrumentation	10CFR72.24(l) 10CFR72.122(i)	7.1.4
	7.III.8 Annual Dose ISG-18	10CFR72.104(a)	7.1
7.2.2 Pressurization of Confinement Vessel	--	--	7.1
7.3 Confinement Requirements for Hypothetical Accident Conditions	7.III.7 Evaluation of Confinement System ISG-18	10CFR72.24(d) 10CFR72.122(b) 10CFR72.236(l)	7.1
7.3.1 Fission Gas Products	--	--	7.1
7.3.2 Release of Contents	ISG-18	--	7.1
NA	--	10CFR72.106(b)	7.1
7.4 Supplemental Data	7.V Supplemental Info.	--	--
8. Operating Procedures			
8.1 Procedures for Loading the Cask	8.III.1 Develop Operating Procedures	10CFR72.40(a)(5)	8.1 to 8.5
	8.III.2 Operational Restrictions for ALARA	10CFR72.24(e) 10CFR72.104(b)	8.1.5
	8.III.3 Radioactive Effluent Control	10CFR72.24(1)(2)	8.1.5, 8.5.2

HI-STORM 100 SYSTEM FSAR REGULATORY COMPLIANCE
CROSS REFERENCE MATRIX

Regulatory Guide 3.61 Section and Content	Associated NUREG-1536 Review Criteria	Applicable 10CFR72 or 10CFR20 Requirement	HI-STORM FSAR
	8.III.4 Written Procedures	10CFR72.212(b)(9)	8.0
	8.III.5 Establish Written Procedures and Tests	10CFR72.234(f)	8.0 Introduction
	8.III.6 Wet or Dry Loading and Unloading Compatibility	10CFR72.236(h)	8.0 Introduction
	8.III.7 Cask Design to Facilitate Decon	10CFR72.236(i)	8.1, 8.3
8.2 Procedures for Unloading the Cask	8.III.1 Develop Operating Procedures	10CFR72.40(a)(5)	8.3
	8.III.2 Operational Restrictions for ALARA	10CFR72.24(e) 10CFR72.104(b)	8.3
	8.III.3 Radioactive Effluent Control	10CFR72.24(1)(2)	8.3.3
	8.III.4 Written Procedures	10CFR72.212(b)(9)	8.0
	8.III.5 Establish Written Procedures and Tests	10CFR72.234(f)	8.0
	8.III.6 Wet or Dry Loading and Unloading Compatibility	10CFR72.236(h)	8.0
	8.III.8 Ready Retrieval	10CFR72.122(1)	8.3
8.3 Preparation of the Cask	--	--	8.3.2
8.4 Supplemental Data	--	--	Tables 8.1.1 to 8.1.10
NA	8.III.9 Design to Minimize Radwaste	10CFR72.24(f) 10CFR72.128(a)(5)	8.1, 8.3
	8.III.10 SSCs Permit Inspection, Maintenance, and Testing	10CFR72.122(f)	Table 8.1.6
9. Acceptance Criteria and Maintenance Program			
9.1 Acceptance Criteria	9.III.1.a Preoperational Testing & Initial Operations	10CFR72.24(p)	8.1, 9.1

Table 1.0.2 (continued)

HI-STORM 100 SYSTEM FSAR REGULATORY COMPLIANCE
CROSS REFERENCE MATRIX

Regulatory Guide 3.61 Section and Content	Associated NUREG-1536 Review Criteria	Applicable 10CFR72 or 10CFR20 Requirement	HI-STORM FSAR
	9.III.1.c SSCs Tested and Maintained to Appropriate Quality Standards	10CFR72.24(c) 10CFR72.122(a)	9.1
	9.III.1.d Test Program	10CFR72.162	9.1
	9.III.1.e Appropriate Tests	10CFR72.236(1)	9.1
	9.III.1.f Inspection for Cracks, Pinholes, Voids and Defects	10CFR72.236(j)	9.1
	9.III.1.g Provisions that Permit Commission Tests	10CFR72.232(b)	9.1[(2)]
9.2 Maintenance Program	9.III.1.b Maintenance	10CFR72.236(g)	9.2
	9.III.1.c SSCs Tested and Maintained to Appropriate Quality Standards	10CFR72.122(f) 10CFR72.128(a)(1)	9.2
	9.III.1.h Records of Maintenance	10CFR72.212(b)(8)	9.2
NA	9.III.2 Resolution of Issues Concerning Adequacy of Reliability	10CFR72.24(i)	[(3)]
	9.III.1.d Submit Pre-Op Test Results to NRC	10CFR72.82(e)	[(4)]
	9.III.1.i Casks Conspicuously and Durably Marked 9.III.3 Cask Identification	10CFR72.236(k)	9.1.7, 9.1.1.(12)
10. Radiation Protection			
10.1 Ensuring that Occupational Exposures are as Low as Reasonably Achievable (ALARA)	10.III.4 ALARA	10CFR20.1101 10CFR72.24(e) 10CFR72.104(b) 10CFR72.126(a)	10.1
10.2 Radiation Protection Design Features	10.V.1.b Design Features	10CFR72.126(a)(6)	10.2

HI-STORM 100 SYSTEM FSAR REGULATORY COMPLIANCE
CROSS REFERENCE MATRIX

Regulatory Guide 3.61 Section and Content	Associated NUREG-1536 Review Criteria	Applicable 10CFR72 or 10CFR20 Requirement	HI-STORM FSAR
10.3 Estimated Onsite Collective Dose Assessment	10.III.2 Occupational Exposures	10CFR20.1201 10CFR20.1207 10CFR20.1208 10CFR20.1301	10.3
N/A	10.III.3 Public Exposure	10CFR72.104 10CFR72.106	10.4
	10.III.1 Effluents and Direct Radiation	10CFR72.104	
11. Accident Analyses			
11.1 Off-Normal Operations	11.III.2 Meet Dose Limits for Anticipated Events	10CFR72.24(d) 10CFR72.104(a) 10CFR72.236(d)	11.1
	11.III.4 Maintain Subcritical Condition	10CFR72.124(a) 10CFR72.236(c)	11.1
	11.III.7 Instrumentation and Control for Off-Normal Condition	10CFR72.122(i)	11.1
11.2 Accidents	11.III.1 SSCs Important to Safety Designed for Accidents	10CFR72.24(d)(2) 10CFR72.122b(2) 10CFR72.122b(3) 10CFR72.122(d) 10CFR72.122(g)	11.2
	11.III.5 Maintain Confinement for Accident	10CFR72.236(1)	11.2
	11.III.4 Maintain Subcritical Condition	10CFR72.124(a) 10CFR72.236(c)	11.2, 6.0
	11.III.3 Meet Dose Limits for Accidents	10CFR72.24(d)(2) 10CFR72.24(m) 10CFR72.106(b)	11.2, 5.1.2, 7.3

Table 1.0.2 (continued)

HI-STORM 100 SYSTEM FSAR REGULATORY COMPLIANCE
CROSS REFERENCE MATRIX

Regulatory Guide 3.61 Section and Content	Associated NUREG-1536 Review Criteria	Applicable 10CFR72 or 10CFR20 Requirement	HI-STORM FSAR
	11.III.6 Retrieval	10CFR72.122(l)	8.3
	11.III.7 Instrumentation and Control for Accident Conditions	10CFR72.122(i)	(5)
NA	11.III.8 Confinement Monitoring	10CFR72.122h(4)	7.1.4
12. Operating Controls and Limits			
12.1 Proposed Operating Controls and Limits	--	10CFR72.44(c)	12.0
	12.III.1.e Administrative Controls	10CFR72.44(c)(5)	12.0
12.2 Development of Operating Controls and Limits	12.III.1 General Requirement for Technical Specifications	10CFR72.24(g) 10CFR72.26 10CFR72.44(c) 10CFR72 Subpart E 10CFR72 Subpart F	12.0
12.2.1 Functional and Operating Limits, Monitoring Instruments, and Limiting Control Settings	12.III.1.a Functional/ Operating Units, Monitoring Instruments and Limiting Controls	10CFR72.44(c)(1)	Appendix 12.A
12.2.2 Limiting Conditions for Operation	12.III.1.b Limiting Controls	10CFR72.44(c)(2)	Appendix 12.A
	12.III.2.a Type of Spent Fuel	10CFR72.236(a)	Appendix 12.A
	12.III.2.b Enrichment		
	12.III.2.c Burnup		
	12.III.2.d Minimum Acceptance Cooling Time		
	12.III.2.f Maximum Spent Fuel Loading Limit		
	12.III.2g Weights and Dimensions		
	12.III.2.h Condition of Spent Fuel		
	12.III.2e Maximum Heat Dissipation	10CFR72.236(a)	Appendix 12.A

Table 1.0.2 (continued)

HI-STORM 100 SYSTEM FSAR REGULATORY COMPLIANCE
CROSS REFERENCE MATRIX

Regulatory Guide 3.61 Section and Content	**Associated NUREG-1536 Review Criteria**	**Applicable 10CFR72 or 10CFR20 Requirement**	**HI-STORM FSAR**
	12.III.2.i Inerting Atmosphere Requirements	10CFR72.236(a)	Appendix 12.A
12.2.3 Surveillance Specifications	12.III.1.c Surveillance Requirements	10CFR72.44(c)(3)	Chapter 12
12.2.4 Design Features	12.III.1.d Design Features	10CFR72.44(c)(4)	Chapter 12
12.2.4 Suggested Format for Operating Controls and Limits	--	--	Appendix 12.A
NA	12.III.2 SCC Design Bases and Criteria	10CFR72.236(b)	2.0
NA	12.III.2 Criticality Control	10CFR72.236(c)	2.3.4, 6.0
NA	12.III.2 Shielding and Confinement	10CFR20 10CFR72.236(d)	2.3.5, 7.0, 5.0, 10.0
NA	12.III.2 Redundant Sealing	10CFR72.236(e)	7.1, 2.3.2
NA	12.III.2 Passive Heat Removal	10CFR72.236(f)	2.3.2.2, 4.0
NA	12.III.2 20 Year Storage and Maintenance	10CFR72.236(g)	1.2.1.5, 9.0, 3.4.10, 3.4.11
NA	12.III.2 Decontamination	10CFR72.236(i)	8.0, 10.1
NA	12.III.2 Wet or Dry Loading	10CFR72.236(h)	8.0
NA	12.III.2 Confinement Effectiveness	10CFR72.236(j)	9.0
NA	12.III.2 Evaluation for Confinement	10CFR72.236(l)	7.1, 7.2, 9.0
13. Quality Assurance			
13.1 Quality Assurance	13.III Regulatory Requirements	10CFR72.24(n) 10CFR72.140(d)	13.0
	13.IV Acceptance Criteria	10CFR72, Subpart G	

Table 1.0.2 (continued)

HI-STORM 100 SYSTEM FSAR REGULATORY COMPLIANCE
CROSS REFERENCE MATRIX

Notes:

(1) The stated requirement is the responsibility of the licensee (i.e., utility) as part of the ISFSI pad and is therefore not addressed in this application.

(2) It is assumed that approval of the FSAR by the NRC is the basis for the Commission's acceptance of the tests defined in Chapter 9.

(3) Not applicable to HI-STORM 100 System. The functional adequacy of all important to safety components is demonstrated by analyses.

(4) The stated requirement is the responsibility of licensee (i.e., utility) as part of the ISFSI and is therefore not addressed in this application.

(5) The stated requirement is not applicable to the HI-STORM 100 System. No monitoring is required for accident conditions.

"—" There is no corresponding NUREG-1536 criteria, no applicable 10CFR72 or 10CFR20 regulatory requirement, or the item is not addressed in the FSAR.

"NA" There is no Regulatory Guide 3.61 section that corresponds to the NUREG-1536, 10CFR72, or 10CFR20 requirement being addressed.

Table 1.0.3

HI-STORM 100 SYSTEM FSAR CLARIFICATIONS AND EXCEPTIONS TO NUREG-1536

NUREG-1536 Requirement	Alternate Method to Meet NUREG-1536 Intent	Justification
2.V.2.(b)(1) "The NRC accepts as the maximum and minimum "normal" temperatures the highest and lowest ambient temperatures recorded in each year, averaged over the years of record."	Exception: Section 2.2.1.4 for environmental temperatures utilizes an upper bounding value of 80^{o}F on the annual average ambient temperatures for the United States.	The 80^{o}F temperature set forth in Table 2.2.2 is greater than the annual average ambient temperature at any location in the continental United States. Inasmuch as the primary effect of the environmental temperature is on the computed fuel cladding temperature to establish long-term fuel cladding integrity, the annual average ambient temperature for each ISFSI site should be below 80^{o}F. The large thermal inertia of the HI-STORM 100 System ensures that the daily fluctuations in temperatures do not affect the temperatures of the system. Additionally, the 80^{o}F ambient temperature is combined with insolation in accordance with 10CFR71.71 averaged over 24 hours.
2.V.2.(b)(3)(f) "10CFR Part 72 identifies several other natural phenomena events (including seiche, tsunami, and hurricane) that should be addressed for spent fuel storage."	Clarification: A site-specific safety analysis of the effects of seiche, tsunami, and hurricane on the HI-STORM 100 System must be performed prior to use if these events are applicable to the site.	In accordance with NUREG-1536, 2.V.(b)(3)(f), if seiche, tsunami, and hurricane are not addressed in the SAR and they prove to be applicable to the site, a safety analysis is required prior to approval for use of the DCSS under either a site specific, or general license.

Table 1.0.3 (continued)

HI-STORM 100 SYSTEM FSAR CLARIFICATIONS AND EXCEPTIONS TO NUREG-1536

NUREG-1536 Requirement	Alternate Method to Meet NUREG-1536 Intent	Justification
3.V.1.d.i.(2)(a), page 3-11, "Drops with the axis generally vertical should be analyzed for both the conditions of a flush impact and an initial impact at a corner of the cask..."	Clarification: As stated in NUREG-1536, 3.V.(d), page 3-11, "Generally, applicants establish the design basis in terms of the maximum height to which the cask is lifted outside the spent fuel building, or the maximum deceleration that the cask could experience in a drop." The maximum deceleration for a corner drop is specified as 45g's for the HI-STORM overpack. No carry height limit is specified for the corner drop.	In Chapter 3, the MPC and HI-STORM overpack are evaluated under a 45g radial loading. A 45g axial loading on the MPC is bounded by the analysis presented in the HI-STAR FSAR, Docket 72-1008, under a 60g loading, and is not repeated in this FSAR. In Chapter 3, the HI-STORM overpack is evaluated under a 45g axial loading. Therefore, the HI-STORM overpack and MPC are qualified for a 45g loading as a result of a corner drop. Depending on the design of the lifting device, the type of rigging used, the administrative vertical carry height limit, and the stiffness of the impacted surface, site-specific analyses may be required to demonstrate that the deceleration limit of 45g's is not exceeded.
3.V.2.b.i.(1), Page 3-19, Para. 1, "All concrete used in storage cask system ISFSIs, and subject to NRC review, should be reinforced..." 3.V.2.b.i.(2)(b), Page 3-20, Para. 1, "The NRC accepts the use of ACI 349 for the design, material selection and specification, and construction of all reinforced concrete structures that are not addressed within the scope of ACI 359". 3.V.2.c.i, Page 3-22, Para. 3, "Materials and material properties used for the design and construction of reinforced concrete structures important to safety but not within the scope	Exception: The HI-STORM overpack concrete is not reinforced. However, ACI 349 [1.0.4] is used as guidance for the material selection and specification, and placement of the plain concrete. Appendix 1.D provides the relevant sections of ACI 349 applicable to the plain concrete in the overpack, including clarifications on implementation of this code. ACI 318-95 [1.0.5] is used for the calculation of the compressive strength of the plain concrete.	Concrete is provided in the HI-STORM overpack primarily for the purpose of radiation shielding during normal operations. During lifting and handling operations and under certain accident conditions, the compressive strength of the concrete (which is not impaired by the absence of reinforcement) is utilized. However, since the structural reliance under loadings which produce section flexure and tension is entirely on the steel structure of the overpack, reinforcement in the concrete will serve no useful purpose. To ensure the quality of the shielding concrete, all relevant provisions of ACI 349 are imposed as clarified in Appendix 1.D. The temperature limits for normal conditions are per Paragraph A.4.3 of Appendix A to ACI 349 and temperature limits for

Table 1.0.3 (continued)

HI-STORM 100 SYSTEM FSAR CLARIFICATIONS AND EXCEPTIONS TO NUREG-1536

NUREG-1536 Requirement	Alternate Method to Meet NUREG-1536 Intent	Justification
of ACI 359 should comply with the requirements of ACI 349".		off-normal and accident conditions are per Paragraph A.4.2 of Appendix A to ACI 349. Finally, the Fort St. Vrain ISFSI (Docket No. 72-9) also utilized plain concrete for shielding purposes, which is important to safety.
3.V.3.b.i.(2), Page 3-29, Para. 1, "The NRC accepts the use of ANSI/ANS-57.9 (together with the codes and standards cited therein) as the basic reference for ISFSI structures important to safety that are not designed in accordance with Section III of the ASME B&PV Code."	Clarification: The HI-STORM overpack steel structure is designed in accordance with the ASME B&PV Code, Section III, Subsection NF, Class 3. Any exceptions to the Code are listed in Table 2.2.15.	The overpack structure is a steel weldment consisting of "plate and shell type" members. As such, it is appropriate to design the structure to Section III, Class 3 of Subsection NF. The very same approach has been used in the structural evaluation of the "intermediate shells" in the HI-STAR 100 overpack (Docket Number 72-1008) previously reviewed and approved by the USNRC.
4.IV.5, Page 4-2 "for each fuel type proposed for storage, the DCSS should ensure a very low probability (e.g., 0.5 percent per fuel rod) of cladding breach during long-term storage." 4.IV.1, Page 4-3, Para. 1 "the staff should verify that cladding temperatures for each fuel type proposed for storage will be below the expected damage thresholds for normal conditions of storage." 4.IV.1, Page 4-3, Para. 2 "fuel cladding limits for each fuel type should be defined in the SAR with thermal restrictions in the DCSS technical specifications." 4.V.1, Page 4-3, Para. 4 "the applicant should	Clarification: As described in Section 4.3, all fuel array types authorized for storage are assigned a single peak fuel cladding temperature limit.	As described in Section 4.3, all fuel array types authorized for storage have been evaluated for the peak normal fuel cladding temperature limit of 400^{o}C.

Table 1.0.3 (continued)

HI-STORM 100 SYSTEM FSAR CLARIFICATIONS AND EXCEPTIONS TO NUREG-1536

NUREG-1536 Requirement	Alternate Method to Meet NUREG-1536 Intent	Justification
verify that these cladding temperature limits are appropriate for all fuel types proposed for storage, and that the fuel cladding temperatures will remain below the limit for facility operations (e.g., fuel transfer) and the worst-case credible accident."		
4.V.4.a, Page 4-6, Para. 6 "the basket wall temperature of the hottest assembly can then be used to determine the peak rod temperature of the hottest assembly using the Wooten-Epstein correlation."	Clarification: As discussed in Subsection 4.4.2, conservative maximum fuel temperatures are obtained directly from the cask thermal analysis. The peak fuel cladding temperatures are then used to determine the corresponding peak basket wall temperatures using a finite-element based update of Wooten-Epstein (described in Subsection 4.4.1.1.2)	The finite-element based thermal conductivity is greater than a Wooten-Epstein based value. This larger thermal conductivity minimizes the fuel-to-basket temperature difference. Since the basket temperature is less than the fuel temperature, minimizing the temperature difference conservatively maximizes the basket wall temperature.
4.V.4.b, Page 4-7, Para. 2 "high burnup effects should also be considered in determining the fuel region effective thermal conductivity."	Exception: All calculations of fuel assembly effective thermal conductivities, described in Subsection 4.4.1.1.2, use nominal fuel design dimensions, neglecting wall thinning associated with high burnup.	Within Subsection 4.4.1.1.2, the calculated effective thermal conductivities based on nominal design fuel dimensions are compared with available literature values and are demonstrated to be conservative by a substantial margin.
4.V.4.c, Page 4-7, Para. 5 "a heat balance on the surface of the cask should be given and the results presented."	Clarification: No additional heat balance is performed or provided.	The FLUENT computational fluid dynamics program used to perform evaluations of the HI-STORM Overpack and HI-TRAC transfer cask, which uses a

Table 1.0.3 (continued)

HI-STORM 100 SYSTEM FSAR CLARIFICATIONS AND EXCEPTIONS TO NUREG-1536

NUREG-1536 Requirement	Alternate Method to Meet NUREG-1536 Intent	Justification
		discretized numerical solution algorithm, enforces an energy balance on all discretized volumes throughout the computational domain. This solution method, therefore, ensures a heat balance at the surface of the cask.
4.V.5.a, Page 4-8, Para. 2 "the SAR should include input and output file listings for the thermal evaluations."	<u>Exception:</u> No input or output file listings are provided in Chapter 4.	A complete set of computer program input and output files would be in excess of three hundred pages. All computer files are considered proprietary because they provide details of the design and analysis methods. In order to minimize the amount of proprietary information in the FSAR, computer files are provided in the proprietary calculation packages.
4.V.5.c, Page 4-10, Para. 3 "free volume calculations should account for thermal expansion of the cask internal components and the fuel when subjected to accident temperatures.	<u>Exception:</u> All free volume calculations use nominal confinement boundary dimensions, but the volume occupied by the fuel assemblies is calculated using maximum weights and minimum densities.	Calculating the volume occupied by the fuel assemblies using maximum weights and minimum densities conservatively overpredicts the volume occupied by the fuel and correspondingly underpredicts the remaining free volume.

Table 1.0.3 (continued)

HI-STORM 100 SYSTEM FSAR CLARIFICATIONS AND EXCEPTIONS TO NUREG-1536

NUREG-1536 Requirement	Alternate Method to Meet NUREG-1536 Intent	Justification
7.V.4 “Confinement Analysis. Review the applicant’s confinement analysis and the resulting annual dose at the controlled area boundary.”	Exception: No confinement analysis is performed and no effluent dose at the controlled area boundary is calculated.	The MPC uses redundant closures to assure that there is no release of radioactive materials under all credible conditions. Analyses presented in Chapters 3 and 11 demonstrate that the confinement boundary does not degrade under all normal, off-normal, and accident conditions. Multiple inspection methods are used to verify the integrity of the confinement boundary (e.g.,non-destructive examinations and pressure testing). Pursuant to ISG-18, the Holtec MPC is constructed in a manner that supports leakage from the confinement boundary being non-credible. Therefore, no confinement analysis is required.
9.V.1.a, Page 9-4, Para. 4 "Acceptance criteria should be defined in accordance with NB/NC-5330, "Ultrasonic Acceptance Standards"."	Clarification: Section 9.1.1.1 and the Design Drawings specify that the ASME Code, Section III, Subsection NB, Article NB-5332 will be used for the acceptance criteria for the volumetric examination of the MPC lid-to-shell weld.	In accordance with the first line on page 9-4, the NRC endorses the use of "...appropriate acceptance criteria as defined by either the ASME code, or an alternative approach..." The ASME Code, Section III, Subsection NB, Paragraph NB-5332 is appropriate acceptance criteria for pre-service examination.

Table 1.0.3 (continued)

HI-STORM 100 SYSTEM FSAR CLARIFICATIONS AND EXCEPTIONS TO NUREG-1536

NUREG-1536 Requirement	Alternate Method to Meet NUREG-1536 Intent	Justification
9.V.1.d, Para. 1 "Tests of the effectiveness of both the gamma and neutron shielding may be required if, for example, the cask contains a poured lead shield or a special neutron absorbing material."	Exception: Subsection 9.1.5 describes the control of special processes, such as neutron shield material installation, to be performed in lieu of scanning or probing with neutron sources.	The dimensional compliance of all shielding cavities is verified by inspection to design drawing requirements prior to shield installation. The Holtite-A shield material is installed in accordance with written, approved, and qualified special process procedures. The composition of the Holtite-A is confirmed by inspection and tests prior to first use. Following the first loading for the HI-TRAC transfer cask and each HI-STORM overpack, a shield effectiveness test is performed in accordance with written approved procedures, as specified in Section 9.1.
13.III, " the application must include, at a minimum, a description that satisfies the requirements of 10 CFR Part 72, Subpart G, 'Quality Assurance'..."	Exception: Section 13.0 incorporates the NRC-approved Holtec International Quality Assurance Program Manual by reference rather than describing the Holtec QA program in detail.	The NRC has approved Revision 13 of the Holtec Quality Assurance Program Manual under 10 CFR 71 (NRC QA Program Approval for Radioactive Material Packages No. 0784, Rev. 3). Pursuant to 10 CFR 72.140(d), Holtec will apply this QA program to all important-to-safety dry storage cask activities. Incorporating the Holtec QA Program Manual by reference eliminates duplicate documentation.

1.5 DRAWINGS

The following HI-STORM 100 System drawings and bills of materials are provided on subsequent pages in this subsection:

Drawing Number/Sheet	Description	Rev.
3923	MPC Enclosure Vessel	~~14~~15
3925	MPC-24E/EF Fuel Basket Assembly	5
3926	MPC-24 Fuel Basket Assembly	7
3927	MPC-32 Fuel Basket Assembly	~~9~~10
3928	MPC-68/68F/68FF Basket Assembly	8
1495 Sht 1/6	HI-STORM 100 Assembly	13
1495 Sht 2/6	Cross Section "Z" - "Z" View of HI-STORM	18
1495 Sht 3/6	Section "Y" - "Y" of HI-STORM	12
1495 Sht 4/6	Section "X" -"X" of HI-STORM	13
1495 Sht 5/6	Section "W" -"W" of HI-STORM	15
1561 Sht 1/6	View "A" -"A" of HI-STORM	11
1561 Sht 2/6	Detail "B" of HI-STORM	15
1561 Sht 3/6	Detail of Air Inlet of HI-STORM	11
1561 Sht 4/6	Detail of Air Outlet of HI-STORM	12
3669	HI-STORM 100S Assembly	9
1880 Sht 1/10	125 Ton HI-TRAC Outline with Pool Lid	9
1880 Sht 2/10	125 Ton HI-TRAC Body Sectioned Elevation	10
1880 Sht 3/10	125 Ton HI-TRAC Body Sectioned Elevation "B" - "B"	9
1880 Sht 4/10	125 Ton Transfer Cask Detail of Bottom Flange	10
1880 Sht 5/10	125 Ton Transfer Cask Detail of Pool Lid	10
1880 Sht 6/10	125 Ton Transfer Cask Detail of Top Flange	10
1880 Sht 7/10	125 Ton Transfer Cask Detail of Top Lid	9
1880 Sht 8/10	125 Ton Transfer Cask View "Y" - "Y"	9
1880 Sht 9/10	125 Ton Transfer Cask Lifting Trunnion and Locking Pad	7
1880 Sht 10/10	125 Ton Transfer Cask View "Z" - "Z"	9
1928 Sht 1/2	125 Ton HI-TRAC Transfer Lid Housing Detail	11
1928 Sht 2/2	125 Ton HI-TRAC Transfer Lid Door Detail	10
2145 Sht 1/10	100 Ton HI-TRAC Outline with Pool Lid	8
2145 Sht 2/10	100 Ton HI-TRAC Body Sectioned Elevation	8
2145 Sht 3/10	100 Ton HI-TRAC Body Sectioned Elevation 'B-B'	8
2145 Sht 4/10	100 Ton HI-TRAC Detail of Bottom Flange	7
2145 Sht 5/10	100 Ton HI-TRAC Detail of Pool Lid	6
2145 Sht 6/10	100 Ton HI-TRAC Detail of Top Flange	8
2145 Sht 7/10	100 Ton HI-TRAC Detail of Top Lid	8
2145 Sht 8/10	100 Ton HI-TRAC View Y-Y	8
2145 Sht 9/10	100 Ton HI-TRAC Lifting Trunnions and Locking Pad	5
2145 Sht 10/10	100 Ton HI-TRAC View Z-Z	7
2152 Sht 1/2	100 Ton HI-TRAC Transfer Lid Housing Detail	10

Drawing Number/Sheet	Description	Rev.
2152 Sht 2/2	100 Ton HI-TRAC Transfer Lid Door Detail	8
3187	Lug and Anchoring Detail for HI-STORM 100A	2
BM-1575, Sht 1/2	Bill-of-Materials HI-STORM 100 Storage Overpack	19
BM-1575, Sht 2/2	Bill-of-Materials HI-STORM 100 Storage Overpack	19
BM-1880, Sht 1/2	Bill-of-Material for 125 Ton HI-TRAC	9
BM-1880, Sht 2/2	Bill-of-Material for 125 Ton HI-TRAC	7
BM-1928, Sht 1/1	Bill-of-Material for 125 Ton HI-TRAC Transfer Lid	10
BM-2145 Sht 1/2	Bill-of-Material for 100 Ton HI-TRAC	6
BM-2145 Sht 2/2	Bill-of-Material for 100 Ton HI-TRAC	5
BM-2152 Sht 1/1	Bill-of-Material for 100 Ton HI-TRAC Transfer Lid	8
3768	125 Ton HI-TRAC 125D Assembly	7
4116	HI-STORM 100S Version B	~~11~~*12*
4128	100 Ton HI-TRAC 100D Assembly	4
4724	~~100 Ton~~ *HI-TRAC 100D Version IP1 Assembly*	*0*

2.1 SPENT FUEL TO BE STORED

2.1.1 Determination of The Design Basis Fuel

The HI-STORM 100 System is designed to store most types of fuel assemblies generated in the commercial U.S. nuclear industry. Boiling-water reactor (BWR) fuel assemblies have been supplied by The General Electric Company (GE), Siemens, Exxon Nuclear, ANF, UNC, ABB Combustion Engineering, and Gulf Atomic. Pressurized-water reactor (PWR) fuel assemblies are generally supplied by Westinghouse, Babcock & Wilcox, ANF, and ABB Combustion Engineering. ANF, Exxon, and Siemens are historically the same manufacturing company under different ownership. Within this report, SPC is used to designate fuel manufactured by ANF, Exxon, or Siemens. Publications such as Refs. [2.1.1] and [2.1.2] provide a comprehensive description of fuel discharged from U.S. reactors. A central object in the design of the HI-STORM 100 System is to ensure that a majority of SNF discharged from the U.S. reactors can be stored in one of the MPCs.

The cell openings and lengths in the fuel basket have been sized to accommodate the BWR and PWR assemblies listed in Refs. [2.1.1] and [2.1.2] except as noted below. Similarly, the cavity length of the multi-purpose canisters has been set at a dimension, which permits storing most types of PWR fuel assemblies and BWR fuel assemblies with or without fuel channels. The one exception is as follows:

i. The South Texas Units 1 & 2 SNF, and CE 16x16 System 80 SNF are too long to be accommodated in the available MPC cavity length.

In addition to satisfying the cross sectional and length compatibility, the active fuel region of the SNF must be enveloped in the axial direction by the neutron absorber located in the MPC fuel basket. Alignment of the neutron absorber with the active fuel region is ensured by the use of upper and lower fuel spacers suitably designed to support the bottom and restrain the top of the fuel assembly. The spacers axially position the SNF assembly such that its active fuel region is properly aligned with the neutron absorber in the fuel basket. Figure 2.1.5 provides a pictorial representation of the fuel spacers positioning the fuel assembly active fuel region. Both the upper and lower fuel spacers are designed to perform their function under normal, off-normal, and accident conditions of storage.

In summary, the geometric compatibility of the SNF with the MPC designs does not require the definition of a design basis fuel assembly. This, however, is not the case for structural, confinement, shielding, thermal-hydraulic, and criticality criteria. In fact, a particular fuel type in a category (PWR or BWR) may not control the cask design in all of the above-mentioned criteria. To ensure that no SNF listed in Refs. [2.1.1] and [2.1.2] which is geometrically admissible in the MPC is precluded, it is necessary to determine the governing fuel specification for each analysis criterion. To make the necessary determinations, potential candidate fuel assemblies for each qualification criterion were considered. Table 2.1.1 lists the PWR fuel assemblies that were evaluated. These fuel assemblies were evaluated to define the governing design criteria for PWR fuel. The BWR fuel assembly designs evaluated are listed in Table 2.1.2. Tables 2.1.3 and 2.1.4 provide the fuel characteristics determined to be acceptable for storage in the HI-STORM 100 System. Section 2.1.9 summarizes the authorized contents for the HI-STORM 100 System. Any fuel assembly that has fuel characteristics within the range of Tables 2.1.3 and 2.1.4 and meets the other limits specified in Section 2.1.9 is acceptable for storage in the HI-STORM 100 System. Tables 2.1.3 and 2.1.4 present the groups of fuel assembly types defined as "array/classes" as described in further detail in Chapter 6. Table 2.1.5 lists the BWR and PWR fuel assembly designs, which are found to govern for three qualification criteria, namely reactivity, shielding, and thermal. Substantiating results of analyses for the governing assembly types are presented in the respective chapters dealing with the specific qualification topic. Additional information on the design basis fuel definition is presented in the following subsections.

2.1.2 Intact SNF Specifications

Intact fuel assemblies are defined as fuel assemblies without known or suspected cladding defects greater than pinhole leaks and hairline cracks, and which can be handled by normal means. The design payload for the HI-STORM 100 System is intact ZR or stainless steel (SS) clad fuel assemblies with the characteristics listed in Tables 2.1.17 through 2.1.24.

Intact fuel assemblies without fuel rods in fuel rod locations cannot be loaded into the HI-STORM 100 unless dummy fuel rods, which occupy a volume greater than or equal to the original fuel rods, replace the missing rods prior to loading. Any intact fuel assembly that falls within the geometric, thermal, and nuclear limits established for the design basis intact fuel assembly, as defined in Section 2.1.9 can be safely stored in the HI-STORM 100 System.

The range of fuel characteristics specified in Tables 2.1.3 and 2.1.4 have been evaluated in this FSAR and are acceptable for storage in the HI-STORM 100 System within the decay heat, burnup, and cooling time limits specified in Section 2.1.9 for intact fuel assemblies.

2.1.3 Damaged SNF and Fuel Debris Specifications

Damaged fuel and fuel debris are defined in Table 1.0.1.

To aid in loading and unloading, damaged fuel assemblies and fuel debris will be loaded into stainless steel damaged fuel containers (DFCs) provided with 250 x 250 fine mesh screens, for storage in the HI-STORM 100 System (see Figures 2.1.1 and 2.1.2B, C, and D). The MPC-24E and MPC 32 are designed to accommodate PWR damaged fuel. The MPC-24EF and MPC-32F are designed to accommodate PWR damaged fuel and fuel debris. The MPC-68 is designed to accommodate BWR damaged fuel. The MPC-68F and MPC-68FF are designed to accommodate BWR damaged fuel and fuel debris. The appropriate structural, thermal, shielding, criticality, and confinement analyses have been performed to account for damaged fuel and fuel debris and are described in their respective chapters that follow. The limiting design characteristics for damaged fuel assemblies and restrictions on the number and location of damaged fuel containers authorized for loading in each MPC model are provided in Section 2.1.9. Dresden Unit 1 fuel assemblies contained in Transnuclear-designed damaged fuel canisters and one Dresden Unit 1 thoria rod canister have been approved for storage directly in the HI-STORM 100 System without re-packaging (see Figures 2.1.2 and 2.1.2A).

MPC contents classified as fuel debris are required to be stored in DFCs and in the applicable “F” model MPC as specified in Section 2.1.9. The “F”(or “FF”) indicates the MPC is qualified for storage of intact fuel, damaged fuel, and fuel debris, in quantities and locations specified in Section 2.1.9. The basket designs for the standard and “F” model MPCs are identical. The lid and shell designs of the “F” models are unique in that the upper shell portion of the canister is thickened for additional strength needed under hypothetical accident conditions of transportation under 10 CFR 71. This design feature is not required for dry storage, but must be considered in fuel loading for dry storage to ensure the dual-purpose function of the MPC by eliminating the need to re-package the fuel for transportation. Figure 2.1.9 shows the details of the differences between the standard and “F” model MPC shells. These details are common for both the PWR and BWR series MPC models.

2.1.4 Deleted

2.1.5 Structural Parameters for Design Basis SNF

The main physical parameters of an SNF assembly applicable to the structural evaluation are the fuel assembly length, envelope (cross sectional dimensions), and weight. These parameters, which define the mechanical and structural design, are specified in Section 2.1.9. The centers of gravity reported in Section 3.2 are based on the maximum fuel assembly weight. Upper and lower fuel spacers (as appropriate) maintain the axial position of the fuel assembly within the MPC basket and, therefore, the location of the center of gravity. The upper and lower fuel spacers are designed to withstand normal, off-normal, and accident conditions of storage. An axial clearance of approximately 2 to 2-1/2 inches is provided to account for the irradiation and thermal growth of the fuel assemblies. The suggested upper and lower fuel spacer lengths are listed in Tables 2.1.9 and 2.1.10. In order to qualify for storage in the MPC, the SNF must satisfy the physical parameters listed in Section 2.1.9.

2.1.6 Thermal Parameters for Design Basis SNF

The principal thermal design parameter for the stored fuel is the peak fuel cladding temperature, which is a function of the maximum heat generation rate per assembly and the decay heat removal capabilities of the HI-STORM 100 System. No attempt is made to link the maximum allowable decay heat per fuel assembly with burnup, enrichment, or cooling time. Rather, the decay heat per fuel assembly is adjusted to yield peak fuel cladding temperatures with an allowance for margin to the temperature limit. The same fuel assembly decay heats are used for all fuel assembly designs within a given class of fuel assemblies (i.e., ZR clad PWR, stainless steel clad BWR, etc.).

To ensure the permissible fuel cladding temperature limits are not exceeded, Section 2.1.9 specifies the allowable decay heat per assembly for each MPC model. For both uniform and regionalized loading of moderate and high burnup fuel assemblies, the allowable decay heat per assembly is presented in Section 2.1.9.

Section 2.1.9 also includes separate cooling time, burnup, and decay heat limits for uniform fuel loading and regionalized fuel loading. Regionalized loading allows higher heat emitting fuel assemblies to be stored in the center fuel storage locations than would otherwise be authorized for storage under uniform loading conditions.

The fuel cladding temperature is also affected by the heat transfer characteristics of the fuel assemblies. There is no single fuel assembly design used in all thermal calculations that is bounding of all others. Instead, each thermal calculation, comprising the overall thermal analysis presented in Chapter 4, was performed using the fuel assembly design that results in the most conservative result for the individual calculation. By always using the fuel assembly design that is most conservative for a particular calculation, it is ensured that each calculation is bounding for all fuel assembly designs. The bounding fuel assembly design for each thermal calculation and fuel type is provided in Table 2.1.5.

Finally, the axial variation in the heat generation rate in the design basis fuel assembly is defined based on the axial burnup distribution. For this purpose, the data provided in Refs. [2.1.7] and [2.1.8] are utilized and summarized in Table 2.1.11 and Figures 2.1.3 and 2.1.4 for reference. These distributions are representative of fuel assemblies with the design basis burnup levels considered. These distributions are used for analyses only, and do not provide a criteria for fuel assembly acceptability for storage in the HI-STORM 100 System.

Except for MPC-68F, fuel may be stored in the MPC using one of two storage strategies, namely, uniform loading and regionalized loading. Uniform loading allows storage of any fuel assembly in any fuel storage location, subject to additional restrictions, such as those for loading of fuel assemblies containing non-fuel hardware as defined in Table 1.0.1. Regionalized fuel loading allows for higher heat emitting fuel assemblies to be stored in the central core basket storage locations (inner region) with lower heat emitting fuel assemblies in the peripheral fuel storage locations (outer region). Regionalized loading allows storage of higher heat emitting fuel assemblies than would otherwise be permitted using the uniform loading strategy. The definition of the regions for each MPC model provided in Table 2.1.13. Regionalized fuel loading is not permitted in MPC-68F.

2.1.7 Radiological Parameters for Design Basis SNF

The principal radiological design criteria for the HI-STORM 100 System are the 10CFR72.104 site boundary dose rate limits and maintaining operational dose rates as low as reasonably achievable (ALARA). The radiation dose is directly affected by the gamma and neutron source terms of the SNF assembly.

The gamma and neutron sources are separate and are affected differently by enrichment, burnup, and cooling time. It is recognized that, at a given burnup, the radiological source terms increase monotonically as the initial enrichment is reduced. The shielding design basis fuel assembly, therefore, is evaluated at conservatively high burnups, low cooling times, and low enrichments, as discussed in Chapter 5. The shielding design basis fuel assembly thus bounds all other fuel assemblies.

The design basis dose rates can be met by a variety of burnup levels and cooling times. Section 2.1.9 provides the procedure for determining burnup and cooling time limits for all of the authorized fuel assembly array/classes for both uniform fuel loading and regionalized loading. Table 2.1.11 and Figures 2.1.3 and 2.1.4 provide the axial distribution for the radiological source terms for PWR and BWR fuel assemblies based on the axial burnup distribution. The axial burnup distributions are representative of fuel assemblies with the design basis burnup levels considered. These distributions are used for analyses only, and do not provide a criteria for fuel assembly acceptability for storage in the HI-STORM 100 System.

Thoria rods placed in Dresden Unit 1 Thoria Rod Canisters meeting the requirements of Table 2.1.12 and Dresden Unit 1 fuel assemblies with one Antimony-Beryllium neutron source have been qualified for storage. Up to one Thoria Rod Canister is authorized for storage in combination with other intact and damaged fuel, and fuel debris as specified in Section 2.1.9.

Non-fuel hardware, as defined in Table 1.0.1, has been evaluated and is authorized for storage in the PWR MPCs as specified in Section 2.1.9.

2.1.8 Criticality Parameters for Design Basis SNF

As discussed earlier, the MPC-68, MPC-68F, MPC-68FF, MPC-32 and MPC-32F feature a basket without flux traps. In the aforementioned baskets, there is one panel of neutron absorber between two adjacent fuel assemblies. The MPC-24, MPC-24E, and MPC-24EF employ a construction wherein two neighboring fuel assemblies are separated by two panels of neutron absorber with a water gap between them (flux trap construction).

The minimum ^{10}B areal density in the neutron absorber panels for each MPC model is shown in Table 2.1.15.

For all MPCs, the ^{10}B areal density used for the criticality analysis is conservatively established below the minimum values shown in Table 2.1.15. For Boral, the value used in the analysis is 75% of the minimum value, while for METAMIC, it is 90% of the minimum value. This is consistent with NUREG-1536 [2.1.5] which suggests a 25% reduction in ^{10}B areal density credit when subject to standard acceptance tests, and which allows a smaller reduction when more comprehensive tests of the areal density are performed.

The criticality analyses for the MPC-24, MPC-24E and MPC-24EF (all with higher enriched fuel) and for the MPC-32 and MPC-32F were performed with credit taken for soluble boron in the MPC water during wet loading and unloading operations. Table 2.1.14 and 2.1.16 provide the required soluble boron concentrations for these MPCs.

2.1.9 Summary of Authorized Contents

Tables 2.1.3, 2.1.4, 2.1.12, and 2.1.17 through 2.1.29 together specify the limits for spent fuel and non-fuel hardware authorized for storage in the HI-STORM 100 System. The limits in these tables are derived from the safety analyses described in the following chapters of this FSAR. Fuel classified as damaged fuel assemblies or fuel debris must be stored in damaged fuel containers for storage in the HI-STORM 100 System.

Tables 2.1.17 through 2.1.24 are the baseline tables that specify the fuel assembly limits for each of the MPC models, with appropriate references to the other tables in this section for certain other limits. Tables 2.1.17 through 2.1.24 refer to Section 2.1.9.1 for ZR-clad fuel limits on minimum cooling time, maximum decay heat, and maximum burnup for uniform and regionalized fuel loading. Limits on decay heat, burnup, and cooling time for stainless steel-clad fuel are provided in Tables 2.1.17 through 2.1.24.

2.1.9.1 Decay Heat, Burnup, and Cooling Time Limits for ZR-Clad Fuel

Each ZR-clad fuel assembly and any PWR integral non-fuel hardware (NFH) to be stored in the HI-STORM 100 System must meet the following limits, in addition to meeting the physical limits specified elsewhere in this section, to be authorized for storage in the HI-STORM 100 System. The contents of each fuel storage location (fuel assembly and NFH) to be stored must be verified to have, as applicable:

- A decay heat less than or equal to the maximum allowable value.

- An assembly average enrichment greater than or equal to the minimum value used in determining the maximum allowable burnup.

- A burnup less than or equal to the maximum allowable value.

- A cooling time greater than or equal to the minimum allowable value.

The maximum allowable ZR-clad fuel storage location decay heat values are determined using the methodology described in Section 2.1.9.1.1 or 2.1.9.1.2 depending on whether uniform fuel loading or regionalized fuel loading is being implemented[†]. The decay heat limits are independent of burnup, cooling time, or enrichment and are based strictly on the thermal analysis described in Chapter 4. Decay heat limits must be met for all contents in a fuel storage location (i.e., fuel and PWR non-fuel hardware, as applicable).

The maximum allowable average burnup per fuel storage location is determined by calculation as a function of minimum enrichment, maximum allowable decay heat, and minimum cooling time from 3 to 20 years, as described in Section 2.1.9.1.3.

Section 12.2.10 describes how compliance with these limits may be verified, including practical examples.

2.1.9.1.1 Uniform Fuel Loading Decay Heat Limits for ZR-Clad Fuel

Table 2.1.26 provides the maximum allowable decay heat per fuel storage location for ZR-clad fuel in uniform fuel loading for each MPC model.

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† Note that the stainless steel-clad fuel limits apply to all fuel in the MPC, if a mixture of stainless steel and ZR-clad fuel is stored in the same MPC. The stainless steel-clad fuel assembly decay heat limits may be found in Table 2.1.17 through 2.1.24

2.1.9.1.2 Regionalized Fuel Loading Decay Heat Limits for ZR-Clad Fuel

Table 2.1.27 provides the maximum allowable decay heat per fuel storage location for ZR-clad fuel in both the inner and outer regions for regionalized fuel loading in each MPC model.

2.1.9.1.3 Burnup Limits as a Function of Cooling Time for ZR-Clad Fuel

The maximum allowable ZR-clad fuel assembly average burnup varies with the following parameters, based on the shielding analysis in Chapter 5:

- Minimum required fuel assembly cooling time
- Maximum allowable fuel assembly decay heat
- Minimum fuel assembly average enrichment

The calculation described in this section is used to determine the maximum allowable fuel assembly burnup for minimum cooling times between 3 and 20 years, using maximum decay heat and minimum enrichment as input values. This calculation may be used to create multiple burnup versus cooling time tables for a particular fuel assembly array/class and different minimum enrichments. The allowable maximum burnup for a specific fuel assembly may be calculated based on the assembly's particular enrichment and cooling time.

(i) Choose a fuel assembly minimum enrichment, E_{235}.

(ii) Calculate the maximum allowable fuel assembly average burnup for a minimum cooling time between 3 and 20 years using the equation below:

$$Bu = (A \times q) + (B \times q^2) + (C \times q^3) + [D \times (E_{235})^2] + (E \times q \times E_{235}) + (F \times q^2 \times E_{235}) + G$$

Equation 2.1.9.3

Where:

Bu = Maximum allowable assembly average burnup (MWD/MTU)

q = Maximum allowable decay heat per fuel storage location determined in Section 2.1.9.1 or 2.1.9.2 (kW)

E_{235} = Minimum fuel assembly average enrichment (wt. % ^{235}U)
(e.g., for 4.05 wt. %, use 4.05)

A through G = Coefficients from Tables 2.1.28 or 2.1.29 for the applicable fuel assembly array/class and minimum cooling time.

2.1.9.1.4 Other Considerations

In computing the allowable maximum fuel storage location decay heats and fuel assembly average burnups, the following requirements apply:

- Calculated burnup limits shall be rounded down to the nearest integer
- Calculated burnup limits greater than 68,200 MWD/MTU for PWR fuel and 65,000 MWD/MTU for BWR fuel must be reduced to be equal to these values.
- Linear interpolation of calculated burnups between cooling times for a given fuel assembly maximum decay heat and minimum enrichment is permitted. For example, the allowable burnup for a minimum cooling time of 4.5 years may be interpolated between those burnups calculated for 4 and 5 years.
- ZR-clad fuel assemblies must have a minimum enrichment, as defined in Table 1.0.1, greater than or equal to the value used in determining the maximum allowable burnup per Section 2.1.9.1.3 to be authorized for storage in the MPC.
- When complying with the maximum fuel storage location decay heat limits, users must account for the decay heat from both the fuel assembly and any PWR non-fuel hardware, as applicable for the particular fuel storage location, to ensure the decay heat emitted by all contents in a storage location does not exceed the limit.

Section 12.2.10 provides a practical example of determining fuel storage location decay heat, burnup, and cooling time limits and verifying compliance for a set of example fuel assemblies.

Table 2.1.1

PWR FUEL ASSEMBLIES EVALUATED TO DETERMINE DESIGN BASIS SNF

Assembly Class	**Array Type**
B&W 15x15	All
B&W 17x17	All
CE 14x14	All
CE 16x16	All except System 80™
WE 14x14	All
WE 15x15	All
WE 17x17	All
St. Lucie	All
Ft. Calhoun	All
Haddam Neck (Stainless Steel Clad)	All
San Onofre 1 (Stainless Steel Clad)	All
Indian Point 1	All

Table 2.1.2

BWR FUEL ASSEMBLIES EVALUATED TO DETERMINE DESIGN BASIS SNF

Assembly Class	Array Type			
GE BWR/2-3	All 7x7	All 8x8	All 9x9	All 10x10
GE BWR/4-6	All 7x7	All 8x8	All 9x9	All 10x10
Humboldt Bay	All 6x6	All 7x7 (ZR Clad)		
Dresden-1	All 6x6	All 8x8		
LaCrosse (Stainless Steel Clad)	All			

Table 2.1.3
PWR FUEL ASSEMBLY CHARACTERISTICS (Note 1)

Fuel Assembly Array/ Class	**14x14 A**	**14x14 B**	**14x14 C**	**14x14 D**	**14x14E**
Clad Material (Note 2)	ZR	ZR	ZR	SS	SS
Design Initial U (kg/assy.) (Note 3)	≤ 365	≤ 412	≤ 438	≤ 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)	*N/A* ~~≤ 5.0~~*~~4.5~~* ~~(24)~~ *N/A* ~~≤ 5.0~~*~~4.5~~* ~~(24E/24EF)~~
Initial Enrichment (MPC-24, 24E, 24EF, 32 or 32F with soluble boron credit - see Note 5) (wt % ^{235}U)	≤ 5.0	≤ 5.0	≤ 5.0	≤ 5.0	~~5.0~~ ≤ *4.5* *(MPC-32/32F only – Note 8)*
No. of Fuel Rod Locations	179	179	176	180	173
Fuel Clad O.D. (in.)	≥ 0.400	≥ 0.417	≥ 0.440	≥ 0.422	≥ 0.3415
Fuel 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 2.1.3 (continued)
PWR FUEL ASSEMBLY CHARACTERISTICS (Note 1)

Fuel Assembly Array/Class	**15x15 A**	**15x15 B**	**15x15 C**	**15x15 D**	**15x15 E**	**15x15 F**
Clad Material (Note 2)	ZR	ZR	ZR	ZR	ZR	ZR
Design Initial U (kg/assy.) (Note 3)	≤ 473	≤ 473	≤ 473	≤ 495	≤ 495	≤ 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 5) (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 Clad O.D. (in.)	≥ 0.418	≥ 0.420	≥ 0.417	≥ 0.430	≥ 0.428	≥ 0.428
Fuel 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 2.1.3 (continued)
PWR FUEL ASSEMBLY CHARACTERISTICS (Note 1)

Fuel Assembly Array and Class	**15x15 G**	**15x15H**	**16x16 A**	**17x17A**	**17x17 B**	**17x17 C**
Clad Material (Note 2)	SS	ZR	ZR	ZR	ZR	ZR
Design Initial U (kg/assy.) (Note 3)	≤ 420	≤ 495	≤ 448	≤ 433	≤ 474	≤ 480
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 5) (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 Clad O.D. (in.)	≥ 0.422	≥ 0.414	≥ 0.382	≥ 0.360	≥ 0.372	≥ 0.377
Fuel 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	≥ 0.140	≥ 0.0400	≥ 0.016	≥ 0.014	≥ 0.020

Table 2.1.3 (continued)
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. See Table 1.0.1 for the definition of "ZR."

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 Tables 2.1.14 and 2.1.16, as applicable.

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.

8. *This fuel assembly array/class includes only the Indian Point Unit 1 fuel assembly. This assembly class has been analyzed throughout this FSAR in all PWR MPCs, however it is only to be loaded into the MPC-32/32F.*

Table 2.1.4
BWR FUEL ASSEMBLY CHARACTERISTICS (Note 1)

Fuel Assembly Array and Class	**6x6 A**	**6x6 B**	**6x6 C**	**7x7 A**	**7x7 B**	**8x8 A**
Clad Material (Note 2)	ZR	ZR	ZR	ZR	ZR	ZR
Design Initial U (kg/assy.) (Note 3)	≤ 110	≤ 110	≤ 110	≤ 100	≤ 198	≤ 120
Maximum Planar-Average Initial Enrichment (wt.% ^{235}U) (Note 14)	≤ 2.7	≤ 2.7 for 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 Clad O.D. (in.)	≥ 0.5550	≥ 0.5625	≥ 0.5630	≥ 0.4860	≥ 0.5630	≥ 0.4120
Fuel 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 2.1.4 (continued)
BWR FUEL ASSEMBLY CHARACTERISTICS (Note 1)

Fuel Assembly Array and Class	**8x8 B**	**8x8 C**	**8x8 D**	**8x8 E**	**8x8F**	**9x9 A**
Clad Material (Note 2)	ZR	ZR	ZR	ZR	ZR	ZR
Design Initial U (kg/assy.) (Note 3)	$\leq$ 192	$\leq$ 190	$\leq$ 190	$\leq$ 190	$\leq$ 191	$\leq$ 180
Maximum Planar-Average Initial Enrichment (wt.% ^{235}U) (Note 14)	$\leq$ 4.2	$\leq$ 4.2	$\leq$ 4.2	$\leq$ 4.2	$\leq$ 4.0	$\leq$ 4.2
Initial Maximum Rod Enrichment (wt.% ^{235}U)	$\leq$ 5.0	$\leq$ 5.0	$\leq$ 5.0	$\leq$ 5.0	$\leq$ 5.0	$\leq$ 5.0
No. of Fuel Rod Locations	63 or 64	62	60 or 61	59	64	74/66 (Note 5)
Fuel Clad O.D. (in.)	$\geq$ 0.4840	$\geq$ 0.4830	$\geq$ 0.4830	$\geq$ 0.4930	$\geq$ 0.4576	$\geq$ 0.4400
Fuel Clad I.D. (in.)	$\leq$ 0.4295	$\leq$ 0.4250	$\leq$ 0.4230	$\leq$ 0.4250	$\leq$ 0.3996	$\leq$ 0.3840
Fuel Pellet Dia. (in.)	$\leq$ 0.4195	$\leq$ 0.4160	$\leq$ 0.4140	$\leq$ 0.4160	$\leq$ 0.3913	$\leq$ 0.3760
Fuel Rod Pitch (in.)	$\leq$ 0.642	$\leq$ 0.641	$\leq$ 0.640	$\leq$ 0.640	$\leq$ 0.609	$\leq$ 0.566
Design Active Fuel Length (in.)	$\leq$ 150	$\leq$ 150	$\leq$ 150	$\leq$ 150	$\leq$ 150	$\leq$ 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.)	$\geq$ 0.034	> 0.00	> 0.00	$\geq$ 0.034	$\geq$ 0.0315	> 0.00
Channel Thickness (in.)	$\leq$ 0.120	$\leq$ 0.120	$\leq$ 0.120	$\leq$ 0.100	$\leq$ 0.055	$\leq$ 0.120

Table 2.1.4 (continued)
BWR FUEL ASSEMBLY CHARACTERISTICS (Note 1)

Fuel Assembly Array and Class	**9x9 B**	**9x9 C**	**9x9 D**	**9x9 E (Note 13)**	**9x9 F (Note 13)**	**9x9 G**
Clad Material (Note 2)	ZR	ZR	ZR	ZR	ZR	*ZR*
Design Initial U (kg/assy.) (Note 3)	≤ 180	≤ 182	≤ 182	≤ 183	≤ 183	≤ 164
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 Clad O.D. (in.)	≥ 0.4330	≥ 0.4230	≥ 0.4240	≥ 0.4170	≥ 0.4430	≥ 0.4240
Fuel 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 2.1.4 (continued)
BWR FUEL ASSEMBLY CHARACTERISTICS (Note 1)

Fuel Assembly Array and Class	**10x10 A**	**10x10 B**	**10x10 C**	**10x10 D**	**10x10 E**
Clad Material (Note 2)	ZR	ZR	ZR	SS	SS
Design Initial U (kg/assy.) (Note 3)	≤ 188	≤ 188	≤ 179	≤ 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 Clad O.D. (in.)	≥ 0.4040	≥ 0.3957	≥ 0.3780	≥ 0.3960	≥ 0.3940
Fuel 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.)	≥ 0.030	> 0.00	≥ 0.031	N/A	≥ 0.022
Channel Thickness (in.)	≤ 0.120	≤ 0.120	≤ 0.055	≤ 0.080	≤ 0.080

Table 2.1.4 (continued)
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. See Table 1.0.1 for the definition of "ZR."

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 or 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 2.1.5

DESIGN BASIS FUEL ASSEMBLY FOR EACH DESIGN CRITERION

Criterion	BWR	PWR
Reactivity (Criticality)	GE12/14 10x10 with Partial Length Rods (Array/Class 10x10A)	B&W 15x15 (Array/Class 15x15F)
Shielding	GE 7x7	B&W 15x15
Fuel Assembly Effective Planar Thermal Conductivity	GE-11 9x9	W 17x17 OFA
Fuel Basket Effective Axial Thermal Conductivity	GE 7x7	W 14x14 OFA
MPC Density and Heat Capacity	Dresden 6x6	W 14x14 OFA
MPC Fuel Basket Axial Resistance to Thermosiphon Flow	GE-11 9x9	W 17x17 OFA

Table 2.1.6

TABLE INTENTIONALLY DELETED

Table 2.1.7

TABLE INTENTIONALLY DELETED

Table 2.1.8

TABLE INTENTIONALLY DELETED

Table 2.1.9

SUGGESTED PWR UPPER AND LOWER FUEL SPACER LENGTHS

Fuel Assembly Type	Assembly Length w/o NFH[1] (in.)	Location of Active Fuel from Bottom (in.)	Max. Active Fuel Length (in.)	Upper Fuel Spacer Length (in.)	Lower Fuel Spacer Length (in.)
CE 14x14	157	4.1	137	9.5	10.0
CE 16x16	176.8	4.7	150	0	0
BW 15x15	165.7	8.4	141.8	6.7	4.1
W 17x17 OFA	159.8	3.7	144	8.2	8.5
W 17x17 Std	159.8	3.7	144	8.2	8.5
W 17x17 V5H	160.1	3.7	144	7.9	8.5
W 15x15	159.8	3.7	144	8.2	8.5
W 14x14 Std	159.8	3.7	145.2	9.2	7.5
W 14x14 OFA	159.8	3.7	144	8.2	8.5
Ft. Calhoun	146	6.6	128	10.25	20.25
St. Lucie 2	158.2	5.2	136.7	10.25	8.05
B&W 15x15 SS	137.1	3.873	120.5	19.25	19.25
W 15x15 SS	137.1	3.7	122	19.25	19.25
W 14x14 SS	137.1	3.7	120	19.25	19.25
Indian Point 1	137.2	17.705	101.5	18.75	20.0

Note: Each user shall specify the fuel spacer length based on their fuel assembly length, presence of a DFC, and allowing an approximate two to 2-1/2 inch gap under the MPC lid. Fuel spacers shall be sized to ensure that the active fuel region of intact fuel assemblies remains within the neutron poison region of the MPC basket with water in the MPC.

[1] NFH is an abbreviation for non-fuel hardware, including control components. Fuel assemblies with control components may require shorter fuel spacers.

Table 2.1.10

SUGGESTED BWR UPPER AND LOWER FUEL SPACER LENGTHS

Fuel Assembly Type	Assembly Length (in.)	Location of Active Fuel from Bottom (in.)	Max. Active Fuel Length (in.)	Upper Fuel Spacer Length (in.)	Lower Fuel Spacer Length (in.)
GE/2-3	171.2	7.3	150	4.8	0
GE/4-6	176.2	7.3	150	0	0
Dresden 1	134.4	11.2	110	18.0	28.0
Humboldt Bay	95.0	8.0	79	40.5	40.5
Dresden 1 Damaged Fuel or Fuel Debris	142.1†	11.2	110	17.0	16.9
Humboldt Bay Damaged Fuel or Fuel Debris	105.5†	8.0	79	35.25	35.25
LaCrosse	102.5	10.5	83	37.0	37.5

Note: Each user shall specify the fuel spacer length based on their fuel assembly length, presence of a DFC, and allowing an approximate two to 2-1/2 inch gap under the MPC lid. Fuel spacers shall be sized to ensure that the active fuel region of intact fuel assemblies remains within the neutron poison region of the MPC basket with water in the MPC.

† Fuel assembly length includes the damaged fuel container.

Table 2.1.11
NORMALIZED DISTRIBUTION BASED ON BURNUP PROFILE

PWR DISTRIBUTION[1]		
Interval	**Axial Distance From Bottom of Active Fuel (% of Active Fuel Length)**	**Normalized Distribution**
1	0% to 4-1/6%	0.5485
2	4-1/6% to 8-1/3%	0.8477
3	8-1/3% to 16-2/3%	1.0770
4	16-2/3% to 33-1/3%	1.1050
5	33-1/3% to 50%	1.0980
6	50% to 66-2/3%	1.0790
7	66-2/3% to 83-1/3%	1.0501
8	83-1/3% to 91-2/3%	0.9604
9	91-2/3% to 95-5/6%	0.7338
10	95-5/6% to 100%	0.4670

BWR DISTRIBUTION[2]		
Interval	**Axial Distance From Bottom of Active Fuel (% of Active Fuel Length)**	**Normalized Distribution**
1	0% to 4-1/6%	0.2200
2	4-1/6% to 8-1/3%	0.7600
3	8-1/3% to 16-2/3%	1.0350
4	16-2/3% to 33-1/3%	1.1675
5	33-1/3% to 50%	1.1950
6	50% to 66-2/3%	1.1625
7	66-2/3% to 83-1/3%	1.0725
8	83-1/3% to 91-2/3%	0.8650
9	91-2/3% to 95-5/6%	0.6200
10	95-5/6% to 100%	0.2200

1 Reference 2.1.7
2 Reference 2.1.8

Table 2.1.12

DESIGN CHARACTERISTICS FOR THORIA RODS IN D-1 THORIA ROD CANISTERS

PARAMETER	MPC-68 or MPC-68F
Cladding Type	Zircaloy
Composition	98.2 wt.% ThO_2, 1.8 wt.% UO_2 with an enrichment of 93.5 wt. % ^{235}U
Number of Rods Per Thoria Canister	$\leq$ 18
Decay Heat Per Thoria Canister	$\leq$ 115 watts
Post-Irradiation Fuel Cooling Time and Average Burnup Per Thoria Canister	Cooling time $\geq$ 18 years and average burnup $\leq$ 16,000 MWD/MTIHM
Initial Heavy Metal Weight	$\leq$ 27 kg/canister
Fuel Cladding O.D.	$\geq$ 0.412 inches
Fuel Cladding I.D.	$\leq$ 0.362 inches
Fuel Pellet O.D.	$\leq$ 0.358 inches
Active Fuel Length	$\leq$ 111 inches
Canister Weight	$\leq$ 550 lbs., including Thoria Rods
Canister Material	Type 304 SS

Table 2.1.13
MPC Fuel Loading Regions

MPC MODEL	REGION 1 FUEL STORAGE LOCATIONS*	REGION 2 FUEL STORAGE LOCATIONS
MPC-24, 24E and 24EF	9, 10, 15, and 16	All Other Locations
MPC-32/32F	7, 8, 12 through 15, 18 through 21, 25, and 26	All Other Locations
MPC-68/68F/68FF	11 through 14, 18 through 23, 27 through 32, 37 through 42, 46 through 51, 55 through 58	All Other Locations

*Note: Refer to Figures 1.2.2 through 1.2.4

Table 2.1.14

Soluble Boron Requirements for MPC-24/24E/24EF Fuel Wet Loading and Unloading Operations

MPC MODEL	**FUEL ASSEMBLY MAXIMUM AVERAGE ENRICHMENT (wt % ^{235}U)**	**MINIMUM SOLUBLE BORON CONCENTRATION (ppmb)**
MPC-24	All fuel assemblies with initial enrichment[1] less than the prescribed value for soluble boron credit	0
MPC-24	One or more fuel assemblies with an initial enrichment[1] greater than or equal to the prescribed value for no soluble boron credit and $\leq$ 5.0 wt. %	$\geq$ 400
MPC-24E/24EF	All fuel assemblies with initial enrichment[1] less than the prescribed value for soluble boron credit	0
MPC-24E/24EF	All fuel assemblies classified as intact fuel assemblies and one or more fuel assemblies with an initial enrichment[1] greater than or equal to the prescribed value for no soluble boron credit and $\leq$ 5.0 wt. %	$\geq$ 300
MPC-24E/24EF	One or more fuel assemblies classified as damaged fuel or fuel debris and one or more fuel assemblies with initial enrichment > 4.0 wt.% and $\leq$ 5.0 wt.%	$\geq$ 600

[1]Refer to Table 2.1.3 for these enrichments.

Table 2.1.15

MINIMUM BORAL ^{10}B LOADING IN NEUTRON ABSORBER PANELS

MPC MODEL	**MINIMUM ^{10}B LOADING (g/cm^2)**	
	Boral Neutron Absorber Panels	**METAMIC Neutron Absorber Panels**
MPC-24	0.0267	0.0223
MPC-24E and MPC-24EF	0.0372	0.0310
MPC-32/32F	0.0372	0.0310
MPC-68 and MPC-68FF	0.0372	0.0310
MPC-68F	0.01	N/A (Note 1)

Notes:

1. All MPC-68F canisters are equipped with Boral neutron absorber panels.

Table 2.1.16

Soluble Boron Requirements for MPC-32 and MPC-32F Wet Loading and Unloading Operations

Fuel Assembly Array/Class *(Note 1)*	**All Intact Fuel Assemblies**		**One or More Damaged Fuel Assemblies or Fuel Debris**	
	Initial Enrichment $\leq$ 4.1 wt.% ^{235}U (ppmb)	**Initial Enrichment > 4.1 wt.% and $\leq$ 5.0 wt.% ^{235}U (ppmb)**	**Initial Enrichment $\leq$ 4.1 wt.% ^{235}U (ppmb)**	**Initial Enrichment > 4.1 wt.% and $\leq$ 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	2,500	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

Notes:

1. *The soluble boron requirements for array/class 14x14E are specified in Supplement 2.II.*

Table 2.1.17

LIMITS FOR MATERIAL TO BE STORED IN MPC-24

PARAMETER	VALUE
Fuel Type	Uranium oxide, PWR intact fuel assemblies meeting the limits in Table 2.1.3 for the applicable array/class
Cladding Type	ZR or Stainless Steel (SS) as specified in Table 2.1.3 for the applicable array/class
Maximum Initial Enrichment per Assembly	As specified in Table 2.1.3 for the applicable array/class
Post-irradiation Cooling Time and Average Burnup per Assembly	ZR clad: As specified in Section 2.1.9.1 SS clad: ≥ 8 years and ≤ 40,000 MWD/MTU
Decay Heat Per Fuel Storage Location	ZR clad: As specified in Section 2.1.9.1 SS clad: ≤ 710 Watts
Non-Fuel Hardware Burnup and Cooling Time	As specified in Table 2.1.25
Fuel Assembly Length	≤ 176.8 in. (nominal design)
Fuel Assembly Width	≤ 8.54 in. (nominal design)
Fuel Assembly Weight	≤ 1,680 lbs (including non-fuel hardware)
Other Limitations	▪ Quantity is limited to up to 24 PWR intact fuel assemblies. ▪ Neutron sources, damaged fuel assemblies and fuel debris are not permitted for storage in MPC-24. ▪ BPRAs, TPDs, WABAs, water displacement guide tube plugs, orifice rod assemblies, and/or vibration suppressor inserts may be stored with fuel assemblies in any fuel cell location. ▪ CRAs, RCCAs, CEAs, and/or APSRs may be stored with fuel assemblies in fuel cell locations 9, 10, 15, and/or 16 ▪ Soluble boron requirements during wet loading and unloading are specified in Table 2.1.14.

Table 2.1.18

LIMITS FOR MATERIAL TO BE STORED IN MPC-68

PARAMETER	**VALUE (Note 1)**			
Fuel Type(s)	Uranium oxide, BWR intact fuel assemblies meeting the limits in Table 2.1.4 for the applicable array/class, with or without channels	Uranium oxide, BWR damaged fuel assemblies meeting the limits in Table 2.1.4 for the applicable array/class, with or without channels, placed in Damaged Fuel Containers (DFCs)	Mixed Oxide (MOX) BWR intact fuel assemblies meeting the limits in Table 2.1.4 for array/class 6x6B, with or without channels	Mixed Oxide (MOX) BWR damaged fuel assemblies meeting the limits in Table 2.1.4 for array/class 6x6B, with or without channels, placed in Damaged Fuel Containers (DFCs)
Cladding Type	ZR or Stainless Steel (SS) as specified in Table 2.1.4 for the applicable array/class	ZR or Stainless Steel (SS) as specified in Table 2.1.4 for the applicable array/class	ZR	ZR
Maximum Initial Planar-Average Enrichment per Assembly and Rod Enrichment	As specified in Table 2.1.4 for the applicable array/class	Planar Average: $\leq$ 2.7 wt% ^{235}U for array/classes 6x6A, 6x6C, 7x7A, and 8x8A; $\leq$ 4.0 wt% ^{235}U for all other array/classes Rod: As specified in Table 2.1.4	As specified in Table 2.1.4 for array/class 6x6B	As specified in Table 2.1.4 for array/class 6x6B

Table 2.1.18 (cont'd)

LIMITS FOR MATERIAL TO BE STORED IN MPC-68

PARAMETER	VALUE (Note 1)			
Post-irradiation Cooling Time and Average Burnup per Assembly	ZR clad: As specified in Section 2.1.9.1; except as provided in Notes 2 and 3. SS clad: Note 4	ZR clad: As specified in Section 2.1.9.1; except as provided in Notes 2 and 3. SS clad: Note 4.	Cooling time ≥ 18 years and average burnup ≤ 30,000 MWD/MTIHM.	Cooling time ≥ 18 years and average burnup ≤ 30,000 MWD/MTIHM.
Decay Heat Per Fuel Storage Location	*ZR clad:* As specified in Section 2.1.9.1; except as provided in Notes 2 and 3. SS clad: ≤ 95 Watts	*ZR clad:* As specified in Section 2.1.9.1; except as provided in Notes 2 and 3. SS clad: ≤ 95 Watts	≤ 115 Watts	≤ 115 Watts
Fuel Assembly Length	≤ 176.5 in. (nominal design)	Array/classes 6x6A, 6x6C, 7x7A, and 8x8A: ≤ 135.0 in. (nominal design) All Other array/classes: ≤ 176.5 in. (nominal design)	≤ 135.0 in. (nominal design)	≤ 135.0 in. (nominal design)
Fuel Assembly Width	≤ 5.85 in. (nominal design)	Array/classes 6x6A, 6x6C, 7x7A, and 8x8A: ≤ 4.7 in. (nominal design) All Other array/classes: ≤ 5.85 in. (nominal design)	≤ 4.70 in. (nominal design)	≤ 4.70 in. (nominal design)

Table 2.1.18 (cont'd)

LIMITS FOR MATERIAL TO BE STORED IN MPC-68

PARAMETER	VALUE (Note 1)			
Fuel Assembly Weight	≤ 700 lbs. (including channels)	Array/classes 6x6A, 6x6C, 7x7A, and 8x8A: ≤ 550 lbs. (including channels and DFC) All Other array/classes: ≤ 700 lbs. (including channels and DFC)	≤ 400 lbs, including channels	≤ 550 lbs, including channels and DFC
Other Limitations	▪ Quantity is limited to up to one (1) Dresden Unit 1 thoria rod canister meeting the specifications listed in Table 2.1.12 plus any combination of array/class 6x6A, 6x6B, 6x6C, 7x7A, and/or 8x8A damaged fuel assemblies in DFCs and intact fuel assemblies up to a total of 68. ▪ Up to 16 damaged fuel assemblies from plants other than Dresden Unit 1 or Humboldt Bay may be stored in DFCs in fuel cell locations 1, 2, 3, 8, 9, 16, 25, 34, 35, 44, 53, 60, 61, 66, 67, and/or 68, with the balance comprised of intact fuel assemblies up to a total of 68 ▪ SS-clad fuel assemblies with stainless steel channels must be stored in fuel cell locations 19 through 22, 28 through 31, 38 through 41, and/or 47 through 50. ▪ Dresden Unit 1 fuel assemblies with one antimony-beryllium neutron source are permitted. The antimony-beryllium neutron source material shall be in a water rod location. ▪ Fuel debris is not permitted for storage in MPC-68.			

Notes:

1. A fuel assembly must meet the requirements of any one column and the other limitations to be authorized for storage.

2. Array/class 6x6A, 6x6C, 7x7A, and 8x8A fuel assemblies shall have a cooling time ≥ 18 years, an average burnup ≤ 30,000 MWD/MTU, and a maximum decay heat ≤ 115 Watts.

3. Array/class 8x8F fuel assemblies shall have a cooling time ≥ 10 years, an average burnup ≤ 27,500 MWD/MTU, and a maximum decay ≤ 183.5 Watts.

4. SS-clad fuel assemblies shall have a cooling time ≥ 10 years, and an average burnup ≤ 22,500 MWD/MTU.

Table 2.1.19

LIMITS FOR MATERIAL TO BE STORED IN MPC-68F

PARAMETER	VALUE (Notes 1 and 2)			
Fuel Type(s)	Uranium oxide, BWR intact fuel assemblies meeting the limits in Table 2.1.4 for array/class 6x6A, 6x6C, 7x7A, or 8x8A, with or without Zircaloy channels	Uranium oxide, BWR damaged fuel assemblies or fuel debris meeting the limits in Table 2.1.4 for array/class 6x6A, 6x6C, 7x7A, or 8x8A, with or without Zircaloy channels, placed in Damaged Fuel Containers (DFCs)	Mixed Oxide (MOX) BWR intact fuel assemblies meeting the limits in Table 2.1.4 for array/class 6x6B, with or without Zircaloy channels	Mixed Oxide (MOX) BWR damaged fuel assemblies or fuel debris meeting the limits in Table 2.1.4 for array/class 6x6B, with or without Zircaloy channels, placed in Damaged Fuel Containers (DFCs))
Cladding Type	ZR	ZR	ZR	ZR
Maximum Initial Planar-Average Enrichment per Assembly and Rod Enrichment	As specified in Table 2.1.4 for the applicable array/class	As specified in Table 2.1.4 for the applicable array/class	As specified in Table 2.1.4 for array/class 6x6B	As specified in Table 2.1.4 for array/class 6x6B
Post-irradiation Cooling Time, Average Burnup, and Minimum Initial Enrichment per Assembly	Cooling time ≥ 18 years and average burnup ≤ 30,000 MWD/MTU.	Cooling time ≥ 18 years and average burnup ≤ 30,000 MWD/MTU.	Cooling time ≥ 18 years and average burnup ≤ 30,000 MWD/MTIHM.	Cooling time ≥ 18 years and average burnup ≤ 30,000 MWD/MTIHM.
Decay Heat Per Fuel Storage Location	≤ 115 Watts	≤ 115 Watts	≤ 115 Watts	≤ 115 Watts
Fuel Assembly Length	≤ 135.0 in. (nominal design)	≤ 135.0 in. (nominal design)	≤ 135.0 in. (nominal design)	≤ 135.0 in. (nominal design)
Fuel Assembly Width	≤ 4.70 in. (nominal design)	≤ 4.70 in. (nominal design)	≤ 4.70 in. (nominal design)	≤ 4.70 in. (nominal design)
Fuel Assembly Weight	≤ 400 lbs, (including channels)	≤ 550 lbs, (including channels and DFC)	≤ 400 lbs, (including channels)	≤ 550 lbs, (including channels and DFC)

Table 2.1.19 (cont'd)

LIMITS FOR MATERIAL TO BE STORED IN MPC-68F

PARAMETER	VALUE
Other Limitations	▪ Quantity is limited to up to four (4) DFCs containing Dresden Unit 1 or Humboldt Bay uranium oxide or MOX fuel debris. The remaining fuel storage locations may be filled with array/class 6x6A, 6x6B, 6x6C, 7x7A, and 8x8A fuel assemblies of the following type, as applicable: - uranium oxide BWR intact fuel assemblies - MOX BWR intact fuel assemblies - uranium oxide BWR damaged fuel assemblies in DFCs - MOX BWR damaged fuel assemblies in DFCs - up to one (1) Dresden Unit 1 thoria rod canister meeting the specifications listed in Table 2.1.12. ▪ Stainless steel channels are not permitted. ▪ Dresden Unit 1 fuel assemblies with one antimony-beryllium neutron source are permitted. The antimony-beryllium neutron source material shall be in a water rod location.

Notes:

1. A fuel assembly must meet the requirements of any one column and the other limitations to be authorized for storage.

2. Only fuel from the Dresden Unit 1 and Humboldt Bay plants are permitted for storage in the MPC-68F.

Table 2.1.20

LIMITS FOR MATERIAL TO BE STORED IN MPC-24E

PARAMETER	**VALUE (Note 1)**	
Fuel Type	Uranium oxide PWR intact fuel assemblies meeting the limits in Table 2.1.3 for the applicable array/class	Uranium oxide PWR damaged fuel assemblies meeting the limits in Table 2.1.3 for the applicable array/class, placed in a Damaged Fuel Container (DFC)
Cladding Type	ZR or Stainless Steel (SS) assemblies as specified in Table 2.1.3 for the applicable array/class	ZR or Stainless Steel (SS) assemblies as specified in Table 2.1.3 for the applicable array/class
Maximum Initial Enrichment per Assembly	As specified in Table 2.1.3 for the applicable array/class	As specified in Table 2.1.3 for the applicable array/class
Post-irradiation Cooling Time, and Average Burnup per Assembly	ZR clad: As specified in Section 2.1.9.1 SS clad: $\geq$ 8 yrs and $\leq$ 40,000 MWD/MTU	ZR clad: As specified in Section 2.1.9.1 SS clad: $\geq$ 8 yrs and $\leq$ 40,000 MWD/MTU
Decay Heat Per Fuel Storage Location	ZR clad: As specified in Section 2.1.9.1 SS clad: $\leq$ 710 Watts	ZR clad: As specified in Section 2.1.9.1 SS clad: $\leq$ 710 Watts
Non-fuel hardware post-irradiation Cooling Time and Burnup	As specified in Table 2.1.25	As specified in Table 2.1.25
Fuel Assembly Length	$\leq$ 176.8 in. (nominal design)	$\leq$ 176.8 in. (nominal design)
Fuel Assembly Width	$\leq$ 8.54 in. (nominal design)	$\leq$ 8.54 in. (nominal design)
Fuel Assembly Weight	$\leq$ 1680 lbs (including non-fuel hardware)	$\leq$ 1680 lbs (including DFC and non-fuel hardware)

Table 2.1.20 (cont'd)

LIMITS FOR MATERIAL TO BE STORED IN MPC-24E

PARAMETER	VALUE
Other Limitations	▪ Quantity is limited to up to 24 PWR intact fuel assemblies or up to four (4) damaged fuel assemblies in DFCs may be stored in fuel storage locations 3, 6, 19, and/or 22. The remaining fuel storage locations may be filled with intact fuel assemblies. ▪ Fuel debris and neutron sources are not authorized for storage in the MPC-24E. ▪ BPRAs, TPDs, WABAs, water displacement guide tube plugs, orifice rod assemblies, and/or vibration suppressor inserts may be stored with fuel assemblies in any fuel cell location. ▪ CRAs, RCCAs, CEAs, and/or APSRs may be stored with fuel assemblies in fuel cell locations 9, 10, 15, and/or 16. ▪ Soluble boron requirements during wet loading and unloading are specified in Table 2.1.14.

Notes:

1. A fuel assembly must meet the requirements of any one column and the other limitations to be authorized for storage.

Table 2.1.21

LIMITS FOR MATERIAL TO BE STORED IN MPC-32

PARAMETER	VALUE (Notes 1 *and 2*)	
Fuel Type	Uranium oxide, PWR intact fuel assemblies meeting the limits in Table 2.1.3 for the applicable fuel assembly array/class.	Uranium oxide, PWR damaged fuel assemblies meeting the limits in Table 2.1.3 for the applicable fuel assembly array/class.
Cladding Type	ZR or Stainless Steel (SS) assemblies as specified in Table 2.1.3 for the applicable array/class	ZR or Stainless Steel (SS) assemblies as specified in Table 2.1.3 for the applicable array/class
Maximum Initial Enrichment per Assembly	As specified in Table 2.1.3 for the applicable fuel assembly array/class	As specified in Table 2.1.3 for the applicable fuel assembly array/class
Post-irradiation Cooling Time and Average Burnup per Assembly	ZR clad: As specified in Section 2.1.9.1 SS clad: $\geq$ 9 years and $\leq$ 30,000 MWD/MTU or $\geq$ 20 years and $\leq$ 40,000 MWD/MTU	ZR clad: As specified in Section 2.1.9.1 SS clad: $\geq$ 9 years and $\leq$ 30,000 MWD/MTU or $\geq$ 20 years and $\leq$ 40,000 MWD/MTU
Decay Heat Per Fuel Storage Location	ZR-clad: As specified in Section 2.1.9.1 SS-clad: $\leq$ 500 Watts	ZR-clad: As specified in Section 2.1.9.1 SS-clad: $\leq$ 500 Watts
Non-fuel hardware post-irradiation cooling time and burnup	As specified in Table 2.1.25	As specified in Table 2.1.25
Fuel Assembly Length	$\leq$ 176.8 in. (nominal design)	$\leq$ 176.8 in. (nominal design)
Fuel Assembly Width	$\leq$ 8.54 in. (nominal design)	$\leq$ 8.54 in. (nominal design)
Fuel Assembly Weight	$\leq$ 1,680 lbs (including non-fuel hardware)	$\leq$ 1,680 lbs (including DFC and non-fuel hardware)

Table 2.1.21 (cont'd)

LIMITS FOR MATERIAL TO BE STORED IN MPC-32

PARAMETER	VALUE
Other Limits	▪ Quantity is limited to up to 32 PWR intact fuel assemblies and/or up to eight (8) damaged fuel assemblies in DFCs in fuel cell locations 1, 4, 5, 10, 23, 28, 29, and/or 32, with the balance intact fuel assemblies up to a total of 32. ▪ Fuel debris and neutron sources are not permitted for storage in MPC-32. ▪ BPRAs, TPDs, WABAs, water displacement guide tube plugs, orifice rod assemblies, and/or vibration suppressor inserts may be stored with fuel assemblies in any fuel cell location. ▪ CRAs, RCCAs, CEAs, and/or APSRs may be stored with fuel assemblies in fuel cell locations 13, 14, 19, and/or 20. ▪ Soluble boron requirements during wet loading and unloading are specified in Table 2.1.16.

NOTES:

1. A fuel assembly must meet the requirements of any one column and the other limitations to be authorized for storage.
2. *The requirements stated in this table, with the exception of fuel assembly length, width, and weight, do not apply to array/class 14x14E, Indian Point Unit 1 fuel. Supplement 2.II provides the limits for array/class 14x14E fuel assemblies to be stored in the MPC-32.*

Table 2.1.22

LIMITS FOR MATERIAL TO BE STORED IN MPC-68FF

PARAMETER	VALUE (Note 1)	
Fuel Type	Uranium oxide or MOX BWR intact fuel assemblies meeting the limits in Table 2.1.4 for the applicable array/class, with or without channels.	Uranium oxide or MOX BWR damaged fuel assemblies or fuel debris meeting the limits in Table 2.1.4 for the applicable array/class, with or without channels, in DFCs.
Cladding Type	ZR or Stainless Steel (SS) assemblies as specified in Table 2.1.4 for the applicable array/class	ZR or Stainless Steel (SS) assemblies as specified in Table 2.1.4 for the applicable array/class
Maximum Initial Planar Average Enrichment per Assembly and Rod Enrichment	As specified in Table 2.1.4 for the applicable fuel assembly array/class	Planar Average: ≤ 2.7 wt% ^{235}U for array/classes 6x6A, 6x6B, 6x6C, 7x7A, and 8x8A; ≤ 4.0 wt% ^{235}U for all other array/classes Rod: As specified in Table 2.1.4
Post-irradiation cooling time and average burnup per Assembly	ZR clad: As specified in Section 2.1.9.1; except as provided in Notes 2 and 3. SS clad: Note 4	ZR clad: As specified in Section 2.1.9.1; except as provided in Notes 2 and 3. SS clad: Note 4.
Decay Heat Per Fuel Storage Location	ZR clad: As specified in Section 2.1.9.1; except as provided in Notes 2 and 3. SS clad: ≤ 95 Watts	ZR clad: As specified in Section 2.1.9.1; except as provided in Notes 2 and 3. SS clad: ≤ 95 Watts
Fuel Assembly Length	Array/classes 6x6A, 6x6B, 6x6C, 7x7A, and 8x8A: ≤ 135.0 in. (nominal design) All Other array/classes: ≤ 176.5 in. (nominal design)	Array/classes 6x6A, 6x6B, 6x6C, 7x7A, and 8x8A: ≤ 135.0 in. (nominal design) All Other array/classes: ≤ 176.5 in. (nominal design)

Table 2.1.22 (cont'd)

LIMITS FOR MATERIAL TO BE STORED IN MPC-68FF

<table>
<tr><th>PARAMETER</th><th colspan="2">VALUE (Note 1)</th></tr>
<tr><td>Fuel Assembly Width</td><td>Array/classes 6x6A, 6x6B, 6x6C, 7x7A, and 8x8A: ≤ 4.7 in. (nominal design)

All Other array/classes:
≤ 5.85 in. (nominal design)</td><td>Array/classes 6x6A, 6x6B, 6x6C, 7x7A, and 8x8A: ≤ 4.7 in. (nominal design)

All Other array/classes:
≤ 5.85 in. (nominal design)</td></tr>
<tr><td>Fuel Assembly Weight</td><td>Array/classes 6x6A, 6x6B, 6x6C, 7x7A, and 8x8A: ≤ 550 lbs. (including channels)

All Other array/classes:
≤ 700 lbs. (including channels)</td><td>Array/classes 6x6A, 6x6B, 6x6C, 7x7A, and 8x8A: ≤ 550 lbs. (including channels and DFC)

All Other array/classes:
≤ 700 lbs. (including channels and DFC)</td></tr>
<tr><td>Other Limitations</td><td colspan="2">▪ Quantity is limited to up to one (1) Up to eight (8) Dresden Unit 1 or Humboldt Bay fuel assemblies classified as fuel debris in DFCs, and any combination of Dresden Unit 1 or Humboldt Bay damaged fuel assemblies in DFCs and intact fuel assemblies up to a total of 68.

▪ Up to 16 damaged fuel assemblies and/or up to eight (8) fuel assemblies classified as fuel debris from plants other than Dresden Unit 1 or Humboldt Bay may be stored in DFCs in MPC-68FF. DFCs shall be located only in fuel cell locations 1, 2, 3, 8, 9, 16, 25, 34, 35, 44, 53, 60, 61, 66, 67, and/or 68, with the balance comprised of intact fuel assemblies meeting the above specifications, up to a total of 68.

▪ SS-clad fuel assemblies with stainless steel channels must be stored in fuel cell locations 19 through 22, 28 through 31, 38 through 41, and/or 47 through 50.

▪ Dresden Unit 1 fuel assemblies with one antimony-beryllium neutron source are permitted. The antimony-beryllium neutron source material shall be in a water rod location.</td></tr>
</table>

NOTES:

1. A fuel assembly must meet the requirements of any one column and the other limitations to be authorized for storage.

2. Array/class 6x6A, 6x6B, 6x6C, 7x7A, and 8x8A fuel assemblies shall have a cooling time ≥ 18 years, an average burnup ≤ 30,000 MWD/MTU, and a maximum decay heat ≤ 115 Watts.

3. Array/class 8x8F fuel assemblies shall have a cooling time ≥ 10 years, an average burnup ≤ 27,500 MWD/MTU, and a maximum decay ≤ 183.5 Watts.

4. SS-clad fuel assemblies shall have a cooling time ≥ 10 years, and an average burnup ≤ 22,500 MWD/MTU.

Table 2.1.23

LIMITS FOR MATERIAL TO BE STORED IN MPC-24EF

PARAMETER	VALUE (Note 1)	
Fuel Type	Uranium oxide PWR intact fuel assemblies meeting the limits in Table 2.1.3 for the applicable array/class	Uranium oxide PWR damaged fuel assemblies and/or fuel debris meeting the limits in Table 2.1.3 for the applicable array/class, placed in a Damaged Fuel Container (DFC)
Cladding Type	ZR or Stainless Steel (SS) assemblies as specified in Table 2.1.3 for the applicable array/class	ZR or Stainless Steel (SS) assemblies as specified in Table 2.1.3 for the applicable array/class
Maximum Initial Enrichment per Assembly	As specified in Table 2.1.3 for the applicable array/class	As specified in Table 2.1.3 for the applicable array/class
Post-irradiation Cooling Time, and Average Burnup per Assembly	ZR clad: As specified in Section 2.1.9.1 SS clad: $\geq$ 8 yrs and $\leq$ 40,000 MWD/MTU	ZR clad: As specified in Section 2.1.9.1 SS clad: $\geq$ 8 yrs and $\leq$ 40,000 MWD/MTU
Decay Heat Per Fuel Storage Location	ZR clad: As specified in Section 2.1.9.1 SS clad: $\leq$ 710 Watts	ZR clad: As specified in Section 2.1.9.1 SS clad: $\leq$ 710 Watts
Non-fuel hardware post-irradiation Cooling Time and Burnup	As specified in Table 2.1.25	As specified in Table 2.1.25
Fuel Assembly Length	$\leq$ 176.8 in. (nominal design)	$\leq$ 176.8 in. (nominal design)
Fuel Assembly Width	$\leq$ 8.54 in. (nominal design)	$\leq$ 8.54 in. (nominal design)
Fuel Assembly Weight	$\leq$ 1680 lbs (including non-fuel hardware)	$\leq$ 1680 lbs (including DFC and non-fuel hardware)

Table 2.1.23 (cont'd)

LIMITS FOR MATERIAL TO BE STORED IN MPC-24EF

PARAMETER	VALUE
Other Limitations	▪ Quantity per MPC: up to 24 PWR intact fuel assemblies or up to four (4) damaged fuel assemblies and/or fuel classified as fuel debris in DFCs may be stored in fuel storage locations 3, 6, 19, and/or 22. The remaining fuel storage locations may be filled with intact fuel assemblies. ▪ Neutron sources are not authorized for storage in the MPC-24EF. ▪ BPRAs, TPDs, WABAs, water displacement guide tube plugs, orifice rod assemblies, and/or vibration suppressor inserts may be stored with fuel assemblies in any fuel cell location. ▪ CRAs, RCCAs, CEAs, and/or APSRs may be stored with fuel assemblies in fuel cell locations 9, 10, 15, and/or 16. ▪ Soluble boron requirements during wet loading and unloading are specified in Table 2.1.14.

Notes:

1. A fuel assembly must meet the requirements of any one column and the other limitations to be authorized for storage.

Table 2.1.24

LIMITS FOR MATERIAL TO BE STORED IN MPC-32F

PARAMETER	VALUE (Notes 1 *and 2*)	
Fuel Type	Uranium oxide, PWR intact fuel assemblies meeting the limits in Table 2.1.3 for the applicable fuel assembly array/class	Uranium oxide, PWR damaged fuel assemblies and fuel debris in DFCs meeting the limits in Table 2.1.3 for the applicable fuel assembly array/class
Cladding Type	ZR or Stainless Steel (SS) as specified in Table 2.1.3 for the applicable fuel assembly array/class	ZR or Stainless Steel (SS) as specified in Table 2.1.3 for the applicable fuel assembly array/class
Maximum Initial Enrichment per Assembly	As specified in Table 2.1.3	As specified in Table 2.1.3
Post-irradiation Cooling Time, Average Burnup, and Minimum Initial Enrichment per Assembly	ZR clad: As specified in Section 2.1.9.1 SS clad: ≥ 9 years and ≤ 30,000 MWD/MTU or ≥ 20 years and ≤ 40,000MWD/MTU	ZR clad: As specified in Section 2.1.9.1 SS clad: ≥ 9 years and ≤ 30,000 MWD/MTU or ≥ 20 years and ≤ 40,000MWD/MTU
Decay Heat Per Fuel Storage Location	ZR clad: As specified in Section 2.1.9.1 SS clad: ≤ 500 Watts	ZR clad: As specified in Section 2.1.9.1 SS clad: ≤ 500 Watts
Non-fuel hardware post-irradiation Cooling Time and Burnup	As specified in Table 2.1.25	As specified in Table 2.1.25
Fuel Assembly Length	≤ 176.8 in. (nominal design)	≤ 176.8 in. (nominal design)
Fuel Assembly Width	≤ 8.54 in. (nominal design)	≤ 8.54 in. (nominal design)
Fuel Assembly Weight	≤ 1,680 lbs (including non-fuel hardware)	≤ 1,680 lbs (including DFC and non-fuel hardware)

Table 2.1.24 (cont'd)

LIMITS FOR MATERIAL TO BE STORED IN MPC-32F

PARAMETER	VALUE
Other Limitations	▪ Quantity is limited to up to 32 PWR intact fuel assemblies and/or up to eight (8) damaged fuel assemblies in DFCs in fuel cell locations 1, 4, 5, 10, 23, 28, 29, and/or 32, with the balance intact fuel assemblies up to a total of 32. ▪ Neutron sources are not permitted for storage in MPC-32. ▪ BPRAs, TPDs, WABAs, water displacement guide tube plugs, orifice rod assemblies, and/or vibration suppressor inserts may be stored with fuel assemblies in any fuel cell location. ▪ CRAs, RCCAs, CEAs, and/or APSRs may be stored with fuel assemblies in fuel cell locations 13, 14, 19, and/or 20. ▪ Soluble boron requirements during wet loading and unloading are specified in Table 2.1.16.

NOTES:

1. A fuel assembly must meet the requirements of any one column and the other limitations to be authorized for storage.
2. *The requirements stated in this table, with the exception of fuel assembly length, width, and weight, do not apply to array/class 14x14E, Indian Point Unit 1 fuel. Supplement 2.II provides the limits for array/class 14x14E fuel assemblies to be stored in the MPC-32F.*

Table 2.1.25

NON-FUEL HARDWARE BURNUP AND COOLING TIME LIMITS (Notes 1, 2, and 3)

Post-irradiation Cooling Time (yrs)	**Inserts (Note 4) Maximum Burnup (MWD/MTU)**	**Guide Tube Hardware (Note 5) Maximum Burnup (MWD/MTU)**	**Control Component (Note 6) Maximum Burnup (MWD/MTU)**	**APSR Maximum Burnup (MWD/MTU)**
≥ 3	≤ 24,635	N/A (Note 7)	N/A	N/A
≥ 4	≤ 30,000	≤ 20,000	N/A	N/A
≥ 5	≤ 36,748	≤ 25,000	≤ 630,000	≤ 45,000
≥ 6	≤ 44,102	≤ 30,000	-	≤ 54,500
≥ 7	≤ 52,900	≤ 40,000	-	≤ 68,000
≥ 8	≤ 60,000	≤ 45,000	-	≤ 83,000
≥ 9	-	≤ 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. Burnups for non-fuel hardware are to be determined based on the burnup and uranium mass of the fuel assemblies in which the component was inserted during reactor operation.

2. 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.

3. Applicable to uniform loading and regionalized loading.

4. Includes Burnable Poison Rod Assemblies (BPRAs), Wet Annular Burnable Absorbers (WABAs), and vibration suppressor inserts.

5. Includes Thimble Plug Devices (TPDs), water displacement guide tube plugs, and orifice rod assemblies.

6. Includes Control Rod Assemblies (CRAs), Control Element Assemblies (CEAs), and Rod Cluster Control Assemblies (RCCAs).

7. N/A means not authorized for loading at this cooling time.

Table 2.1.26

MAXIMUM ALLOWABLE DECAY HEAT PER FUEL STORAGE LOCATION
(UNIFORM LOADING, ZR-CLAD)

MPC Model	**Decay Heat per Fuel Assembly (kW)**
Intact Fuel Assemblies	
MPC-24	$\leq$ 1.157
MPC-24E/24EF	$\leq$ 1.173
MPC-32/32F	$\leq$ 0.898
MPC-68/68FF	$\leq$ 0.414
Damaged Fuel Assemblies and Fuel Debris	
MPC-24	$\leq$ 1.099
MPC-24E/24EF	$\leq$ 1.114
MPC-32/32F	$\leq$ 0.718
MPC-68/68FF	$\leq$ 0.393

Table 2.1.27

MPC FUEL STORAGE REGIONS AND MAXIMUM DECAY HEAT

MPC Model	Number of Fuel Storage Locations in Inner and Outer Regions	Inner Region Maximum Decay Heat per Assembly (kW)	Outer Region Maximum Decay Heat per Assembly (kW)
MPC-24	4 and 20	1.470	0.900
MPC-24E/24EF	4 and 20	1.540	0.900
MPC-32/32F	12 and 20	1.131	0.600
MPC-68/68FF	32 and 36	0.500	0.275

Note: These limits apply to intact fuel assemblies, damaged fuel assemblies and fuel debris.

Table 2.1.28

PWR FUEL ASSEMBLY COOLING TIME-DEPENDENT COEFFICIENTS
(ZR-CLAD FUEL)

Cooling Time **(years)**	**Array/Class 14x14A**						
	A	**B**	**C**	**D**	**E**	**F**	**G**
≥ 3	20277.1	303.592	-68.329	-139.41	2993.67	-498.159	-615.411
≥ 4	35560.1	-6034.67	985.415	-132.734	3578.92	-723.721	-609.84
≥ 5	48917.9	-14499.5	2976.09	-150.707	4072.55	-892.691	-54.8362
≥ 6	59110.3	-22507	5255.61	-177.017	4517.03	-1024.01	613.36
≥ 7	67595.6	-30158.1	7746.6	-200.128	4898.71	-1123.21	716.004
≥ 8	74424.9	-36871.1	10169.4	-218.676	5203.64	-1190.24	741.163
≥ 9	81405.8	-44093.1	12910.8	-227.916	5405.34	-1223.27	250.224
≥ 10	86184.3	-49211.7	15063.4	-237.641	5607.96	-1266.21	134.435
≥ 11	92024.9	-55666.8	17779.6	-240.973	5732.25	-1282.12	-401.456
≥ 12	94775.8	-58559.7	19249.9	-246.369	5896.27	-1345.42	-295.435
≥ 13	100163	-64813.8	22045.1	-242.572	5861.86	-1261.66	-842.159
≥ 14	103971	-69171	24207	-242.651	5933.96	-1277.48	-1108.99
≥ 15	108919	-75171.1	27152.4	-243.154	6000.2	-1301.19	-1620.63
≥ 16	110622	-76715.2	28210.2	-240.235	6028.33	-1307.74	-1425.5
≥ 17	115582	-82929.7	31411.9	-235.234	5982.3	-1244.11	-1948.05
≥ 18	119195	-87323.5	33881.4	-233.28	6002.43	-1245.95	-2199.41
≥ 19	121882	-90270.6	35713.7	-231.873	6044.42	-1284.55	-2264.05
≥ 20	124649	-93573.5	37853.1	-230.22	6075.82	-1306.57	-2319.63

Table 2.1.28 (cont'd)

PWR FUEL ASSEMBLY COOLING TIME-DEPENDENT COEFFICIENTS
(ZR-CLAD FUEL)

Cooling Time **(years)**	**Array/Class 14x14B**						
	A	**B**	**C**	**D**	**E**	**F**	**G**
≥ 3	18937.9	70.2997	-28.6224	-130.732	2572.36	-383.393	-858.17
≥ 4	32058.7	-4960.63	745.224	-125.978	3048.98	-551.656	-549.108
≥ 5	42626.3	-10804.1	1965.09	-139.722	3433.49	-676.643	321.88
≥ 6	51209.6	-16782.3	3490.45	-158.929	3751.01	-761.524	847.282
≥ 7	57829.9	-21982	5009.12	-180.026	4066.65	-846.272	1200.45
≥ 8	62758	-26055.3	6330.88	-196.804	4340.18	-928.336	1413.17
≥ 9	68161.4	-30827.6	7943.87	-204.454	4500.52	-966.347	1084.69
≥ 10	71996.8	-34224.3	9197.25	-210.433	4638.94	-1001.83	1016.38
≥ 11	75567.3	-37486.1	10466.9	-214.95	4759.55	-1040.85	848.169
≥ 12	79296.7	-40900.3	11799.6	-212.898	4794.13	-1040.51	576.242
≥ 13	82257.3	-43594	12935	-212.8	4845.81	-1056.01	410.807
≥ 14	83941.2	-44915.2	13641	-215.389	4953.19	-1121.71	552.724
≥ 15	87228.5	-48130	15056.9	-212.545	4951.12	-1112.5	260.194
≥ 16	90321.7	-50918.3	16285.5	-206.094	4923.36	-1106.35	-38.7487
≥ 17	92836.2	-53314.5	17481.7	-203.139	4924.61	-1109.32	-159.673
≥ 18	93872.8	-53721.4	17865.1	-202.573	4956.21	-1136.9	30.0594
≥ 19	96361.6	-56019.1	19075.9	-199.068	4954.59	-1156.07	-125.917
≥ 20	98647.5	-57795.1	19961.8	-191.502	4869.59	-1108.74	-217.603

Table 2.1.28 (cont'd)

PWR FUEL ASSEMBLY COOLING TIME-DEPENDENT COEFFICIENTS
(ZR-CLAD FUEL)

Cooling Time **(years)**	**Array/Class 14x14C**						
	A	**B**	**C**	**D**	**E**	**F**	**G**
≥ 3	19176.9	192.012	-66.7595	-138.112	2666.73	-407.664	-1372.41
≥ 4	32040.3	-4731.4	651.014	-124.944	3012.63	-530.456	-890.059
≥ 5	43276.7	-11292.8	2009.76	-142.172	3313.91	-594.917	-200.195
≥ 6	51315.5	-16920.5	3414.76	-164.287	3610.77	-652.118	463.041
≥ 7	57594.7	-21897.6	4848.49	-189.606	3940.67	-729.367	781.46
≥ 8	63252.3	-26562.8	6273.01	-199.974	4088.41	-732.054	693.879
≥ 9	67657.5	-30350.9	7533.4	-211.77	4283.39	-772.916	588.456
≥ 10	71834.4	-34113.7	8857.32	-216.408	4383.45	-774.982	380.243
≥ 11	75464.1	-37382.1	10063	-218.813	4460.69	-776.665	160.668
≥ 12	77811.1	-39425.1	10934.3	-225.193	4604.68	-833.459	182.463
≥ 13	81438.3	-42785.4	12239.9	-220.943	4597.28	-803.32	-191.636
≥ 14	84222.1	-45291.6	13287.9	-218.366	4608.13	-791.655	-354.59
≥ 15	86700.1	-47582.6	14331.2	-218.206	4655.34	-807.366	-487.316
≥ 16	88104.7	-48601.1	14927.9	-219.498	4729.97	-849.446	-373.196
≥ 17	91103.3	-51332.5	16129	-212.138	4679.91	-822.896	-654.296
≥ 18	93850.4	-53915.8	17336.9	-207.666	4652.65	-799.697	-866.307
≥ 19	96192.9	-55955.8	18359.3	-203.462	4642.65	-800.315	-1007.75
≥ 20	97790.4	-57058.1	19027.7	-200.963	4635.88	-799.721	-951.122

Table 2.1.28 (cont'd)

PWR FUEL ASSEMBLY COOLING TIME-DEPENDENT COEFFICIENTS
(ZR-CLAD FUEL)

Cooling Time **(years)**	**Array/Class 15x15A/B/C**						
	A	**B**	**C**	**D**	**E**	**F**	**G**
≥ 3	15789.2	119.829	-21.8071	-127.422	2152.53	-267.717	-580.768
≥ 4	26803.8	-3312.93	415.027	-116.279	2550.15	-386.33	-367.168
≥ 5	36403.6	-7831.93	1219.66	-126.065	2858.32	-471.785	326.863
≥ 6	44046.1	-12375.9	2213.52	-145.727	3153.45	-539.715	851.971
≥ 7	49753.5	-16172.6	3163.61	-166.946	3428.38	-603.598	1186.31
≥ 8	55095.4	-20182.5	4287.03	-183.047	3650.42	-652.92	1052.4
≥ 9	58974.4	-23071.6	5156.53	-191.718	3805.41	-687.18	1025
≥ 10	62591.8	-25800.8	5995.95	-195.105	3884.14	-690.659	868.556
≥ 11	65133.1	-27747.4	6689	-203.095	4036.91	-744.034	894.607
≥ 12	68448.4	-30456	7624.9	-202.201	4083.52	-753.391	577.914
≥ 13	71084.4	-32536.4	8381.78	-201.624	4117.93	-757.16	379.105
≥ 14	73459.5	-34352.3	9068.86	-197.988	4113.16	-747.015	266.536
≥ 15	75950.7	-36469.4	9920.52	-199.791	4184.91	-779.222	57.9429
≥ 16	76929.1	-36845.6	10171.3	-197.88	4206.24	-794.541	256.099
≥ 17	79730	-39134.8	11069.4	-190.865	4160.42	-773.448	-42.6853
≥ 18	81649.2	-40583	11736.1	-187.604	4163.36	-785.838	-113.614
≥ 19	83459	-41771.8	12265.9	-181.461	4107.51	-758.496	-193.442
≥ 20	86165.4	-44208.8	13361.2	-178.89	4107.62	-768.671	-479.778

Table 2.1.28 (cont'd)

PWR FUEL ASSEMBLY COOLING TIME-DEPENDENT COEFFICIENTS
(ZR-CLAD FUEL)

Cooling Time **(years)**	**Array/Class 15x15D/E/F/H**						
	A	**B**	**C**	**D**	**E**	**F**	**G**
≥ 3	15192.5	50.5722	-12.3042	-126.906	2009.71	-235.879	-561.574
≥ 4	25782.5	-3096.5	369.096	-113.289	2357.75	-334.695	-254.964
≥ 5	35026.5	-7299.87	1091.93	-124.619	2664	-414.527	470.916
≥ 6	42234.9	-11438.4	1967.63	-145.948	2945.81	-474.981	1016.84
≥ 7	47818.4	-15047	2839.22	-167.273	3208.95	-531.296	1321.12
≥ 8	52730.7	-18387.2	3702.43	-175.057	3335.58	-543.232	1223.61
≥ 9	56254.6	-20999.9	4485.93	-190.489	3547.98	-600.64	1261.55
≥ 10	59874.6	-23706.5	5303.88	-193.807	3633.01	-611.892	1028.63
≥ 11	62811	-25848.4	5979.64	-194.997	3694.14	-618.968	862.738
≥ 12	65557.6	-27952.4	6686.74	-198.224	3767.28	-635.126	645.139
≥ 13	67379.4	-29239.2	7197.49	-200.164	3858.53	-677.958	652.601
≥ 14	69599.2	-30823.8	7768.51	-196.788	3868.2	-679.88	504.443
≥ 15	71806.7	-32425	8360.38	-191.935	3851.65	-669.917	321.146
≥ 16	73662.6	-33703.5	8870.78	-187.366	3831.59	-658.419	232.335
≥ 17	76219.8	-35898.1	9754.72	-189.111	3892.07	-694.244	-46.924
≥ 18	76594.4	-35518.2	9719.78	-185.11	3897.04	-712.82	236.047
≥ 19	78592.7	-36920.8	10316.5	-179.54	3865.84	-709.551	82.478
≥ 20	80770.5	-38599.9	11051.3	-175.106	3858.67	-723.211	-116.014

Table 2.1.28 (cont'd)

PWR FUEL ASSEMBLY COOLING TIME-DEPENDENT COEFFICIENTS
(ZR-CLAD FUEL)

Cooling Time **(years)**	**Array/Class 16x16A**						
	A	**B**	**C**	**D**	**E**	**F**	**G**
≥ 3	17038.2	158.445	-37.6008	-136.707	2368.1	-321.58	-700.033
≥ 4	29166.3	-3919.95	508.439	-125.131	2782.53	-455.722	-344.199
≥ 5	40285	-9762.36	1629.72	-139.652	3111.83	-539.804	139.67
≥ 6	48335.7	-15002.6	2864.09	-164.702	3444.97	-614.756	851.706
≥ 7	55274.9	-20190	4258.03	-185.909	3728.11	-670.841	920.035
≥ 8	60646.6	-24402.4	5483.54	-199.014	3903.29	-682.26	944.913
≥ 9	64663.2	-27753.1	6588.21	-215.318	4145.34	-746.822	967.914
≥ 10	69306.9	-31739.1	7892.13	-218.898	4237.04	-746.815	589.277
≥ 11	72725.8	-34676.6	8942.26	-220.836	4312.93	-750.85	407.133
≥ 12	76573.8	-38238.7	10248.1	-224.934	4395.85	-757.914	23.7549
≥ 13	78569	-39794.3	10914.9	-224.584	4457	-776.876	69.428
≥ 14	81559.4	-42453.6	11969.6	-222.704	4485.28	-778.427	-203.031
≥ 15	84108.6	-44680.4	12897.8	-218.387	4460	-746.756	-329.078
≥ 16	86512.2	-46766.8	13822.8	-216.278	4487.79	-759.882	-479.729
≥ 17	87526.7	-47326.2	14221	-218.894	4567.68	-805.659	-273.692
≥ 18	90340.3	-49888.6	15349.8	-212.139	4506.29	-762.236	-513.316
≥ 19	93218.2	-52436.7	16482.4	-207.653	4504.12	-776.489	-837.1
≥ 20	95533.9	-54474.1	17484.2	-203.094	4476.21	-760.482	-955.662

Table 2.1.28 (cont'd)

PWR FUEL ASSEMBLY COOLING TIME-DEPENDENT COEFFICIENTS
(ZR-CLAD FUEL)

Cooling Time **(years)**	**Array/Class 17x17A**						
	A	**B**	**C**	**D**	**E**	**F**	**G**
≥ 3	16784.4	3.90244	-10.476	-128.835	2256.98	-287.108	-263.081
≥ 4	28859	-3824.72	491.016	-120.108	2737.65	-432.361	-113.457
≥ 5	40315.9	-9724	1622.89	-140.459	3170.28	-547.749	425.136
≥ 6	49378.5	-15653.1	3029.25	-164.712	3532.55	-628.93	842.73
≥ 7	56759.5	-21320.4	4598.78	-190.58	3873.21	-698.143	975.46
≥ 8	63153.4	-26463.8	6102.47	-201.262	4021.84	-685.431	848.497
≥ 9	67874.9	-30519.2	7442.84	-218.184	4287.23	-754.597	723.305
≥ 10	72676.8	-34855.2	8928.27	-222.423	4382.07	-741.243	387.877
≥ 11	75623	-37457.1	9927.65	-232.962	4564.55	-792.051	388.402
≥ 12	80141.8	-41736.5	11509.8	-232.944	4624.72	-787.134	-164.727
≥ 13	83587.5	-45016.4	12800.9	-230.643	4623.2	-745.177	-428.635
≥ 14	86311.3	-47443.4	13815.2	-228.162	4638.89	-729.425	-561.758
≥ 15	87839.2	-48704.1	14500.3	-231.979	4747.67	-775.801	-441.959
≥ 16	91190.5	-51877.4	15813.2	-225.768	4692.45	-719.311	-756.537
≥ 17	94512	-55201.2	17306.1	-224.328	4740.86	-747.11	-1129.15
≥ 18	96959	-57459.9	18403.8	-220.038	4721.02	-726.928	-1272.47
≥ 19	99061.1	-59172.1	19253.1	-214.045	4663.37	-679.362	-1309.88
≥ 20	100305	-59997.5	19841.1	-216.112	4721.71	-705.463	-1148.45

Table 2.1.28 (cont'd)

PWR FUEL ASSEMBLY COOLING TIME-DEPENDENT COEFFICIENTS
(ZR-CLAD FUEL)

Cooling Time **(years)**	**Array/Class 17x17B/C**						
	A	**B**	**C**	**D**	**E**	**F**	**G**
≥ 3	15526.8	18.0364	-9.36581	-128.415	2050.81	-243.915	-426.07
≥ 4	26595.4	-3345.47	409.264	-115.394	2429.48	-350.883	-243.477
≥ 5	36190.4	-7783.2	1186.37	-130.008	2769.53	-438.716	519.95
≥ 6	44159	-12517.5	2209.54	-150.234	3042.25	-489.858	924.151
≥ 7	50399.6	-16780.6	3277.26	-173.223	3336.58	-555.743	1129.66
≥ 8	55453.9	-20420	4259.68	-189.355	3531.65	-581.917	1105.62
≥ 9	59469.3	-23459.8	5176.62	-199.63	3709.99	-626.667	1028.74
≥ 10	63200.5	-26319.6	6047.8	-203.233	3783.02	-619.949	805.311
≥ 11	65636.3	-28258.3	6757.23	-214.247	3972.8	-688.56	843.457
≥ 12	68989.7	-30904.4	7626.53	-212.539	3995.62	-678.037	495.032
≥ 13	71616.6	-32962.2	8360.45	-210.386	4009.11	-666.542	317.009
≥ 14	73923.9	-34748	9037.75	-207.668	4020.13	-662.692	183.086
≥ 15	76131.8	-36422.3	9692.32	-203.428	4014.55	-655.981	47.5234
≥ 16	77376.5	-37224.7	10111.4	-207.581	4110.76	-703.37	161.128
≥ 17	80294.9	-39675.9	11065.9	-201.194	4079.24	-691.636	-173.782
≥ 18	82219.8	-41064.8	11672.1	-195.431	4043.83	-675.432	-286.059
≥ 19	84168.9	-42503.6	12309.4	-190.602	4008.19	-656.192	-372.411
≥ 20	86074.2	-43854.4	12935.9	-185.767	3985.57	-656.72	-475.953

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Table 2.1.29

BWR FUEL ASSEMBLY COOLING TIME-DEPENDENT COEFFICIENTS
(ZR-CLAD FUEL)

Cooling Time **(years)**	**Array/Class 7x7B**						
	A	**B**	**C**	**D**	**E**	**F**	**G**
≥ 3	26409.1	28347.5	-16858	-147.076	5636.32	-1606.75	1177.88
≥ 4	61967.8	-6618.31	-4131.96	-113.949	6122.77	-2042.85	-96.7439
≥ 5	91601.1	-49298.3	17826.5	-132.045	6823.14	-2418.49	-185.189
≥ 6	111369	-80890.1	35713.8	-150.262	7288.51	-2471.1	86.6363
≥ 7	126904	-108669	53338.1	-167.764	7650.57	-2340.78	150.403
≥ 8	139181	-132294	69852.5	-187.317	8098.66	-2336.13	97.5285
≥ 9	150334	-154490	86148.1	-193.899	8232.84	-2040.37	-123.029
≥ 10	159897	-173614	100819	-194.156	8254.99	-1708.32	-373.605
≥ 11	166931	-186860	111502	-193.776	8251.55	-1393.91	-543.677
≥ 12	173691	-201687	125166	-202.578	8626.84	-1642.3	-650.814
≥ 13	180312	-215406	137518	-201.041	8642.19	-1469.45	-810.024
≥ 14	185927	-227005	148721	-197.938	8607.6	-1225.95	-892.876
≥ 15	191151	-236120	156781	-191.625	8451.86	-846.27	-1019.4
≥ 16	195761	-244598	165372	-187.043	8359.19	-572.561	-1068.19
≥ 17	200791	-256573	179816	-197.26	8914.28	-1393.37	-1218.63
≥ 18	206068	-266136	188841	-187.191	8569.56	-730.898	-1363.79
≥ 19	210187	-273609	197794	-182.151	8488.23	-584.727	-1335.59
≥ 20	213731	-278120	203074	-175.864	8395.63	-457.304	-1364.38

Table 2.1.29 (cont'd)

BWR FUEL ASSEMBLY COOLING TIME-DEPENDENT COEFFICIENTS
(ZR-CLAD FUEL)

Cooling Time **(years)**	**Array/Class 8x8B**						
	A	**B**	**C**	**D**	**E**	**F**	**G**
≥ 3	28219.6	28963.7	-17616.2	-147.68	5887.41	-1730.96	1048.21
≥ 4	66061.8	-10742.4	-1961.82	-123.066	6565.54	-2356.05	-298.005
≥ 5	95790.7	-53401.7	19836.7	-134.584	7145.41	-2637.09	-298.858
≥ 6	117477	-90055.9	41383.9	-154.758	7613.43	-2612.69	-64.9921
≥ 7	134090	-120643	60983	-168.675	7809	-2183.3	-40.8885
≥ 8	148186	-149181	81418.7	-185.726	8190.07	-2040.31	-260.773
≥ 9	159082	-172081	99175.2	-197.185	8450.86	-1792.04	-381.705
≥ 10	168816	-191389	113810	-195.613	8359.87	-1244.22	-613.594
≥ 11	177221	-210599	131099	-208.3	8810	-1466.49	-819.773
≥ 12	183929	-224384	143405	-207.497	8841.33	-1227.71	-929.708
≥ 13	191093	-240384	158327	-204.95	8760.17	-811.708	-1154.76
≥ 14	196787	-252211	169664	-204.574	8810.95	-610.928	-1208.97
≥ 15	203345	-267656	186057	-208.962	9078.41	-828.954	-1383.76
≥ 16	207973	-276838	196071	-204.592	9024.17	-640.808	-1436.43
≥ 17	213891	-290411	211145	-202.169	9024.19	-482.1	-1595.28
≥ 18	217483	-294066	214600	-194.243	8859.35	-244.684	-1529.61
≥ 19	220504	-297897	219704	-190.161	8794.97	-10.9863	-1433.86
≥ 20	227821	-318395	245322	-194.682	9060.96	-350.308	-1741.16

Table 2.1.29 (cont'd)

BWR FUEL ASSEMBLY COOLING TIME-DEPENDENT COEFFICIENTS
(ZR-CLAD FUEL)

Cooling Time **(years)**	**Array/Class 8x8C/D/E**						
	A	**B**	**C**	**D**	**E**	**F**	**G**
≥ 3	28592.7	28691.5	-17773.6	-149.418	5969.45	-1746.07	1063.62
≥ 4	66720.8	-12115.7	-1154	-128.444	6787.16	-2529.99	-302.155
≥ 5	96929.1	-55827.5	21140.3	-136.228	7259.19	-2685.06	-334.328
≥ 6	118190	-92000.2	42602.5	-162.204	7907.46	-2853.42	-47.5465
≥ 7	135120	-123437	62827.1	-172.397	8059.72	-2385.81	-75.0053
≥ 8	149162	-152986	84543.1	-195.458	8559.11	-2306.54	-183.595
≥ 9	161041	-177511	103020	-200.087	8632.84	-1864.4	-433.081
≥ 10	171754	-201468	122929	-209.799	8952.06	-1802.86	-755.742
≥ 11	179364	-217723	137000	-215.803	9142.37	-1664.82	-847.268
≥ 12	186090	-232150	150255	-216.033	9218.36	-1441.92	-975.817
≥ 13	193571	-249160	165997	-213.204	9146.99	-1011.13	-1119.47
≥ 14	200034	-263671	180359	-210.559	9107.54	-694.626	-1312.55
≥ 15	205581	-275904	193585	-216.242	9446.57	-1040.65	-1428.13
≥ 16	212015	-290101	207594	-210.036	9212.93	-428.321	-1590.7
≥ 17	216775	-299399	218278	-204.611	9187.86	-398.353	-1657.6
≥ 18	220653	-306719	227133	-202.498	9186.34	-181.672	-1611.86
≥ 19	224859	-314004	235956	-193.902	8990.14	145.151	-1604.71
≥ 20	228541	-320787	245449	-200.727	9310.87	-230.252	-1570.18

Table 2.1.29 (cont'd)

BWR FUEL ASSEMBLY COOLING TIME-DEPENDENT COEFFICIENTS
(ZR-CLAD FUEL)

Cooling Time **(years)**	**Array/Class 9x9A**						
	A	**B**	**C**	**D**	**E**	**F**	**G**
≥ 3	30538.7	28463.2	-18105.5	-150.039	6226.92	-1876.69	1034.06
≥ 4	71040.1	-16692.2	1164.15	-128.241	7105.27	-2728.58	-414.09
≥ 5	100888	-60277.7	24150.1	-142.541	7896.11	-3272.86	-232.197
≥ 6	124846	-102954	50350.8	-161.849	8350.16	-3163.44	-91.1396
≥ 7	143516	-140615	76456.5	-185.538	8833.04	-2949.38	-104.802
≥ 8	158218	-171718	99788.2	-196.315	9048.88	-2529.26	-259.929
≥ 9	172226	-204312	126620	-214.214	9511.56	-2459.19	-624.954
≥ 10	182700	-227938	146736	-215.793	9555.41	-1959.92	-830.943
≥ 11	190734	-246174	163557	-218.071	9649.43	-1647.5	-935.021
≥ 12	199997	-269577	186406	-223.975	9884.92	-1534.34	-1235.27
≥ 13	207414	-287446	204723	-228.808	10131.7	-1614.49	-1358.61
≥ 14	215263	-306131	223440	-220.919	9928.27	-988.276	-1638.05
≥ 15	221920	-321612	239503	-217.949	9839.02	-554.709	-1784.04
≥ 16	226532	-331778	252234	-216.189	9893.43	-442.149	-1754.72
≥ 17	232959	-348593	272609	-219.907	10126.3	-663.84	-1915.3
≥ 18	240810	-369085	296809	-219.729	10294.6	-859.302	-2218.87
≥ 19	244637	-375057	304456	-210.997	10077.8	-425.446	-2127.83
≥ 20	248112	-379262	309391	-204.191	9863.67	100.27	-2059.39

Table 2.1.29 (cont'd)

BWR FUEL ASSEMBLY COOLING TIME-DEPENDENT COEFFICIENTS
(ZR-CLAD FUEL)

Cooling Time **(years)**	**Array/Class 9x9B**						
	A	**B**	**C**	**D**	**E**	**F**	**G**
≥ 3	30613.2	28985.3	-18371	-151.117	6321.55	-1881.28	988.92
≥ 4	71346.6	-15922.9	631.132	-128.876	7232.47	-2810.64	-471.737
≥ 5	102131	-60654.1	23762.7	-140.748	7881.6	-3156.38	-417.979
≥ 6	127187	-105842	51525.2	-162.228	8307.4	-2913.08	-342.13
≥ 7	146853	-145834	79146.5	-185.192	8718.74	-2529.57	-484.885
≥ 8	162013	-178244	103205	-197.825	8896.39	-1921.58	-584.013
≥ 9	176764	-212856	131577	-215.41	9328.18	-1737.12	-1041.11
≥ 10	186900	-235819	151238	-218.98	9388.08	-1179.87	-1202.83
≥ 11	196178	-257688	171031	-220.323	9408.47	-638.53	-1385.16
≥ 12	205366	-280266	192775	-223.715	9592.12	-472.261	-1661.6
≥ 13	215012	-306103	218866	-231.821	9853.37	-361.449	-1985.56
≥ 14	222368	-324558	238655	-228.062	9834.57	3.47358	-2178.84
≥ 15	226705	-332738	247316	-224.659	9696.59	632.172	-2090.75
≥ 16	233846	-349835	265676	-221.533	9649.93	913.747	-2243.34
≥ 17	243979	-379622	300077	-222.351	9792.17	1011.04	-2753.36
≥ 18	247774	-386203	308873	-220.306	9791.37	1164.58	-2612.25
≥ 19	254041	-401906	327901	-213.96	9645.47	1664.94	-2786.2
≥ 20	256003	-402034	330566	-215.242	9850.42	1359.46	-2550.06

Table 2.1.29 (cont'd)

BWR FUEL ASSEMBLY COOLING TIME-DEPENDENT COEFFICIENTS
(ZR-CLAD FUEL)

Cooling Time **(years)**	**Array/Class 9x9C/D**						
	A	**B**	**C**	**D**	**E**	**F**	**G**
≥ 3	30051.6	29548.7	-18614.2	-148.276	6148.44	-1810.34	1006
≥ 4	70472.7	-14696.6	-233.567	-127.728	7008.69	-2634.22	-444.373
≥ 5	101298	-59638.9	23065.2	-138.523	7627.57	-2958.03	-377.965
≥ 6	125546	-102740	49217.4	-160.811	8096.34	-2798.88	-259.767
≥ 7	143887	-139261	74100.4	-184.302	8550.86	-2517.19	-275.151
≥ 8	159633	-172741	98641.4	-194.351	8636.89	-1838.81	-486.731
≥ 9	173517	-204709	124803	-212.604	9151.98	-1853.27	-887.137
≥ 10	182895	-225481	142362	-218.251	9262.59	-1408.25	-978.356
≥ 11	192530	-247839	162173	-217.381	9213.58	-818.676	-1222.12
≥ 12	201127	-268201	181030	-215.552	9147.44	-232.221	-1481.55
≥ 13	209538	-289761	203291	-225.092	9588.12	-574.227	-1749.35
≥ 14	216798	-306958	220468	-222.578	9518.22	-69.9307	-1919.71
≥ 15	223515	-323254	237933	-217.398	9366.52	475.506	-2012.93
≥ 16	228796	-334529	250541	-215.004	9369.33	662.325	-2122.75
≥ 17	237256	-356311	273419	-206.483	9029.55	1551.3	-2367.96
≥ 18	242778	-369493	290354	-215.557	9600.71	659.297	-2589.32
≥ 19	246704	-377971	302630	-210.768	9509.41	1025.34	-2476.06
≥ 20	249944	-382059	308281	-205.495	9362.63	1389.71	-2350.49

Table 2.1.29 (cont'd)

BWR FUEL ASSEMBLY COOLING TIME-DEPENDENT COEFFICIENTS
(ZR-CLAD FUEL)

Cooling Time **(years)**	**Array/Class 9x9E/F**						
	A	**B**	**C**	**D**	**E**	**F**	**G**
≥ 3	30284.3	26949.5	-16926.4	-147.914	6017.02	-1854.81	1026.15
≥ 4	69727.4	-17117.2	1982.33	-127.983	6874.68	-2673.01	-359.962
≥ 5	98438.9	-58492	23382.2	-138.712	7513.55	-3038.23	-112.641
≥ 6	119765	-95024.1	45261	-159.669	8074.25	-3129.49	221.182
≥ 7	136740	-128219	67940.1	-182.439	8595.68	-3098.17	315.544
≥ 8	150745	-156607	88691.5	-193.941	8908.73	-2947.64	142.072
≥ 9	162915	-182667	109134	-198.37	8999.11	-2531	-93.4908
≥ 10	174000	-208668	131543	-210.777	9365.52	-2511.74	-445.876
≥ 11	181524	-224252	145280	-212.407	9489.67	-2387.49	-544.123
≥ 12	188946	-240952	160787	-210.65	9478.1	-2029.94	-652.339
≥ 13	193762	-250900	171363	-215.798	9742.31	-2179.24	-608.636
≥ 14	203288	-275191	196115	-218.113	9992.5	-2437.71	-1065.92
≥ 15	208108	-284395	205221	-213.956	9857.25	-1970.65	-1082.94
≥ 16	215093	-301828	224757	-209.736	9789.58	-1718.37	-1303.35
≥ 17	220056	-310906	234180	-201.494	9541.73	-1230.42	-1284.15
≥ 18	224545	-320969	247724	-206.807	9892.97	-1790.61	-1381.9
≥ 19	226901	-322168	250395	-204.073	9902.14	-1748.78	-1253.22
≥ 20	235561	-345414	276856	-198.306	9720.78	-1284.14	-1569.18

Table 2.1.29 (cont'd)

BWR FUEL ASSEMBLY COOLING TIME-DEPENDENT COEFFICIENTS
(ZR-CLAD FUEL)

Cooling Time **(years)**	**Array/Class 9x9G**						
	A	**B**	**C**	**D**	**E**	**F**	**G**
≥ 3	35158.5	26918.5	-17976.7	-149.915	6787.19	-2154.29	836.894
≥ 4	77137.2	-19760.1	2371.28	-130.934	8015.43	-3512.38	-455.424
≥ 5	113405	-77931.2	35511.2	-150.637	8932.55	-4099.48	-629.806
≥ 6	139938	-128700	68698.3	-173.799	9451.22	-3847.83	-455.905
≥ 7	164267	-183309	109526	-193.952	9737.91	-3046.84	-737.992
≥ 8	182646	-227630	146275	-210.936	10092.3	-2489.3	-1066.96
≥ 9	199309	-270496	184230	-218.617	10124.3	-1453.81	-1381.41
≥ 10	213186	-308612	221699	-235.828	10703.2	-1483.31	-1821.73
≥ 11	225587	-342892	256242	-236.112	10658.5	-612.076	-2134.65
≥ 12	235725	-370471	285195	-234.378	10604.9	118.591	-2417.89
≥ 13	247043	-404028	323049	-245.79	11158.2	-281.813	-2869.82
≥ 14	253649	-421134	342682	-243.142	11082.3	400.019	-2903.88
≥ 15	262750	-448593	376340	-245.435	11241.2	581.355	-3125.07
≥ 16	270816	-470846	402249	-236.294	10845.4	1791.46	-3293.07
≥ 17	279840	-500272	441964	-241.324	11222.6	1455.84	-3528.25
≥ 18	284533	-511287	458538	-240.905	11367.2	1459.68	-3520.94
≥ 19	295787	-545885	501824	-235.685	11188.2	2082.21	-3954.2
≥ 20	300209	-556936	519174	-229.539	10956	2942.09	-3872.87

Table 2.1.29 (cont'd)

BWR FUEL ASSEMBLY COOLING TIME-DEPENDENT COEFFICIENTS
(ZR-CLAD FUEL)

Cooling Time **(years)**	**Array/Class 10x10A/B**						
	A	**B**	**C**	**D**	**E**	**F**	**G**
≥ 3	29285.4	27562.2	-16985	-148.415	5960.56	-1810.79	1001.45
≥ 4	67844.9	-14383	395.619	-127.723	6754.56	-2547.96	-369.267
≥ 5	96660.5	-55383.8	21180.4	-137.17	7296.6	-2793.58	-192.85
≥ 6	118098	-91995	42958	-162.985	7931.44	-2940.84	60.9197
≥ 7	135115	-123721	63588.9	-171.747	8060.23	-2485.59	73.6219
≥ 8	148721	-151690	84143.9	-190.26	8515.81	-2444.25	-63.4649
≥ 9	160770	-177397	104069	-197.534	8673.6	-2101.25	-331.046
≥ 10	170331	-198419	121817	-213.692	9178.33	-2351.54	-472.844
≥ 11	179130	-217799	138652	-209.75	9095.43	-1842.88	-705.254
≥ 12	186070	-232389	151792	-208.946	9104.52	-1565.11	-822.73
≥ 13	192407	-246005	164928	-209.696	9234.7	-1541.54	-979.245
≥ 14	200493	-265596	183851	-207.639	9159.83	-1095.72	-1240.61
≥ 15	205594	-276161	195760	-213.491	9564.23	-1672.22	-1333.64
≥ 16	209386	-282942	204110	-209.322	9515.83	-1506.86	-1286.82
≥ 17	214972	-295149	217095	-202.445	9292.34	-893.6	-1364.97
≥ 18	219312	-302748	225826	-198.667	9272.27	-878.536	-1379.58
≥ 19	223481	-310663	235908	-194.825	9252.9	-785.066	-1379.62
≥ 20	227628	-319115	247597	-199.194	9509.02	-1135.23	-1386.19

Table 2.1.29 (cont'd)

BWR FUEL ASSEMBLY COOLING TIME-DEPENDENT COEFFICIENTS
(ZR-CLAD FUEL)

Cooling Time **(years)**	**Array/Class 10x10C**						
	A	**B**	**C**	**D**	**E**	**F**	**G**
≥ 3	31425.3	27358.9	-17413.3	-152.096	6367.53	-1967.91	925.763
≥ 4	71804	-16964.1	1000.4	-129.299	7227.18	-2806.44	-416.92
≥ 5	102685	-62383.3	24971.2	-142.316	7961	-3290.98	-354.784
≥ 6	126962	-105802	51444.6	-164.283	8421.44	-3104.21	-186.615
≥ 7	146284	-145608	79275.5	-188.967	8927.23	-2859.08	-251.163
≥ 8	162748	-181259	105859	-199.122	9052.91	-2206.31	-554.124
≥ 9	176612	-214183	133261	-217.56	9492.17	-1999.28	-860.669
≥ 10	187756	-239944	155315	-219.56	9532.45	-1470.9	-1113.42
≥ 11	196580	-260941	174536	-222.457	9591.64	-944.473	-1225.79
≥ 12	208017	-291492	204805	-233.488	10058.3	-1217.01	-1749.84
≥ 13	214920	-307772	221158	-234.747	10137.1	-897.23	-1868.04
≥ 14	222562	-326471	240234	-228.569	9929.34	-183.47	-2016.12
≥ 15	228844	-342382	258347	-226.944	9936.76	117.061	-2106.05
≥ 16	233907	-353008	270390	-223.179	9910.72	360.39	-2105.23
≥ 17	244153	-383017	304819	-227.266	10103.2	380.393	-2633.23
≥ 18	249240	-395456	321452	-226.989	10284.1	169.947	-2623.67
≥ 19	254343	-406555	335240	-220.569	10070.5	764.689	-2640.2
≥ 20	260202	-421069	354249	-216.255	10069.9	854.497	-2732.77

SUPPLEMENT 5.II

SHIELDING EVALUATION OF THE HI-STORM 100 SYSTEM FOR IP1

5.II.0 INTRODUCTION

Indian Point Unit 1 (IP1) fuel assemblies, which have a maximum burnup of 30,000 MWD/MTU and a minimum cooling time of 30 years, are considerably shorter (approximately 137 inches) than most PWR assemblies. As a result of this reduced height and a crane capacity of 75 tons at IP1, the HI-STORM 100 System has been expanded to include options specific for use at IP1 as described in Supplement 1.II.

This supplement is focused on providing a shielding evaluation of the HI-STORM 100 system as modified for IP1. The evaluation presented herein supplements those evaluations of the HI-STORM overpacks contained in the main body of Chapter 5 of this FSAR and information in the main body of Chapter 5 that remains applicable to the HI-STORM 100 system at IP1 is not repeated in this supplement. To aid the reader, the sections in this supplement are numbered in the same fashion as the corresponding sections in the main body of this chapter, i.e., Sections 5.II.1 through 5.II.5 correspond to Sections 5.1 through 5.5. Tables and figures in this supplement are labeled sequentially.

The purpose of this supplement is to show that the dose rates from the HI-STORM system for IP1 are bounded by the dose rates calculated in the main section of this chapter, thereby demonstrating that the HI-STORM system for IP1 will comply with the radiological regulatory requirements.

5.II.1 DISCUSSION AND RESULTS

The HI-STORM 100 system for IP1 differs slightly from the HI-STORM system evaluated in the main body of this chapter. From a shielding perspective, the only difference in the overpack and MPC is the height. The top and bottom and radial thickness are identical. Therefore, considering the low burnup and long cooling time of the IP1 fuel, the dose rates from a HI-STORM 100S Version B overpack at IP1 containing the IP1 MPC-32 are bounded by the results presented in the main body of the chapter. Therefore, no specific analysis is provided in this supplement for the HI-STORM 100S Version B at IP1.

The HI-TRAC 100D Version IP1 is also shorter than the HI-TRAC 100D analyzed in the main body of this chapter. In addition to a shorter height, the radial thicknesses of the lead and outer shell have been reduced. However, the top and bottom of the HI-TRAC 100D Version IP1 are identical to the HI-TRAC 100D. Section 5.II.3 describes the HI-TRAC 100D Version IP1 as it was modeled in this supplement.

5.II.1.1 Normal Conditions

Shielding analyses were performed for the HI-TRAC 100D Version IP1 loaded with an IP1 MPC-32. A single burnup and cooling time combination of 30,000 MWD/MTU and 30 years was analyzed. Table 5.II.1 presents the results for the normal condition, where the MPC is dry and the HI-TRAC water jacket is filled with water ~~at the midplane of the overpack. Since the only change in shielding between the HI-TRAC 100D and the 100D Version IP1 is in the radial direction, it is reasonable to present only the dose rates at the midplane of the overpack.~~ A comparison of the results in Table 5.II.1 to the results in Tables 5.4.11, ~~and~~ 5.4.12 and 5.4.19 demonstrate that the dose rates from the HI-TRAC 100D Version IP1 are considerably less than and bounded by the dose rates from the HI-TRAC 100 and HI-TRAC 100D with design basis fuel.

5.II.1.2 Accident Conditions

The bounding accident condition for the HI-TRAC 100D Version IP1 is the loss of all water in the water jacket during a transfer operation with a dry MPC. Shielding analyses were performed for this condition for the same burnup and cooling time used in the analysis of the normal condition. Table 5.II.2 presents the results of the analysis. ~~Consistent with evaluations in the main part of this chapter, only the 1 meter dose rates for dose point 2 are reported.~~ A comparison of the results in Table 5.II.2 to the results in Tables 5.1.10 demonstrate that the dose rates from the HI-TRAC 100D Version IP1 are considerably less than and are bounded by the dose rates from the HI-TRAC 100 with design basis fuel. Further, since the dose rates at 1 meter are considerably less than those of the HI-TRAC 100 it can be concluded that dose rates at the 100 meter controlled area boundary for HI-TRAC 100D Version IP1 are also bounded by those of the HI-TRAC 100.

5.II.1.3 Fuel Condition

The Indian Point 1 assemblies are assumed damaged and are to be placed into DFCs for the purpose of compliance with the damaged fuel definition. However, they are not actually considered damaged. All assemblies have been inspected and are considered intact. In actuality, the design of the assemblies with the shroud surrounding the rods and the cladding made out of stainless steel, they would be much less likely to be damaged under any accident condition than standard PWR assemblies. The distinction between intact and damaged fuel is of primary importance from a criticality perspective, specifically for the situation at Indian Point Unit 1 where the assemblies are located in a non-borated pool. Nevertheless, to show the potential effect on dose rates from damage to the assemblies, studies were performed consistent with the calculations discussed in Section 5.4.2.2. The analysis consisted of modeling the fuel assemblies in all locations in the MPC-32 with a fuel density that was twice the normal fuel amount per unit length and correspondingly increasing the source rate for these locations by a factor of two. The fuel is spread over the entire cross section of the DFC. A flat axial power distribution was used which is approximately representative of the source distribution if the top half of an assembly collapsed into the bottom half of the assembly.

Increasing the fuel amount per unit length over the entire fuel length, rather than in the top half or bottom half of the fuel assembly, is conservative and provides the dose rate change in both the top and bottom portion of the cask. Results are presented in Table 5.II.3 for both normal and accident conditions (see Sections 5.II.1.1 and 5.II.1.2). The results for the normal condition show a small increase of about 3.7% for the maximum dose rate at dose location 2, and increases of up to 21% and 43% at the top and bottom of the casks, respectively. The results for the accident condition show a small increase of about 10% for the maximum dose rate at dose location 2, and increases of up to 28% and 46% at the top and bottom of the casks, respectively. Several other configurations were evaluated, involving different combinations of increased or decreased fuel amount and/or fuel cross section. They all resulted in a smaller increase or even decrease of dose rates. The condition identified above therefore presents a bounding condition for damaged fuel. In that context also note that the shielding effect of the damaged fuel container was neglected in the MCNP model.

5.II.2 SOURCE SPECIFICATION

The characteristics of the Indian Point Unit 1 fuel assembly are shown in Table 5.II.4~~2~~. The maximum length of the active fuel zone in this assembly is 102 inches. However, the source term was calculated assuming an active fuel length of 144 inches. The longer active fuel length was used for ease of modeling as described in Section 5.II.3. The end fittings above and below the active fuel zone were assumed to be identical to the end fittings of the design basis zircaloy PWR fuel assembly described in Section 5.2. Tables 5.II.~~3~~5 and 5.II.~~4~~6 presents the neutron and gamma source term for the active fuel region of the IP1 fuel assemblies.

Earlier manufactured fuel such as the IP1 fuel potentially has a higher cobalt content in the stainless steel parts of the assembly than more recent fuel. As a bounding approach, a high cobalt content of 2.2 g/kg is assumed for all stainless steel parts of the fuel assembly, including the cladding. This value bounds the highest measurement value documented in [5.2.3]

The source term for the IP1 fuel was based on an initial minimum enrichment of 3.5 w/o ^{235}U and burnup of 30,000 MWD/MTU. IP1 has four fuel assemblies that have an initial enrichment less than 3.5 wt% ^{235}U. These four assemblies have a burnup less than 10,000 MWD/MTU and an enrichment that is greater than 2.7 wt%. The source term from the design basis IP1 fuel assembly with an enrichment of 3.5 wt% and a burnup of 30,000 MWD/MTU bounds the source term from a fuel assembly with 2.7 wt% and a burnup of 10,000 MWD/MTU. The calculations provided here therefore bound all IP1 assemblies.

IP1 fuel assemblies resemble BWR fuel assemblies in that they have a shroud that encompasses the fuel rods similar to the channel around BWR fuel. However, unlike BWR channels, the shroud is perforated with uniformly spaced holes. Characteristics of the shroud are shown in Table 5.II.4. The 47% open area due to these holes was used to calculate t~~T~~he source term from *the activation of the~~is~~ shroud with a cobalt-59 impurity level of 2.2 gm/kg [5.2.3] ~~was considered in this analysis~~ and is included in Table 5.II.~~4~~6.*

5.II.2.1 Secondary Sources

Antimony-beryllium sources were used as secondary (regenerative) neutron sources in IP1. The Sb-Be source produces neutrons from a gamma-n reaction in the beryllium, where the gamma originates from the decay of neutron-activated antimony. The very short half-life of ^{124}Sb, 60.2 days, however results in a complete decay of the initial amount generated in the reactor within a few years after removal from the reactor. ~~The production of neutrons by the Sb-Be source through regeneration in the MPC is orders of magnitude lower than the design-basis fuel assemblies. Therefore Sb-Be sources do not contribute to the total neutron source in the MPC and are not specifically analyzed in this supplement.~~ Analyses also show that the re-generation of ^{124}Sb through the fuel neutrons is too small to generate a noticeable neutron source from the Be. However, neutrons are generated in the Be through Be's gamma-n reaction and the gamma radiation from the fuel. A detailed analysis of this situation has been analyzed for the MPC-32 and a 14x14 assembly type with zircaloy clad fuel. Results from this assembly bound the condition with IP1 fuel in the MPC-32, since the IP1 fuel has stainless steel cladding. This would result in reduced gamma radiation levels for the same burnup and cooling time. The IP1 assemblies contain the source in a single rod that replaces one of the fuel rods. However, the length of the source in the rod is not known. It is therefore conservatively assumed that the length of the source is equal to the active fuel length. Under these conditions, the neutron generation from a single source would be 3.83E+4 n/s. With a neutron source strength of a fuel assembly of 2.17E+7 n/s, this represents less than 0.5% of the neutron source strength of the assembly, and is in fact similar to the source strength of the rod that is replaced by the secondary source. Therefore, it is not necessary to explicitly consider the sources in the dose rate analyses.

Regarding the steel portions of the neutron source, it is important to note that Indian Point Unit 1 secondary source devices were not removable inserts. Instead, these devices replaced a stainless steel clad fuel rod in the fuel assembly. Therefore, the secondary sources were in the core for the same amount of time as the assembly in which they were placed and have achieved the same burnup as the fuel assembly. As a result, the gamma source term from a fuel assembly containing all fuel rods bounds the gamma source term from a fuel assembly containing a secondary source device.

5.II.3 MODEL SPECIFICATIONS

The shielding analyses of the HI-TRAC 100D Version IP1 are performed with MCNP-4A, which is the same code used for the analyses presented in the main body of this chapter.

Section 1.5 provides the drawings that describe the HI-TRAC 100D Version IP1. These drawings, using nominal dimensions, were used to create the MCNP models used in the radiation transport calculations. Since the HI-TRAC 100D Version IP1 is a variation of the HI-

TRAC 100D, the model of the 100D was modified by appropriately reducing the radial dimensions of the 100D model. Conservatively, the axial height was not changed. Table 5.II.7~~5~~ shows the radial thicknesses of the shielding materials in the 100D Version IP1 compared to the 100D.

In order to represent the IP1 fuel assemblies, the 144 inch active fuel region of the design basis PWR fuel assembly was not changed to represent the IP1 fuel assemblies. This conservatively modeled the active fuel region as 144 inches in length rather than 102 inches. The shielding effect of the shroud around the fuel assembly was conservatively neglected in the MCNP model.

~~n S 5.4.2.2h assemblies on the periphery dominate the radial dose rates~~*Note that the* ~~*IP1 fuel assemblies are considered intact and have all been inspected with no visible damage identified. However, it may not be possible to classify these assemblies as intact due to insufficient records. Therefore, all IP1 fuel assemblies are required to be stored in a damaged fuel container. Conservatively, the*~~ *shielding effect of the damaged fuel container was neglected in the MCNP model.* ~~*Since the IP1 fuel assemblies are considered intact, the analysis in this supplement treated the fuel assemblies as intact.*~~

5.II.4 SHIELDING EVALUATION

Table 5.II.1 provides dose rates adjacent to and at 1 meter distance from ~~*the midplane of*~~ *the HI-TRAC 100D Version IP1 during normal conditions for the MPC-32. Table 5.II.2 provides dose rate at 1 meter distance on the mid-plane for the HI-TRAC 100D Version IP1 during accident conditions for the MPC-32.* Table 5.II.3 provides dose rates assuming damaged condition for the fuel. *These results demonstrate that the dose rates around the HI-TRAC 100D Version IP1 are considerably lower than the HI-TRAC 100 and 100D as documented in Section 5.4.*

5.II.5 REGULATORY COMPLIANCE

In summary it can be concluded that dose rates from the HI-STORM 100 system as modified for IP1 are bounded by the dose rates for the overpacks analyzed in the main body of the report. The shielding system of the HI-STORM 100 system is therefore in compliance with 10CFR72 and satisfies the applicable design and acceptance criteria including 10CFR20. Thus, the shielding evaluation presented in this supplement provides reasonable assurance that the HI-STORM 100 system for IP1 will allow safe storage of IP1 spent fuel.

Table 5.II.1

DOSE RATES ADJACENT TO AND 1 METER FROM THE HI-TRAC 100D VERSION IP1 FOR NORMAL CONDITIONS†††
MPC-32 WITH INTACT IP1 FUEL
30,000 MWD/MTU AND 30-YEAR COOLING

Dose Point† Location	***Fuel Gammas†† (mrem/hr)***	***^{60}Co Gammas (mrem/hr)***	***Neutrons (mrem/hr)***	***Totals (mrem/hr)***
ADJACENT TO HI-TRAC 100D VERSION IP1				
1	*25.42*	*152.92*	*11.72*	*190.06*
2	*480.57*	*0.21*	*10.68*	*491.46*
3	*4.42*	*54.45*	*11.74*	*70.61*
ONE METER FROM HI-TRAC 100D VERSION IP1				
1	*64.00*	*25.32*	*3.02*	*92.35*
2	*205.71*	*1.79*	*4.02*	*211.52*
3	*27.08*	*16.18*	*1.69*	*44.95*

† *Refer to Figure 5.1.4.*

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

††† *Dose rate based on no water within the MPC. For the majority of the duration that the HI-TRAC pool lid is installed, the MPC cavity will be flooded with water. The water within the MPC greatly reduces the dose rate.*

Table 5.II.2

DOSE RATES ONE METER FROM THE
HI-TRAC 100D VERSION IP1 FOR ACCIDENT CONDITIONS†††
MPC-32 WITH INTACT IP1 FUEL
30,000 MWD/MTU AND 30-YEAR COOLING

Dose Point† Location	***Fuel Gammas†† (mrem/hr)***	***^{60}Co Gammas (mrem/hr)***	***Neutrons (mrem/hr)***	***Totals (mrem/hr)***
1	*114.01*	*37.73*	*47.73*	*199.46*
2	*366.25*	*3.23*	*97.04*	*466.52*
3	*49.17*	*24.28*	*22.34*	*95.78*

† *Refer to Figure 5.1.4.*

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

††† *Dose rate based on no water within the MPC and no water in the water jacket.*

Table 5.II.3

DOSE RATES ADJACENT TO AND 1 METER FROM THE HI-TRAC 100D VERSION IP1 FOR NORMAL AND ACCIDENT CONDITIONS ASSUMING DAMAGED FUEL MPC-32 WITH IP1 FUEL 30,000 MWD/MTU AND 30-YEAR COOLING

Dose Point† Location	***Fuel Gammas†† (mrem/hr)***	***^{60}Co Gammas (mrem/hr)***	***Neutrons (mrem/hr)***	***Totals (mrem/hr)***
NORMAL CONDITION ADJACENT TO HI-TRAC 100D VERSION IP1				
1	48.00	152.92	26.08	226.99
2	495.01	2.36	11.97	509.34
3	8.71	54.45	37.69	100.85
NORMAL CONDITION ONE METER FROM HI-TRAC 100D VERSION IP1				
1	81.53	25.32	5.73	112.58
2	212.38	1.79	5.36	219.53
3	40.42	16.18	4.91	61.51
ACCIDENT CONDITION ONE METER FROM HI-TRAC 100D VERSION IP1				
1	143.24	37.73	73.84	254.8
2	379.33	3.23	130.27	512.82
3	70.89	24.28	44.65	139.82

† Refer to Figure 5.1.4.

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

††† *Dose rate based on no water within the MPC. For the majority of the duration that the HI-TRAC pool lid is installed, the MPC cavity will be flooded with water. The water within the MPC greatly reduces the dose rate.*

Table 5.II.~~4~~2

DESCRIPTION OF DESIGN BASIS STAINLESS STEEL CLAD FUEL

Description	***Value***
Fuel type	*14x14*
Active fuel length (in.)	*144*
No. of fuel rods	*173*
Rod pitch (in.)	*0.441*
Cladding material	*Stainless steel*
Rod diameter (in.)	*0.3415*
Cladding thickness (in.)	*0.012*
Pellet diameter (in.)	*0.313*
Pellet material	UO_2
Pellet density (gm/cc)	*10.412 (95% of theoretical)*
Enrichment (w/o ^{235}U*)*	*3.5*
Burnup (MWD/MTU)	*30,000*
Cooling Time (years)	*30*
Specific power (MW/MTU)	*25.09*
No. of guide tubes	*0*
Shroud material	*Stainless steel*
Shroud thickness (in.)	*0.035*
Percent open area of shroud	*47*

Table 5.II.~~3~~5

CALCULATED NEUTRON SOURCE PER ASSEMBLY
FOR STAINLESS STEEL CLAD IP1 FUEL

Lower Energy (MeV)	***Upper Energy (MeV)***	***30,000 MWD/MTU 30-Year Cooling (Neutrons/s)***
1.0e-01	*4.0e-01*	*7.76e+05*
4.0e-01	*9.0e-01*	*3.97e+06*
9.0e-01	*1.4*	*3.72e+06*
1.4	*1.85*	*2.86e+06*
1.85	*3.0*	*5.47e+06*
3.0	*6.43*	*4.55e+06*
6.43	*20.0*	*3.78e+05*
Total		*2.17e+07*

Table 5.II.64

CALCULATED FUEL GAMMA SOURCE PER ASSEMBLY
FOR STAINLESS STEEL CLAD IP1 FUEL

Lower Energy	***Upper Energy***	***30,000 MWD/MTU 30-Year Cooling***	
(MeV)	*(MeV)*	*(MeV/s)*	*(Photons/s)*
4.5e-01	7.0e-01	2.94e+14	5.10e+14
7.0e-01	1.0	4.38e+12	5.15e+12
1.0	1.5	3.15e+13	2.52e+13
1.5	2.0	2.94e+11	1.68e+11
2.0	2.5	2.82e+09	1.25e+09
2.5	3.0	1.85e+08	6.72e+07
Totals		3.13e+14	5.27e+14

Table 5.II.7~~5~~

A COMPARISON OF THE RADIAL SHIELDING THICKNESSES OF THE HI-TRAC 100D VERSION IP1 AND THE HI-TRAC 100D

Shielding Material	***HI-TRAC 100D***	***HI-TRAC 100D Version IP1***
Inner steel shell (in.)	*0.75*	*0.75*
Lead (in.)	*2.875*	*2.5*
Outer steel shell (in.)	*1.0*	*0.75*
Water in water jacket (in.)	*5.0*	*5.0*
Steel water jacket enclosure (in.)	*0.375*	*0.375*
Total thickness (in.)	*10.0*	*9.375*

SUPPLEMENT 11.II

OFF-NORMAL AND ACCIDENT EVALUATION FOR HI-STORM 100S-185

11.II.0 INTRODUCTION

This supplement is focused on the off-normal and accident condition evaluations of the HI-STORM 100S-185 System for storage of IP1 fuel. The evaluations described herein parallel those of the HI-STORM 100 System contained in the main body of Chapter 11 of this FSAR. To ensure readability, the sections in this supplement are numbered to be directly analogous to the sections in the main body of the chapter. For example, the fire accident evaluation presented in Supplement Subsection 11.II.2.4 for the HI-STORM 100S-185 is analogous to the evaluation presented in Subsection 11.2.4 of the main body of Chapter 11 for the HI-STORM 100.

11.II.1 OFF-NORMAL EVENTS

A general discussion of off-normal events is presented in Section 11.1 of the main body of Chapter 11. The following off-normal events are discussed in this supplement:

Off-Normal Pressure
Off-Normal Environmental Temperature
Leakage of One MPC Seal Weld
Partial Blockage of Air Inlets
Off-Normal Handling of HI-TRAC Transfer Cask
FHD System Failure

The results of the evaluations presented herein demonstrate that the HI-STORM 100S-185 System can withstand the effects of off-normal events without affecting its ability to perform its intended function, and is in compliance with the applicable acceptance criteria.

11.II.1.1 Off-Normal Pressure

A discussion of this off-normal condition is presented in Subsection 11.1.1 of the main body of Chapter 11. A description of the cause of, detection of, corrective actions for and radiological impact of this event is presented therein.

Structural

The structural evaluation of the MPC enclosure vessel for off-normal internal pressure conditions is discussed in Section 3.4. The applicable pressure boundary stress limits are confirmed to bound the stresses resulting from the off-normal pressure.

Thermal

The off-normal event is evaluated for the generic HI-STORM in Section 4.6.1 This evaluation is bounding as the MPC temperatures and pressures in a HI-STORM 100S-185 are bounded by the generic HI-STORM System.

Shielding

There is no effect on the shielding performance of the system as a result of this off-normal event.

Criticality

There is no effect on the criticality control features of the system as a result of this off-normal event.

Confinement

There is no effect on the confinement function of the MPC as a result of this off-normal event. As discussed in the structural evaluation mentioned above, all stresses remain within allowable values, assuring confinement boundary integrity.

Radiation Protection

Since there is no degradation in shielding or confinement capabilities as discussed above, there is no effect on occupational or public exposures as a result of this off-normal event.

Based on this evaluation, it is concluded that the off-normal pressure does not affect the safe operation of the HI-STORM 100S-185 System.

11.II.1.2 Off-Normal Environmental Temperatures

A discussion of this off-normal condition is presented in Subsection 11.1.2 of the main body of Chapter 11. A description of the cause of, detection of, corrective actions for and radiological impact of this event is presented therein.

Structural

The effect on the MPC for the upper off-normal thermal conditions (i.e., 100°F) is an increase in the internal pressure. The resultant pressure is below the off-normal design pressure (Table 2.2.1).

Thermal

The effect of off-normal ambient temperature on HI-STORM temperatures and pressures is evaluated in Section 4.II.6.

Shielding

There is no effect on the shielding performance of the system as a result of this off-normal event.

Criticality

There is no effect on the criticality control features of the system as a result of this off-normal event.

Confinement

There is no effect on the confinement function of the MPC as a result of this off-normal event.

Radiation Protection

Since there is no degradation in shielding or confinement capabilities as discussed above, there is no effect on occupational or public exposures as a result of this off-normal event.

Based on this evaluation, it is concluded that the specified off-normal environmental temperatures do not affect the safe operation of the HI-STORM 100S-185 System.

11.II.1.3 Leakage of One MPC Seal Weld

A discussion of this off-normal condition is presented in Subsection 11.1.3 of the main body of Chapter 11. The discussion presented therein is applicable in its entirety to an MPC in a HI-STORM 100S-185.

11.II.1.4 Partial Blockage of Air Inlets

A discussion of this off-normal condition is presented in Subsection 11.1.4 of the main body of Chapter 11. A description of the cause of, detection of, corrective actions for and radiological impact of this event is presented therein.

Structural

There are no structural consequences as a result of this off-normal event.

Thermal

Partial air inlets blockage is evaluated in Section 4.II.6.

Shielding

There is no effect on the shielding performance of the system as a result of this off-normal event.

Criticality

There is no effect on the criticality control features of the system as a result of this off-normal event.

Confinement

There is no effect on the confinement function of the MPC as a result of this off-normal event.

Radiation Protection

Since there is no degradation in shielding or confinement capabilities as discussed above, there is no effect on occupational or public exposures as a result of this off-normal event.

Based on this evaluation, it is concluded that the specified off-normal partial blockage of air inlet ducts event does not affect the safe operation of the HI-STORM 100S-185 System.

11.II.1.5 Off-Normal Handling of HI-TRAC

A discussion of this off-normal condition is presented in Subsection 11.1.5 of the main body of Chapter 11. This off-normal condition does not apply to the HI-TRAC 100D Version IP1, which does not have lower pocket trunnions. ~~*Upending and downending of the HI-TRAC 100D Version IP1 is performed using an L-frame.*~~

11.II.1.6 Failure of FHD System

A discussion of this off-normal condition is presented in Subsection 11.1.6 of the main body of Chapter 11. The discussion presented therein is also applicable to the IP1 cask system.

11.II.2 ACCIDENT EVENTS

A general discussion of accident events is presented in Section 11.1 of the main body of Chapter 11. The following accident events are discussed in this supplement section:

HI-TRAC Transfer Cask Handling Accident
HI-STORM 100S-185 Overpack Handling Accident
Tip-Over
Fire Accident
Partial Blockage of MPC Basket Vent Holes
Tornado
Flood
Earthquake
100% Fuel Rod Rupture
Confinement Boundary Leakage
Explosion
Lightning

100% Blockage of Air Inlets
Burial Under Debris
Extreme Environmental Temperature

The results of the evaluations performed herein demonstrate that the HI-STORM 100S-185 System can withstand the effects of all credible and hypothetical accident conditions and natural phenomena without affecting safety function, and is in compliance with the applicable acceptance criteria.

11.II.2.1 HI-TRAC Transfer Cask Handling Accident

A discussion of this accident condition is presented in Subsection 11.2.1 of the main body of Chapter 11. ~~Unless a site-specific analysis has been performed to determine a lift height limit, t~~The HI-TRAC 100D Version IP1 shall be transported and handled only in the vertical orientation using a device designed in accordance with ANSI N14.6 and having redundant drop protection features. Therefore, a drop of the loaded HI-TRAC 100D Version IP1 is not a credible accident.

11.II.2.2 HI-STORM Overpack Handling Accident

A discussion of this accident condition is presented in Subsection 11.2.2~~1~~ of the main body of Chapter 11. The discussion presented therein applies to the HI-STORM 100S-185 System, except that the height of the loaded overpack above the ground shall be limited to below the vertical handling height limit determined in Supplement 3.II.

11.II.2.3 Tip-Over

A discussion of this accident condition is presented in Subsection 11.2.3~~1~~ of the main body of Chapter 11. The discussion presented therein applies to the HI-STORM 100S-185 System, except that the tip-over analysis of the HI-STORM 100S-185 overpack is provided in Supplement 3.II, Section 3.II.4.

11.II.2.4 Fire Accident

A discussion of this accident condition is presented in Subsection 11.2.4 of the main body of Chapter 11. A description of the cause of and corrective actions for this event is presented therein.

Structural

There are no structural consequences as a result of the fire accident condition.

Thermal

Supplement 4.II, Section 4.II.6 evaluates fire accidents for the HI-STORM 100S-185 System. As justified therein, the evaluation of fires on a generic HI-STORM System presented in Section 11.2 bound the effects on the HI-STORM 100S-185 System.

Shielding

With respect to concrete damage from a fire to the HI-STORM 100S-185 System*, NUREG-1536 (4.0,V,5.b) states: "the loss of a small amount of shielding material is not expected to cause a storage system to exceed the regulatory requirements in 10 CFR 72.106 and, therefore, need not be estimated or evaluated in the SAR."*

For the HI-TRAC 100D Version IP1, the assumed loss of all the water in the water jacket results in an increase in the radiation dose rates at locations adjacent to the water jacket. The shielding evaluation presented in Supplement 5.II demonstrates that the requirements of 10CFR72.106 are not exceeded.

Criticality

There is no effect on the criticality control features of the system as a result of this accident event.

Confinement

There is no effect on the confinement function of the MPC as a result of this accident event.

Radiation Protection

Since there is a very localized reduction in shielding and no effect on the confinement capabilities as discussed above, there is no effect on occupational or public exposures as a result of this accident event.

Based on this evaluation, it is concluded that the fire accident does not affect the safe operation of the HI-STORM 100S-185 System.

For the HI-TRAC 100D Version IP1, there is no degradation in confinement capabilities of the MPC, as discussed above. There are increases in the local dose rates adjacent to the water jacket. Dose rates at 1 meter from the water jacket, after the water is lost, are presented in Supplement 5.II and it is concluded that dose rates at the 100 meter controlled boundary for the HI-TRAC 100D Version IP1 are bounded by the HI-TRAC 100. Immediately after the fire accident a radiological inspection of the HI-TRAC will be performed and temporary shielding shall be installed to limit the exposure to the public.

11.II.2.5 Partial Blockage of MPC Basket Vent Holes

A discussion of this accident condition is presented in Subsection 11.2.5 of the main body of Chapter 11. The discussion presented therein applies to an MPC-32-IP1 in a HI-STORM 100S-185.

11.II.2.6 Tornado

A discussion of this accident condition is presented in Subsection 11.2.6 of the main body of Chapter 11. A description of the cause of and corrective actions for this event is presented therein.

Structural

Analyses presented in Supplement 3.II, Section 3.II.4 show that the impact of tornado and tornado borne missiles on the HI-STORM 100S-185 System does not result in tip-over or a direct missile strike on the MPC.

Thermal

There are no thermal consequences as a result of the tornado.

Shielding

A tornado missile may cause localized damage to the HI-STORM 100S 185 Overpack. As the overpack is heavily shielded, the overall damage consequences (site boundary doses) are insignificant.

A tornado missile may penetrate the HI-TRAC100D Version IP water jacket shell causing the loss of the neutron shielding (water) which results in an increase in dose rates adjacent to the water jacket. The shielding evaluation presented in Supplement 5.II demonstrates that the requirements of 10CFR72.106 are not exceeded.

Criticality

There is no effect on the criticality control features of the system as a result of this accident event.

Confinement

There is no effect on the confinement function of the MPC as a result of this accident event.

Radiation Protection

There is no degradation in confinement capabilities of the MPC, since the tornado missiles do not impact the MPC, as discussed above. A tornado missile may cause localized damage in the

HI-STORM 100S 185 overpack. However, the damage will have a negligible effect on the site boundary dose. Based on this evaluation, it is concluded that the tornado accident does not affect the safe operation of the HI-STORM 100S-185 System.

A tornado missile may penetrate the HI-TRAC 100D Version IP1 water jacket shell causing the loss of the neutron shielding (water). There are increases in the local dose rates adjacent to the water jacket. Dose rates at 1 meter from the water jacket, after the water is lost, are presented in Supplement 5.II and it is concluded that dose rates at the 100 meter controlled boundary for the HI-TRAC 100D Version IP1 are bounded by the HI-TRAC 100. Immediately after the tornado missile accident a radiological inspection of the HI-TRAC shall be performed and temporary shielding shall be installed to limit exposure.

11.II.2.7 Flood

A discussion of this accident condition is presented in Subsection 11.2.7 of the main body of Chapter 11. A description of the cause of this event is presented therein.

Structural

The structural evaluation of the MPC for the accident condition external pressure (Table 2.2.1) is presented in Section 3.4 and the resulting stresses from this event are shown to be well within the allowable values.

Thermal

The thermal consequences of flood are bounded by the all inlet ducts blocked accident.

Shielding

There is no effect on the shielding performance of the system as a result of this accident event. The floodwater provides additional shielding which reduces radiation dose.

Criticality

There is no effect on the criticality control features of the system as a result of this accident event. The criticality analysis is unaffected because under the flooding condition water does not enter the MPC cavity and therefore the reactivity would be less than the loading condition in the spent fuel pool, which is presented in Section 6.1.

Confinement

There is no effect on the confinement function of the MPC as a result of this accident event. As discussed in the structural evaluation above, all stresses remain within allowable values, assuring confinement boundary integrity.

Radiation Protection

Since there is no degradation in shielding or confinement capabilities as discussed above, there is no effect on occupational or public exposures as a result of this accident event.

Based on this evaluation, it is concluded that the flood accident does not affect the safe operation of the HI-STORM 100S-185 System.

Flood Accident Corrective Action

The HI-STORM 100S 185 System is unaffected by flood. Upon recession of floodwaters, exposed surfaces may need debris and adherent foreign matter removal.

11.II.2.8 Earthquake

A discussion of this accident condition is presented in Subsection 11.2.8 of the main body of Chapter 11. A description of the cause of and corrective actions for this event is presented therein.

Structural

An evaluation presented in Supplement 3.II, Section 3.II.4 shows that the HI-STORM 100S-185 does not tip over. It continues to render its intended function during and after the earthquake and the overpack is unaffected by the event.

Thermal

There is no effect on the thermal performance of the system as a result of this accident event.

Shielding

There is no effect on the shielding performance of the system as a result of this accident event.

Criticality

There is no effect on the criticality control features of the system as a result of this accident event.

Confinement

There is no effect on the confinement function of the MPC as a result of this accident event.

Radiation Protection

Since there is no degradation in shielding or confinement capabilities as discussed above, there is no effect on occupational or public exposures as a result of this accident event.

Based on this evaluation, it is concluded that the earthquake does not affect the safe operation of the HI-STORM 100S-185 System.

11.II.2.9 <u>100% Fuel Rod Rupture</u>

A discussion of this accident condition is presented in Subsection 11.2.9 of the main body of Chapter 11. A description of the cause of and corrective actions for this event is presented therein.

<u>Structural</u>

The MPC accident pressure is below the design pressure of the MPC (Table 2.2.1).

<u>Thermal</u>

The 100% fuel rods rupture accident pressure is evaluated in Supplement II, Section 4.II.4.4. The MPC accident pressure is below the vessel design pressure (Table 2.2.1).

<u>Shielding</u>

There is no effect on the shielding performance of the system as a result of this accident event.

<u>Criticality</u>

There is no effect on the criticality control features of the system as a result of this accident event.

<u>Confinement</u>

There is no effect on the confinement function of the MPC as a result of this accident event. As discussed in the structural evaluation above, all stresses remain within allowable values, assuring confinement boundary integrity.
<u>Radiation Protection</u>

Since there is no degradation in shielding or confinement capabilities as discussed above, there is no effect on occupational or public exposures as a result of this accident event.

Based on this evaluation, it is concluded that the non-mechanistic 100% fuel rod rupture accident does not affect the safe operation of the HI-STORM 100S-185 System.

11.II.2.10 <u>Confinement Boundary Leakage</u>

A discussion of this accident condition is presented in Subsection 11.2.10 of the main body of Chapter 11. The discussion presented therein also applies to the MPC-32-IP1.

11.II.2.11 Explosion

A discussion of this accident condition is presented in Subsection 11.2.11 of the main body of Chapter 11. A description of the cause of and corrective actions for this event is presented therein.

Structural

The structural evaluations for the MPC accident condition external pressure and overpack pressure differential are presented in Section 3.4 and demonstrate that all stresses are within allowable limits.

Thermal

There is no effect on the thermal performance of the system as a result of this accident event.

Shielding

There is no effect on the shielding performance of the system as a result of this accident event.

Criticality

There is no effect on the criticality control features of the system as a result of this accident event.

Confinement

There is no effect on the confinement function of the MPC as a result of this accident event. As discussed in the structural evaluation above, all stresses remain well within allowable values, assuring confinement boundary integrity.

Radiation Protection

Since there is no degradation in shielding or confinement capabilities as discussed above, there is no effect on occupational or public exposures as a result of this accident event.

Based on this evaluation, it is concluded that the explosion accident does not affect the safe operation of the HI-STORM 100S-185 System.

11.II.2.12 Lightning

A discussion of this accident condition is presented in Subsection 11.2.12 of the main body of Chapter 11. The discussion presented therein also applies to the HI-STORM 100S-185.

11.II.2.13 <u>100% Blockage of Air Inlets</u>

A discussion of this accident condition is presented in Subsection 11.2.13 of the main body of Chapter 11. A description of the cause of and corrective actions for this event is presented therein.

<u>Structural</u>

There are no structural consequences as a result of this accident event.

<u>Thermal</u>

The 100% air inlets blockage accident is evaluated in Supplement II, Section 4.II.6.

<u>Shielding</u>

There is no effect on the shielding performance of the system as a result of this accident event, since the concrete temperatures do not exceed the accident temperature limit.

<u>Criticality</u>

There is no effect on the criticality control features of the system as a result of this accident event.

<u>Confinement</u>

There is no effect on the confinement function of the MPC as a result of this accident event.

<u>Radiation Protection</u>

Since there is no degradation in shielding or confinement capabilities as discussed above, there is no effect on occupational or public exposures as a result of this accident event.

Based on this evaluation, it is concluded that the 100% blockage of air inlets accident does not affect the safe operation of the HI-STORM 100S-185 System, if the blockage is removed in the specified time period.

11.II.2.14 <u>Burial Under Debris</u>

A discussion of this accident condition is presented in Subsection 11.2.14 of the main body of Chapter 11. A description of the cause of and corrective actions for this event is presented therein.

Structural

The structural evaluation of the MPC enclosure vessel for accident internal pressure conditions bounds the pressure calculated herein. Therefore, the resulting stresses from this event are well within the allowable values, as demonstrated in Section 3.4.

Thermal

The burial under debris accident is evaluated in Supplement II, Section 4.II.6.

Shielding

There is no adverse effect on the shielding performance of the system as a result of this accident event.

Criticality

There is no effect on the criticality control features of the system as a result of this accident event.

Confinement

There is no effect on the confinement function of the MPC as a result of this accident event. As discussed in the structural evaluation above, all stresses remain within allowable values, assuring confinement boundary integrity.

Radiation Protection

Since there is no degradation in shielding or confinement capabilities as discussed above, there is no effect on occupational or public exposures as a result of this accident event.

Based on this evaluation, it is concluded that the burial under debris accident does not affect the safe operation of the HI-STORM 100S-185 System, if the debris is removed within the specified time period.

11.II.2.15 Extreme Environmental Temperature

A discussion of this accident condition is presented in Subsection 11.2.15 of the main body of Chapter 11. A description of the cause of and corrective actions for this event is presented therein.

Structural

The structural evaluation of the MPC enclosure vessel for accident condition internal pressure bounds the pressure resulting from this event. Therefore, the resulting stresses from this event *are bounded by the design-basis internal pressure and are well within the allowable values, as discussed in Section 3.4.*

Thermal

The extreme ambient temperature accident is evaluated in Supplement 4.II, Section 4.II.6.

Shielding

There is no effect on the shielding performance of the system as a result of this accident event, since the concrete temperature does not exceed the short-term temperature limit specified in Table 2.2.3.

Criticality

There is no effect on the criticality control features of the system as a result of this accident event.

Confinement

There is no effect on the confinement function of the MPC as a result of this accident event. As discussed in the structural evaluation above, all stresses remain within allowable values, assuring confinement boundary integrity.

Radiation Protection

Since there is no degradation in shielding or confinement capabilities as discussed above, there is no effect on occupational or public exposures as a result of this accident event.

Based on this evaluation, it is concluded that the extreme environment temperature accident does not affect the safe operation of the HI-STORM 100S-185 System.

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U. S. Nuclear Regulatory Commission
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Document ID 5014623

Holtec Letter 5014623 (Response to RAI-2 on LAR 1014-5)

Attachment 4

Holtec Drawing 4724,
"HI-TRAC 100D Version IP1 Assembly", Revision 0

(10 pages plus this cover sheet)

CLIENT	ENTERGY IPEC 1

LICENSING DRAWING PACKAGE COVER SHEET

PROJECT NO. 1535	P.O. NO. 450054692
DRAWING PACKAGE I.D. 4724	TOTAL SHEETS 10

LICENSING DRAWING PACKAGE CONTENTS:

SHEET	DESCRIPTION
1	COVER SHEET
2	ASSEMBLY DRAWING AND BILL OF MATERIALS
3	OVERALL DIMENSIONS
4	POOL LID ASSEMBLY
5	BASE PLATE ASSEMBLY
6	OUTER SHELL ASSEMBLY
7	TOP FLANGE DETAILS
8	TRUNNION AND INNER SHELL ASSEMBLY
9	WATER JACKET SHELL ASSEMBLY
10	TOP LID ASSEMBLY

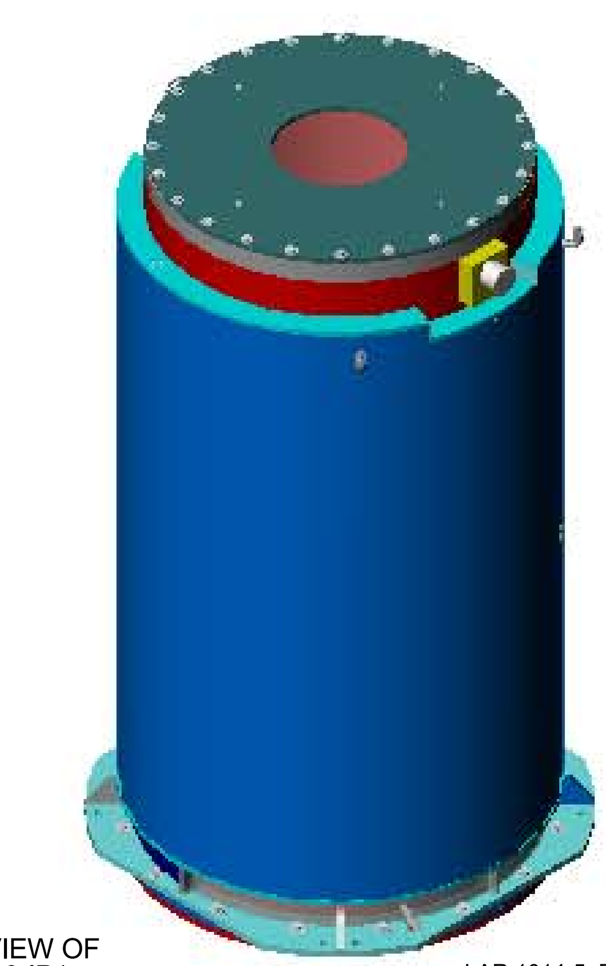
REVISION LOG

REV	AFFECTED DRAWING SHEET NUMBERS	SUMMARY OF CHANGES/ AFFECTED ECOs	PREPARED BY	APPROVAL DATE	VIR# †
0	INITIAL ISSUE	N/A	JJB	05/11/07	39820

† THE VALIDATION IDENTIFICATION RECORD (VIR) NUMBER IS A COMPUTER GENERATED RANDOM NUMBER WHICH CONFIRMS THAT ALL APPROPRIATE REVIEWS OF THIS DRAWING ARE DOCUMENTED IN COMPANY'S NETWORK.

NOTES:

1. THE EQUIPMENT DOCUMENTED IN THIS DRAWING PACKAGE HAS BEEN CONFIRMED BY HOLTEC INTERNATIONAL TO COMPLY WITH THE SAFETY ANALYSES DESCRIBED IN THE HI-STORM FSAR.
2. DIMENSIONAL TOLERANCES ON THIS DRAWING ARE PROVIDED SOLELY FOR LICENSING PURPOSES TO DEFINE REASONABLE LIMITS ON THE NOMINAL DIMENSIONS USED IN LICENSING WORK. HARDWARE IS FABRICATED IN ACCORDANCE WITH THE DESIGN DRAWINGS, WHICH HAVE MORE RESTRICTIVE TOLERANCES, TO ENSURE COMPONENT FIT-UP. DO NOT USE WORST-CASE TOLERANCE STACK-UP FROM THIS DRAWING TO DETERMINE COMPONENT FIT-UP.
3. THE REVISION LEVEL OF EACH INDIVIDUAL SHEET IN THE PACKAGE IS THE SAME AS THE REVISION LEVEL OF THIS COVER SHEET. A REVISION TO ANY SHEET(S) IN THIS PACKAGE REQUIRES UPDATING OF REVISION NUMBERS OF ALL SHEETS TO THE NEXT REVISION NUMBER.
4. APPLICABLE CODES AND STANDARDS ARE DELINEATED IN FSAR SECTION 2.2.4.
5. ALL WELDS REQUIRE VISUAL EXAMINATION. ADDITIONAL NDE INSPECTIONS ARE NOTED ON THE DRAWING. NDE TECHNIQUES AND ACCEPTANCE CRITERIA ARE PROVIDED IN FSAR TABLE 9.1.4.
6. UNLESS OTHEWISE NOTED, FULL PENETRATION WELDS MAY BE MADE FROM EITHER SIDE OF A COMPONENT.
7. THIS COMPONENT IS IMPORTANT-TO-SAFETY, CATEGORY A, BASED ON THE HIGHEST CLASSIFICATION OF ANY SUBCOMPONENT. SUBCOMPONENT CLASSIFICATIONS ARE PROVIDED ON THE DESIGN DRAWING.
8. ALL WELD SIZES ARE MINIMUMS EXCEPT AS ALLOWED BY APPLICABLE CODES AS CLARIFIED IN THE FSAR. FABRICATOR MAY ADD WELDS WITH HOLTEC APPROVAL.



ISOMETRIC VIEW OF HI-TRAC 100-IP1 TRANSFER CASK

LAR 1014-5, RAI-2 Response
May 18, 2007

HOLTEC INTERNATIONAL
HOLTEC CENTER
555 LINCOLN DRIVE WEST
MARLTON, NJ 08053

CLIENT	ENTERGY IPEC 1
DESCRIPTION	HI-TRAC 100D VER. IP1 ASSEMBLY
PROJECT NO.	1535
P.O. NO.	450054692
DRAWING NO.	4724
SHEET	1
TOTAL SHEETS	10
FILE PATH:	N:\DRAWINGS\1535\4724

Figure Withheld Under 10 CFR 2.390

HOLTEC INTERNATIONAL HOLTEC CENTER 555 LINCOLN CENTER MARLTON, NJ 08053	CLIENT: ENTERGY IPEC 1
	DESCRIPTION: HI-TRAC 100D VER. IP1 ASSEMBLY
COMPANION DRAWINGS: -	SIZE: D · DRAWING NO: 4724 · SHEET: 2 · REV: 0
	SCALE: NONE · FILE PATH:

2 1 A

Figure Withheld Under 10 CFR 2.390

HOLTEC INTERNATIONAL HOLTEC CENTER 555 LINCOLN CENTER MARLTON, NJ 08053	ENTERGY IPEC 1
	HI-TRAC 100D VER. IP1 ASSEMBLY
	D | 4724 | 3 | 0
	NONE

A

2

1

Figure Withheld Under 10 CFR 2.390

HOLTEC INTERNATIONAL HOLTEC CENTER 555 LINCOLN CENTER MARLTON, NJ 08053	ENTERGY IPEC 1			
	HI-TRAC 100D VER. IP1 POOL LID ASSEMBLY			
	D	4724	4	0
	NONE			

4692

2 1

D C B A

Figure Withheld Under 10 CFR 2.390

HOLTEC INTERNATIONAL HOLTEC CENTER 555 LINCOLN CENTER MARLTON, NJ 08053	CLIENT: ENTERGY IPEC 1
	DESCRIPTION: HI-TRAC-100D VER. IP1 BASE PLATE ASSEMBLY
COMPANION DRAWINGS	SIZE: D · DRAWING NO.: 4724 · SHEET: 5 · REV: 0
	SCALE: NONE · FILE PATH: N:\DRAWINGS\1538\4724

A

2 1

Figure Withheld Under 10 CFR 2.390

HOLTEC INTERNATIONAL HOLTEC CENTER 555 LINCOLN CENTER MARLTON, NJ 08053	ENTERGY IPEC 1			
	HI-TRAC 100D VER. IP1 OUTER SHELL ASSEMBLY			
	D	4724	6	0
	NONE			

Figure Withheld Under 10 CFR 2.390

HOLTEC INTERNATIONAL	ENTERGY IPEC 1		
	HI-TRAC 100D VER. IP1 TOP FLANGE ASSEMBLY		
D	4724	7	0
	NONE		

Figure Withheld Under 10 CFR 2.390

HOLTEC INTERNATIONAL HOLTEC CENTER 555 LINCOLN CENTER MARLTON, NJ 08053	ENTERGY IPEC 1			
	HI-TRAC 100D VER. IP1 TRUNNION & SHELL ASSEMBLY			
	D	4724	8	0
	NONE			

2 1 A

Figure Withheld Under 10 CFR 2.390

HOLTEC INTERNATIONAL HOLTEC CENTER 555 LINCOLN CENTER MARLTON, NJ 08053	ENTERGY IPEC 1			
	HI-TRAC 100D VER. IP1 ASSEMBLY			
	D	4724	9	0
	NONE			

2 1 A

Figure Withheld Under 10 CFR 2.390

HOLTEC INTERNATIONAL HOLTEC CENTER 555 LINCOLN CENTER MARLTON, NJ 08053	CLIENT ENTERGY IPEC - 1
	DESCRIPTION HI-TRAC 100D VER. IP1 TOP CLOSURE LID
COMPANION DRAWINGS -	SIZE D · DRAWING NO. 4724 · SHEET 10 · REV 0
	SCALE NONE · FILE PATH

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2 1