SAFETY EVALUATION REPORT

DOCKET NO. 72-1014
HOLTEC INTERNATIONAL
HI-STORM 100 CASK SYSTEM
CERTIFICATE OF COMPLIANCE NO. 1014
AMENDMENT NO. 8

SUMMARY

By letter dated November 28, 2009, as supplemented November 4, December 14, 2010, February 25, July 8, and, December 15, 2011, Holtec International (Holtec) submitted amendment request No. 8 to the U. S. Nuclear Regulatory Commission (NRC) for the HI-STORM 100 Certificate of Compliance (CoC) No. 1014. The changes include the following:

1) Addition of a new multi-purpose canister (MPC) – 68M to the approved models presently included in CoC No. 1014 with two new boiling water reactor (BWR) fuel assembly/array classes, and the

2) Addition of a new pressurized water reactor (PWR) fuel assembly array/class to CoC No. 1014 for loading into the MPC-32.

3) CoC No. 1014, Condition No. 3 was revised in accordance with Holtec's commitment to perform ANSI N14.5 He leak testing on confinement boundary base material in addition to confinement boundary welds.

Additionally, the following administrative changes are being provided:

1) CoC No 1014, Condition 5 was revised to add "if applicable" after the reference to Section 3.5 of Appendix B, "Cask Transfer Facility (CTF)." This adds clarification that the CTF is an optional facility.

2) Appendix A, Technical Specifications (TS), Definitions. The CTF definition was modified to clarify that it is an optional facility that can be used in lieu of 10 CFR Part 50 controlled structures for cask transfer evolutions.

3) Appendix A, TS, Table 3-1. MPC Cavity Drying Limits, was changed to correct errors issued in CoC No. 1014, Amendment No. 5 TS. CoC No. 1014, Amendment No. 5, revised TS Table 3-1, but the approved changes were not incorporated in the TS changed pages. This created inconsistencies between it and TS Limiting Condition for Operation 3.1.1.

This amended license, when codified through rulemaking, will be denoted as Amendment No. 8 to the CoC.

This Safety Evaluation Report (SER) documents the review and evaluation of the proposed amendment. The SER uses the same Section-level format provided in NUREG-1536, Rev.1, “Standard Review Plan for Dry Cask Storage Systems,” with some differences implemented for clarity and consistency.

The NRC staff’s (the staff's) assessment is based on whether Holtec meets the applicable requirements of 10 CFR Part 72 for independent storage of spent fuel. The staff’s assessment focused only on modifications requested in the amendment as supported by the submitted revised Final Safety Analysis Report (FSAR) and did not reassess previously approved portions of the FSAR or CoCs through Amendment No. 7.

1.0 GENERAL DESCRIPTION

The objective is to review the design changes made to the HI-STORM 100 Cask System to ensure that Holtec has provided a description that is adequate to familiarize reviewers and other interested parties with the pertinent features of the system, including the changes.

2.0 PRINCIPAL DESIGN CRITERIA EVALUATION

There were no requested changes requiring evaluating the principal design criteria related to the structures, systems, and components (SSCs) important to safety to ensure compliance with the relevant general criteria established in 10 CFR Part 72.

3.0 STRUCTURAL EVALUATION

The objectives of this review were to assess the safety analysis of the structural design features, the structural design criteria, and the structural analysis methodology used to evaluate the expected structural performance capabilities under normal operations, off-normal operations, accident conditions, and natural phenomena events for those SSCs important to safety included in this application.

The review was performed using the appropriate regulations as described in 10 CFR 72.124(a), 72.234(a) and (b), 72.236(b), (c), (d), (l), (g) and (h).

The amendment request that has a direct bearing on the structural aspects of the spent fuel cask storage system is the addition of the MPC-68M. Since only the MPC-68M uses Metamic-HT as a structural material, the evaluation focuses on the MPC-68M. Metamic-HT is also used as neutron absorber.

3.1 MPC-68M

The MPC-68M contains a new type fuel basket made entirely of Metamic-HT which can hold 68 BWR fuel assemblies inside the existing MPC enclosure vessel. The basket is made of interconnected Metamic-HT panels. The bounding weights for the basket is 30,000 lbs (13,607 kg) unloaded (the lightest MPC for this system) and 90,000 lbs (40,823 kg) with fuel and spacers.

Because of the lighter weight of the MPC-68M, as well as the load bearing capacity of the unchanged overpack, most of the load cases analyzed previously bound the applicable loadings with the MPC-68M. The worst load conditions for the MPC-68M basket are the tipover and end drop case.

For the tipover event, the applicant performed an analysis based on a ratio of mass and geometry of the MPC-68M to the previously analyzed HI-STORM 100 configuration to establish the maximum rigid-body deceleration. The applicant determined, based on this revised analysis, that the maximum deceleration for the tipover event with the MPC-68M would be 43.42g, and this is less than the design-basis limit of 45g identified in Table 3.1.2 of the FSAR.

In order to ensure compliance with the dimensionless deformation limit (defined as the maximum total deflection sustained by the basket panels under the loading event over the nominal inside (width) dimension of the storage cell) of 0.005 set forth in the FSAR (Table 2.III.4) the applicant performed a finite element analysis applying a 70 g deceleration on the basket. The calculated results show only small plastic strains that do not significantly alter the basket structure. This demonstrates a significant margin between available basket strength and the design basis loads.

For the end drop event, the applicant established a proportion by weight to the deceleration of the previously analyzed configuration. The resultant maximum deceleration for the end drop for the system using an MPC-68M is 44.39g, which is below the design basis limit of 45g stated in Table 3.1.2 of the FSAR. The applicable maximum lift height of 11 in (28 cm), previously established, still applies.

3.2 Evaluation Findings

Based on evaluation of amendment request 1014-8, supporting documentation and calculations, the staff finds that amendment request 1014-8 acceptably meets the review criteria identified in NUREG-1536, Rev 1. Specifically, the staff finds:

F3.1 The application has met the requirements of 10 CFR 72.122(b) and (c). The SSCs important to safety are designed to accommodate the combined loads of normal, off-normal, accident, and natural phenomena events with an adequate margin of safety. Stresses at various locations of the cask for various design loads are determined by analysis. Total stresses for the combined loads of normal, off normal, accident, and natural phenomena events are acceptable and are found to be within limits of applicable codes, standards, and specifications.

F3.2 The application has met the requirements of 10 CFR 72.124(a), “Criteria for Nuclear Criticality Safety,” and 10 CFR 72.236(b), “Specific Requirements for Spent Fuel Storage Cask Approval and Fabrication.” The structural design and fabrication includes acceptable structural margins of safety for those SSC important to nuclear criticality safety. The applicant has demonstrated acceptable structural safety for the handling, packaging, transfer, and storage under the normal, off-normal, and accident conditions that are identified in the FSAR.

F3.3 The application continues to meet the requirements of 10 CFR 72.236(l), "Specific Requirements for Spent Fuel Storage Cask Approval and Fabrication." The staff evaluated the submitted analyses and supporting documentation and determined that the applicant has acceptably demonstrated that the cask and other systems

important to safety continue to maintain confinement of radioactive material under normal, off-normal, and credible accident conditions identified in the FSAR.

F3.4 The application has met the requirements of 10 CFR 72.122, "Overall Requirements," and 10 CFR 72.236, “Specific requirements for spent fuel storage cask approval and fabrication,” with regard to inclusion of the following provisions in the structural design:
- design, fabrication, erection, and testing to acceptable quality standards
- adequate structural protection against environmental conditions and natural phenomena, fires, and explosions
- appropriate inspection, maintenance, and testing
- adequate accessibility in emergencies
- a confinement barrier that acceptably protects the spent fuel cladding during storage
- structures that are compatible with appropriate monitoring systems
- structural designs that are compatible with ready retrievability of spent fuel.

F3.5 The application has met the specific requirements of 10 CFR 72.236(e), (f), (g), (h), (i), (j), (k) and (m), as they apply to the structural design for spent fuel storage cask approval. The cask system structural design continues to acceptably provide for the following required provisions:
- redundant sealing of confinement systems
- adequate heat removal without active cooling systems
- storage of the spent fuel for a minimum of 20 years
- compatibility with wet or dry spent fuel loading and unloading facilities
- acceptable ease of decontamination
- inspections for defects that might reduce confinement effectiveness
- conspicuous and durable marking
- compatibility with removal of the stored fuel from the site, transportation, and ultimate disposition by the U.S. Department of Energy

4.0 THERMAL EVALUATION

4.1 Review Objectives

The objectives of this review were to assess the safety analysis of the thermal design features, the thermal design criteria, and the thermal analysis methodology used to evaluate the expected thermal performance capabilities under normal operations, off-normal operations, accident conditions and natural phenomena events for those SSCs important to safety included in this application.

The MPC-68M is designed for storage under the array of uniform and regionalized heat load. The MPC-68M thermal design is the same as that of the currently licensed MPC-68 and is pressurized with helium to the same backfill pressure of 31.3 psig, defined for the MPC-68 in FSAR Table 4.4.14. The principal differences between the proposed MPC-68M and the licensed MPC-68 are:

- the basket panel in MPC-68M is made of Metamic-HT, instead of the composite cell walls for MPC-68,
- the aluminum basket shims are inserted between the basket external cell walls and the MPC shell for the MPC-68M basket. No aluminum shims were present in the MPC-68 basket design, and
- the fuel storage cell dimension, basket wall thickness, and size of the flow holes in MPC-68M basket are different from the MPC-68 basket.

In Supplement III for amendment request 1014-8, the applicant evaluated the thermal performance of MPC-68M under normal, off-normal, and accident conditions, and during short-term operations.

4.2 Evaluation of MPC-68M

4.2.1 Input Parameters

The applicant applied the material properties, applied loads, specified boundary conditions, and component geometries in the thermal analysis. The applied loads includes decay heat loads of 36.9 kW and solar heat loads, as described in FSAR Section 4.4.1.1.8 as well as the quantities of backfill gas and gaseous fission products contained within the MPC. The boundary conditions include the normal and off-normal ambient temperatures, 27°C (80°F) and 38°C (100°F). The exposed surface heat transfer coefficients are also defined in the FSAR. The material properties, applied loads and boundary conditions used in all of the analyses for the MPC-68M are identical to those used for MPC-68. The thermal properties of fuel basket (Metamic-HT) and basket shims (Aluminum Alloy 2219) for MPC-68M are provided in Table 4.III.1 of the FSAR, Report HI-2002444, Supplement III for amendment request 1014-8.

The applicant utilized the conservative backfill gas pressure of 48.5 psig at 21°C (70°F) for the MPC internal pressure calculations and a conservative operating pressure of 5 atm (or 73.5 psia) for the MPC temperature distribution calculations. The staff reviewed the calculations provided in the FSAR, and the FSAR Supplement III and ensured that even with the over-estimated pressure of 5 atm, the predicted MPC temperature is still below the allowable temperature limit. Therefore, the staff confirmed that the cask is designed to have adequate heat removal capacity without active cooling and the adequate information of thermal evaluation is provided in Supplement III to satisfy the requirements of thermal evaluation, in accordance with 10 CFR 72.122(h)(1), and 10 CFR 72.236(f).

4.2.2 Thermal-Hydraulics Model for MPC-68M

The MPC-68M thermal design in the amendment request is the same as that of the currently approved MPC-68 with MPC basket design to hold 68 BWR fuel assemblies. The applicant modeled MPC-68M with key features:

- The decay heat is non-uniformly distributed over the active fuel length based on the design basis axial burnup distributions,
- The MPC internal helium circulation is considered laminar flow,
- The heat transport from MPC interior to its outer surface is by a combination of conduction through the MPC basket metal grid structure, and conduction and radiation heat transfer in the relatively small helium gaps between fuel assemblies and basket cell walls,
- The heat dissipation across the gap between MPC basket periphery and MPC shell is by a combination of conduction and radiation,
- The heat rejection from the MPC outer surface to the ambient air is primarily accomplished by convective heat transfer to a buoyancy driven airflow through the MPC-to-overpack annular gap,
- The heat rejection from the MPC outer surface to the ambient air is also accomplished by thermal radiation heat transfer across the annular gap, radial conduction through the

overpack cylinder, and combined natural convection and thermal radiation from the overpack outer surface to the environment,

- The air flow through the annular gap between the MPC and the overpack is characterized by the k-ω turbulence model with the transitional option enabled,
- The air flow in the inlet and outlet vents, and annular gap between the MPC and the concrete inner shell is in transitional regime which allows the circulation of air through the annulus,
- The flow resistance in the fuel assembly is simulated as the 3-zone porous media hydraulic resistance for the porous media to represent the loaded fuel basket in the MPC, which is based on the rigorous computational fluid dynamics modeling of the fuel assembly geometry, and
- The underside of the HI-STORM 100 Cask System concrete pad is assumed to be supported on a subgrade at 25°C (77°F).

The staff reviewed Supplement III for amendment request 1014-8 and found that the model approach for MPC-68M is consistent with the Holtec CoC No.1014, Amendment No. 5, as previously found to be acceptable by NRC.

4.2.3 Thermal Interference

The staff determined that the applicant designed the MPC-68M with adequate gaps to permit free thermal expansion of the fuel basket and MPC in axial and radial directions, and determined the changes in gaps using the temperature field, calculated from the FLUENT thermal model, in the MPC-68M and HI-STORM overpack. The staff reviewed the initial minimum gaps (cold gaps) and their corresponding thermal expansion values (differential expansion) tabulated in Supplement III, Table 4.III.8, of amendment request 1014-8 and found that the thermal expansions and the thermal stresses of the fuel basket and the MPC remain within acceptable design limits identified in NUREG-1536, Rev. 1, and the FSAR.

4.2.4 Removal of Time Limit for MPC-68M during Moisture Removal Operations

The applicant determined the maximum cladding temperature of the MPC-68M under the following vacuum drying scenarios:

(A) MPC-68M is loaded with Moderate Burnup Fuel assemblies (as defined in the FSAR) generating heat at the maximum permissible rate under the bounding regionalized storage scenario. This is a decay heat of 36.9 kW, and the
(B) MPC-68M is loaded with one or more high burnup fuel assemblies (as defined in the FSAR), and this decay heat value is less than the previously staff evaluated heat load of 29.0 kW per assembly.

The staff evaluated the maximum MPC-68M temperatures under the vacuum drying scenarios (Table 4.III.5 of Supplement III and Table K.5 of Holtec Report HI-2043317) and found that all the component temperatures are below the corresponding temperature limits for both scenarios (A) and (B).

In scenario (A), the calculated cladding temperature of 401°C (754°F) is above the 400°C (752°F) evaluation guidance provided in Interim Staff Guidance (ISG)-11, Rev. 3. The 400°C limit is based on (1) the condition of hoop stress and hydride reorientation in identified in ISG-11, and (2) the cladding temperature limit of 570°C (1058°F) for moderate burnup fuel under

short term operations such as the moisture removal process as described in the FSAR. However, based on the staff's evaluation, it is expected that the fuel assemblies with the moderate burnup, as described in scenario (A), are not likely to have a significant amount of hydride reorientation due to limited hydride content. Furthermore, most of the low or moderate burnup fuel has hoop stresses below 90 MPa. Even if hydride reorientation occurred during storage, the network of reoriented hydrides is not expected to be extensive enough in moderate burnup fuel to cause fuel rod failures. Given the conditions of hydride reorientation and hoop stress described above and the fact that the calculated temperature is just 1°C (2°F) over the allowable limit, the staff finds the cladding temperature of 401° in scenario A is acceptable for this application. This is based on the applicant's use of other conservative assumptions (e.g., the water in the HI-TRAC annulus is conservatively assumed to be boiling with a water temperature of 111°C (232°F) in the calculation. Additionally, the hydrostatic head of water at the annulus with the MPC bottom surface insulation causes boiling at higher than 100°C (212°F) used in the model analyses, and that the fuel rods in MPC-68M should not fail during the moisture removal operations in Scenario (A). In scenario (B), the calculated cladding temperature of 389°C (732°F) is well below the allowable limit of 400°C (752°F) as identified in ISG-11.

The staff found the evaluations of scenarios of (A) and (B) acceptable based on two conservative assumptions: (1) the water in the HI-TRAC annulus is assumed to be boiling 111°C (232°F) under the hydrostatic head of water at the annulus bottom and (2) the bottom surface of the MPC is insulated. The staff finds that the maximum cladding temperatures under scenarios (A) and (B) are in compliance with thermal limits review guidance provided in ISG-11, Rev 3.

4.2.5 HI-TRAC Onsite Transfer Operation

The applicant evaluated the thermal performance of the MPC-68M contained in a HI-TRAC (with decay heat of 36.9 kW and HI-TRAC annulus filled with air) and found that a supplemental cooling system (SCS) is not necessary for ensuring cladding safety under the onsite transfer of the MPC-68M. The staff reviewed Table 4.III.6 of Supplement III and Table K.4 of Holtec Report HI-2043317 and determined that the maximum cladding temperature of 338°C (640°F) is below the temperature limit of 400°C (752°F) for normal conditions of storage and short-term loading operations, per the guidance of ISG-11, Rev. 3. The staff found that the MPC-68M has:

(1) lower component temperatures on the fuel cladding, the MPC basket, and the MPC shell and gives a higher temperature margin when compared to the MPC-68,

(2) higher HI-TRAC shell temperature of 166°C (331°F), but still below the limit of 500°C (932°F), and

(3) the aluminum shim temperature of 276°C (528°F) is lower than the short-term operation temperature limit of 500°C (932°F). Therefore from a thermal evaluation perspective, a SCS is not required for the MPC-68M.

The applicant calculated the pressures of 102.1 psig, for 1% rods rupture, which is above the allowable limit of 100 psig and 106.9 psig, for 10% rods rupture, which is below the allowable limit of 110 psig. Given the conditions that instead of 31.3 psig defined in the FSAR, a higher backfill pressure of 48.5 psig is used for evaluation; therefore a predicted pressure of 102.1 psig for 1% rods failure is conservative and acceptable for HI-TRAC onsite transfer operation.

4.2.6 Impact of Fuel Debris on the Intact Fuel Assemblies

The applicant stated that up to 16 damaged fuel containers (DFCs) containing BWR damaged fuel assemblies and/or up to 8 DFCs containing fuel debris may be stored in the MPC-68M with the remaining fuel storage locations filled with undamaged BWR fuel assemblies. The fuel debris can be in a type of rubble that may be concentrated in a smaller area and create hot spots in the cask and increase the cladding temperatures of the adjacent intact assemblies. Although the fuel debris is not required to meet cladding temperature limits, its effects on the fuel rods stored in the interior cells must be assessed. Therefore, the applicant performed the thermal analysis to ensure that both pressure and fuel cladding temperatures are below the limits for the impact of fuel debris on the intact fuel assemblies.

The applicant performed the thermal analysis by assuming that:

(1) the fuel debris is completely pulverized and compacted into a bar enclosed by DFC,

(2) the height of the bar emitting heat is minimized to maximize the linear thermal loading of DSC and the co-incident local heating of the fuel basket and neighboring storage cells,

(3) the fuel debris is assumed to be completely composed of UO_2 with a lower thermal conductivity relative to cladding and therefore the heat dissipation is understated, and

(4) all 16 peripheral storage locations (not just 8 permitted in CoC) contain fuel debris emitting the maximum heat permitted by TS and all interior cells are emitting design heat under the uniform loading storage scenario.

The staff evaluated the submitted revised FSAR 4.III.11 pages for HI-STORM temperatures under fuel debris storage, and found that the peak cladding temperature of 306°C (583°F) is acceptably below the allowable limit of 400°C (752°F) and the temperatures of basket and aluminum shims are below the allowable limits provided in ISG-11, Rev. 3.

4.2.7 Normal Long-Term Storage

The applicant calculated the cask component temperatures and the maximum pressure of the MPC-68M for a decay heat of 36.9 kW and tabulated them in Tables 4.III.3 and 4.III.4 of Supplement III and Tables K.1 and K.3 of Holtec Report HI-2043317 and specified that the temperatures of the cladding and other cask components are below the allowable limits and the MPC internal pressures are within the safety margins for intact rods and 1% rods rupture. The staff checked the allowable limits in the FSAR and the calculated data in Supplement III, and confirmed that 1) the MPC-68M has lower maximum temperatures of fuel cladding, basket, and MPC shell. Therefore, MPC-68M provides higher temperature margins than MPC-68, and 2) all the cask components are maintained within their temperature limits and MPC internal pressure is below the criteria of 100 psig for the long-term normal storage conditions. The staff confirmed that the calculated temperatures of the cask components comply with the design criteria and

meet the requirements of 10 CFR 72.122(h)(1), and 10 CFR 72.236(f) to ensure adequate heat removal is provided.

4.2.8 Off-Normal Conditions

Elevated Ambient Air Temperature

The principal effect of the elevated ambient temperature is a rise of the HI-STORM 100 temperatures from the baseline normal storage temperatures by the difference between elevated ambient and normal ambient temperatures. The normal storage temperature under MPC-68M storage in the HI-STORM 100 overpack is bounded by the temperatures of the MPC-68 (FSAR 4.4). The staff accepted this conclusion because of the better heat transfer capability of MPC-68M in which the fuel basket is entirely made of the highly conducting Metamic-HT.

Partial Blockage of Air Inlets

The principal effect of the partial inlets blockage is a rise in the HI-STORM 100 annulus temperatures from the baseline normal storage temperatures and to a similar rise in the MPC temperatures. The normal storage temperature under the MPC-68M storage in the HI-STORM 100 overpack is bounded by the temperatures of the MPC-68 (FSAR 4.4). The staff fines this conclusion acceptable because of better heat rejection capability of the MPC-68M that uses the highly conducting Metamic-HT fuel basket.

Off-Normal Pressure

The off-normal pressure event is defined as a combination of (a) maximum helium backfill pressure, (b) 10% fuel rods rupture, and (c) limiting fuel storage configuration. The applicant predicted a pressure of 100.5 psig (Supplement III, Table 4.III.4) which is below the off-normal design pressure of 110 psig. The staff checked the FSAR Table 2.2.1 and accepted the pressure margin of 9.5 psi because a conservative value of 48.5 psig is used as the initial backfill pressure. The staff confirmed that the MPC-68M meets the criteria of the FSAR, using ANSI/ANS 57.9 as the basis for the off-normal pressure event.

4.2.9 Accident Conditions

The applicant provided the evaluation of the following accidental conditions to demonstrate that the MPC-68M meets the design criteria limits, in compliance with Part 72.

(a) Fire

The principal effect of the fire accident is a temperature increment in both stored fuel and MPC during HI-STORM storage or under on-site transfer in the HI-TRAC. The applicant stated that the temperatures in the MPC-68M are bounded by those in the MPC-68. Given the same decay heat load, but the better heat rejection capability of the MPC-68M with the Metamic-HT as the fuel basket material, it's possible that (1) the steady-state fuel cladding temperature field for MPC-68M can be lower than that for MPC-68, and (2) more heat can be transferred into MPC-68M than into MPC-68 during a fire event.

The applicant performed a new analysis for the fire accident when the Metamic-HT is used as the fuel basket. The staff performed the confirmatory calculation of temperature rise and checked the resulting fire accident pressure documented in Table 4.III.9 and the fuel temperature rise computed in Supplement 4.III.6.2 in the revised FSAR. The staff found that the maximum pressure of 104.5 psig is below the allowable limit of 200 psig provided in the FSAR, under accident conditions, and the small fuel temperature rise of 0.6^{o}C (1.0^{o}F) does not adversely affect the temperature of the MPC or contained fuel.

(b) Burial under debris

The applicant used a lumped capacitance model that combines the thermal capacity of the MPC and the HI-STORM overpack to analyze the accident condition of the complete burial of the HI-STORM 100 overpack under an indeterminate material. The staff found that the previously approved evaluation of the MPC-68 remains bounding for the MPC-68M because the initial storage temperatures under the MPC-68M are less than the initial storage temperatures under the MPC-68.

(c) Blockage of air ducts

This accident is defined as 100% blockage of the air inlet ducts for 32 hours. The applicant evaluated the MPC-68M under this accident under the decay heat of 36.9 kW and computed the 32-hour temperature rise of the MPC and the stored fuels. The staff reviewed the maximum temperatures and the maximum pressure of Table 4.III.7 of Supplement III and Table K.6 of Holtec Report HI-2043317, and ensured that the MPC internal pressure of 111.6 psig is below the allowable limit of 200 psig and both fuel cladding and component temperatures also remain below their respective accident temperature limits that are specified in HI-STORM FSAR Table 2.2.1 for a 32-hour 100% air inlet blockage accident.

(d) Flood

The flood accident is defined in the FSAR as a deep submergence event. The worst flood event from a thermal evaluation is that in which water rises to the top of the inlets to the point that airflow is prevented and along with preventing the MPC cooling by water. The staff found that this event is bounded by the previously evaluated 100% blockage of air inlet ducts accident event, and this is acceptable.

(e) Extreme Environmental Temperature

The principal effect of the elevated ambient temperature is a rise of the temperature in HI-STORM 100 Cask System from the baseline normal storage temperatures by the difference between the elevated ambient and the normal ambient temperatures. The applicant stated that as the normal storage temperature under MPC-68M storage in the HI-STORM 100 overpack are bounded by the temperatures in the HI-STORM 100 Cask System; the temperatures under this event are also bounded by the extreme ambient evaluation. The staff found this conclusion acceptable because of better heat rejection capability of the MPC-68M in which the fuel basket is entirely made of the highly conducting Metamic-HT material.

(f) 100% fuel rods ruptured

The applicant evaluated the 100% rods failure accident by assuming the release of 100% of the rods fill gases and fission gases in accordance with NUREG-1536, REV. 1, release fractions. A computed MPC-68M internal pressure of 145.8 psig is listed in Supplement III Table 4.III.4. The

staff reviewed the FSAR and ensured that the maximum internal pressure of 145.8 psig is below the allowable limit of 200 psig.

(g) Jacket Water Loss

The principal effect of the jacket water loss accident is a temperature rise in the stored fuel inside the MPC from the baseline conditions while being transferred in the HI-TRAC. The applicant stated that as the temperature limit in the MPC-68M is bounded by the MPC-68

temperatures, the jacket water loss temperatures in MPC-68M are therefore bounded by the HI-TRAC jacket water loss evaluation in MPC-68 system. The staff found this acceptable because of the better heat rejection capability of the MPC-68M.

4.3 Addition of the CE 15x15 (15x15I) Fuel Assembly Array to MPC-32

The applicant proposed to add a new PWR fuel assembly to CoC No.1014 for loading into the MPC-32. The allowable heat load limits per assembly remain unchanged for the MPC-32. Therefore all thermal analyses already performed for the MPC-32 in the HI-STORM 100 Cask System bound the addition of the 15x15I fuel assembly array/class. The staff found this acceptable from a thermal perspective because the heat load is unchanged.

4.4 Evaluation Findings

F4.1 The staff found that the calculated fuel cladding temperatures are below the ISG-11 temperature limits of Moderate Burnup Fuel (Scenario A) and High Burnup Fuel (Scenario B) for normal conditions and 570°C (1058°F) for off-normal and accident conditions, and other cask component temperatures are maintained below the maximum for the accidents evaluated. Based on the temperature and pressure calculations provided in the FSAR, the staff found that the thermal-hydraulic performance of the MPC-68M system components meets the review criteria provided in ISG-11 and is acceptable from a thermal-hydraulic perspective and meets the requirements of 10 CFR 72.122(h)(1) and 72.236(f).

F4.2 The supplemental cooling system is not required for MPC-68M during HI-TRAC onsite transfer operation because the increased thermal conductivity of the Metamic-HT basket maintains the steady-state fuel cladding temperatures below allowable limits.

F4.3 The spent fuel cladding is protected against degradation that leads to gross ruptures by maintaining the cladding temperature for the approved contents below 570°C (1058°F) for normal, and off-normal conditions. Protection of the cladding against degradation will allow ready retrieval of spent fuel assembly for further processing or disposal as required by 10 CFR 72.122(h)(1).

F4.4 As applicable as part of this amendment request, the application includes acceptable analyses of the design and performance of SSCs important to safety under normal, off-normal and accident scenarios, in compliance with 10 CFR 72.236.

F4.5 The MPC-68M, designed to accommodate 16 DFCs containing BWR damaged fuel assemblies and/or up to eight DFCs containing fuel debris, is evaluated to have no impact on the adjacent intact fuel assemblies or the cask and meet the requirements of 10 CFR 72.122(h)(1).

F4.6 The analyses of off-normal and accident events and conditions and reasonable combinations of these and normal conditions show that the design and operation of the HI-STORM 100 Cask System, will continue to meet the requirements without endangering the public health and safety in compliance with the requirements of 10 CFR 72.236.

5.0 CONFINEMENT EVALUATION

The objective of the confinement evaluation of the amendment request 1014-8 is to ensure that radiological releases to the environment continue to remain within the limits established by 10 CFR 72.104(a) and 10 CFR.106(b), and that the spent fuel cladding and fuel assemblies will be sufficiently protected during storage against degradation that otherwise might lead to gross ruptures. Based on the thermal review/evaluation of amendment request 1014-8, the fuel cladding temperatures are continue to be maintained below the corresponding limits for normal, off-normal and accident conditions and the heat removal capability is maintained. Therefore, the fuel cladding and assembly will be protected. The objective includes review of the confinement design characteristics and confinement analyses for the HI-STORM 100 Cask System proposed in the application.

The confinement boundary includes the MPC shell, the bottom baseplate, the MPC lids (including the vent and drain port cover plates), the MPC closure ring, and the associated welds. There are two confinement boundary penetrations, the MPC vent and drain ports. All components of the confinement boundary are important to safety, Category A, as specified in the FSAR Table 2.2.6. The MPC confinement boundary is designed, fabricated and inspected in accordance with the American Society of Mechanical Engineers Boiler and Pressure Vessel Code (Code), Section III, Subsection NB to the maximum practical extent. NRC approved alternatives to the ASME Code are identified in CoC No. 1014, Appendix B, Table 3-1.

The list of MPC models for use in the HI-STORM 100 Cask System has been expanded to include the MPC-68M which consists of a Metamic-HT BWR fuel basket inside the existing MPC enclosure vessel. All design aspects of the MPC enclosure vessel (confinement boundary), including the vent and drain port arrangements in the MPC lid, are unchanged due to the addition of the MPC-68M and the new BWR fuel assembly array/classes 10x10F and 10x10G. The MPC-68M may be loaded in all licensed aboveground HI-STORM 100 overpacks, but not in the 100U version of the system. No changes were made to the MPC-32 enclosure vessel due to the addition of the 15x15I fuel assembly array/class; therefore all confinement evaluations already performed for the MPC-32 in the HI-STORM 100 Cask System bound the addition of the 15x15I fuel assembly array/class. Fuel classified as damaged fuel assemblies or fuel debris to be loaded in the MPC-68M is placed in damaged fuel containers.

In addition, the staff has determined that for an applicant to use ANSI N14.5, “Radioactive Materials - Leakage Tests on Packages for Shipment,” as the basis for leak-tight certification the entire confinement boundary including the confinement welds and base material (i.e., the lid, shell, baseplate, and vent and drain port covers) are required to be helium leak tested. Holtec uses ANSI N14.5 leak testing to certify leak-tightness of the confinement boundary. Therefore, leakage is not considered credible during normal and accident conditions. The staff has reviewed how the base material leak testing additions have been incorporated as proposed changes to Tables 2.0.1 and 9.1.1 of the FSAR Rev. 9, but for the rest of the review the staff referenced FSAR Rev. 7. Text changes to the FSAR associated with the proposed changes to Tables 2.0.1 and 9.1.1 will be updated after the final rule approving the amendment becomes effective. Holtec has committed to testing the base metal of all MPC shells, including the baseplate and lid, in addition to the fabrication welds, beginning on the date the final rule approving the amendment becomes effective. Therefore, CoC condition 3 has been revised to be consistent with these commitments. Also, due to the inability to perform the visual examination of inaccessible portions of the MPC shell welds during the field Code hydrostatic or pneumatic pressure test, the results of helium leakage rate testing applied under ANSI N14.5

standards has been accepted as an alternative based on the review guidance in ISG-25. Based on this, the staff has provided an approved alternative to the Code in CoC No. 1014, Appendix B, Table 3-1.

5.1 Evaluation Findings

Based on the staff's evaluation of information provided in the amendment request 1014-8, the staff finds the amendment acceptable based on the following:

F5.1 The design changes to the HI-STORM 100 Cask System continue to protect the spent fuel cladding against degradation that might otherwise lead to gross ruptures. .

F5.2 The staff finds that the design changes to the confinement system of the HI-STORM 100 Cask System continue to ensure that releases remain in compliance with 10 CFR 72.104(A) and 10 CFR 72.106(b).

6.0 SHIELDING EVALUATION

The objective of this review is to evaluate that amendment request 1014-8 continues to meet the external radiation requirements of 10 CFR Part 72 under normal conditions of transfer and analyzed accident conditions. The proposed changes affecting the shielding analysis are: addition of new BWR (10x10F and 10x10G) and PWR (15x15I) assemblies to the list of authorized contents; addition of the MPC-68M as an authorized canister; and use of a METAMIC fuel basket.

The staff shielding review evaluated the changed features in conjunction with the findings from previous staff analysis to determine whether these changes provide adequate protection from the radioactive contents. This review looked at the methods and calculations employed by Holtec to determine the expected gamma and neutron radiation at locations near the cask surface and at specific distances away from the cask.

6.1 Shielding Design Description

6.1.1 Design Features

The HI-STORM 100 Cask System consists of a steel canister with a concrete overpack. Gamma shielding is provided by the steel and concrete, with the concrete also providing neutron shielding.

The MPC-68M is a variation of the MPC-68 BWR canister previously approved. The Metamic HT basket design of the MPC-68M consists of aluminum oxide and ground boron carbide dispersed in a metal matrix of pure aluminum. The differences between the MPC-68M and the MPC-68 important to shielding are:

- MPC-68M has slightly higher B-10 content
- MPC-68M is lighter since the basket contains no steel
- In the enclosure shell, the MPC-68M is surrounded by aluminum basket shims

The MPC-24 and MPC-32 classes of PWR fuel canisters remain unchanged.

6.2 Radiation Source

The two BWR fuel assembly designs/classes added to the HI-STORM 100 Cask System are the 10x10F and 10x10G. In terms of radiological characteristic, the 7x7 class of BWR fuel is bounding fuel assembly. The new BWR designs have been grouped with fuel classes with a larger heavy metal loading. The stricter limits on burnup are conservative when applied to a fuel assembly class with a lower mass of uranium.

The PWR fuel assembly design/class added is the 15x15I. The applicant has listed the characteristics of the 15x15I in Table 2.1-2 of CoC No. 1014, Appendix P and shown them to be bounded by the B&W 15x15 class of assemblies on the basis of heavy metal loading.

6.3 Shielding Model

No additional shielding model was supplied by the applicant.

6.4 Shielding Evaluation

The applicant supplied a series of qualitative analyses to show the bounding shielding evaluation is still applicable to the MPC-68M. The applicant references the previously reviewed shielding analysis which states that the inner 32 assemblies, comprising 47% of the spent fuel, contribute 2% of the gamma dose and 27% of the neutron dose due to the shielding by the outer assemblies. This effect minimizes the impact that the basket material has on external dose rates. Neither the MPC-68 nor the overpack are changed in amendment request 1014-8, and the 10x10F and 10x10G source term is bounded by the B&W 7x7 fuel class previously evaluated by the staff. Additional shims in the annulus surrounding the basket are provided in the MPC-68M, and the applicant has shown that the total shielding provided by the MPC-68M has not significantly changed from the MPC-68 previously evaluated acceptably by the staff. Thus, the staff finds the MPC-68M acceptable.

6.4.1 Confirmatory Analyses

The staff compared the newly added assembly designs to the design-basis assembly in the previous analyses. Using the SAS2H module in SCALE 5.1, the staff was able to confirm that the new BWR assemblies are bounded by the 7x7 class. The PWR 15x15I, while not demonstrated to be bounded by the design-basis analysis, will likely yield a source term comparable to the B&W 15x15 class burnup and enrichment.

The staff also conducted a shielding comparison using an arbitrary gamma line source to determine the effect of using an aluminum basket in the MPC-68M. The distance from a point on the edge of zone 1 was chosen and a path drawn radially outward to the enclosure vessel inner surface. Since the vessel itself is unchanged, the shielding characteristics have already been evaluated. Microshield models were made using the thickness of the materials encountered on several paths outward from this point. The difference among these models depends on geometry and material assumptions. In all cases the steel enclosure vessel provided significant gamma shielding. During operation, the magnitude of this difference compared to the shielding provided by the overpack or transfer vessel is small.

6.5 Evaluation Findings

F6.1 Based on the information provided by the applicant to support amendment request 1014-8, the staff finds that changes FSAR Sections 1, 2 and 5 acceptably describe the changes to shielding structures, systems, and components important to safety in sufficient detail to allow evaluation of their effectiveness.

F6.2 The staff finds that the changes requested in amendment request 1014-8 to the shielding system of the HI-STORM 100 Cask System are in compliance with 10 CFR 72.104, and 10 CFR 72.106, and that the applicable design and acceptance criteria provided in NUREG-1536, Rev. 1 have been satisfied as discussed above in Sections 6.1.1, 6.2, 6.4 and 6.4.1 of this SER.

7.0 CRITICALITY EVALUATION

The objectives of this evaluation were to determine that amendment request 1014-8 remains in compliance with 10 CFR 72.124, and 10 CFR 72.236(a), (b), (c), (g), (h) and (m) with respect to criticality safety. The staff assessed the criticality safety analyses provided under normal operations, off-normal operations, accident conditions and natural phenomena events for those SSCs important to safety. The staff's evaluation of the criticality safety of the amendment follows:

7.1 Addition of MPC-68M

The MPC-68M is distinguished from the other HI-STORM 100 BWR fuel baskets (MPC-68, MPC-68F and MPC-68FF) in that the basket is made of METAMIC-HT material. This material acts as the structural material as well as the neutron absorber material.

7.1.1 Fuel Specification

The MPC-68M is designed to accommodate up to 68 intact BWR fuel assemblies. The DFCs are allowed in the peripheral fuel locations described in Section 1.III.2.3 and Figure 1.III.2 of the FSAR.

The fuel assemblies that are authorized to be stored in the MPC-68M are the 7x7B, 8x8B, 8x8C, 8x8D, 8x8E, 8x8F, 9x9A, 9x9B, 9x9C, 9x9D, 9x9E, 9x9F, 9x9G, 10x10A, 10x10B, 10x10C, 10x10F, and 10x10G. Except for the 10x10F and 10x10G, these assembly classes are described in Table 2.1.4 of the HI-STORM 100 FSAR (Revision 7, August 9, 2008). The enrichment limits for storage in the MPC-68M are higher than that of the other HI-STORM 100 MPCs and these are defined in Table 2.III.2 of the revised FSAR.

The 10x10F and 10x10G are only authorized for use in the MPC-68M (unlike the other assemblies listed above that are previously approved for storage in other canisters such as the MPC-68, MPC-68F or the MPC-68FF). The description, including enrichment limits, for these assembly classes is located in Table 2.III.3 of the FSAR.

Although the various fuel assembly classes may be vendor and utility specific fuel, they are defined using generic specifications. These include:

- Lattice (array)
- Clad Material

- Uranium weight
- Maximum Planar-Average Initial Enrichment
- Initial Rod Maximum Enrichment
- Number of fuel rods (full and part-length rods)
- Minimum Fuel Clad OD
- Maximum Fuel Clad ID
- Maximum Fuel Pellet Diameter
- Maximum Fuel Rod Pitch
- Maximum Active Fuel Length
- Number of Water Rods
- Minimum Water Rod Thickness
- Maximum Channel Thickness

In addition to the above fuel specifications, there are footnotes that define each fuel type.

The applicant provided information in revised Section 6.III.4.2 of the FSAR discussing why these specifications are limiting. The applicant states that the above characteristics were found to be most reactive based on the assumption that the fuel assemblies are undermoderated. The applicant provided calculations (FSAR Table 6.III.4.2) demonstrating that all of the fuel assemblies are undermoderated. Previous analyses also demonstrate that these fuel specifications are limiting. The staff finds this acceptable.

The applicant is using planar averaged enrichments to specify BWR fuel rather than maximum enrichment. The staff finds this acceptable due to the following. The applicant performed analysis using 8x8, 9x9, and 10x10 assembly types, including higher enrichments, within the geometry of the MPC-68M to compare the use of planar averaged enrichments to discrete radial enrichments and showed the results in FSAR Table 6.III.2.1. Since the actual (as-built) enrichment distributions were not available, the applicant assumed a conservative enrichment distribution and found that for the 10x10 assemblies it is possible that the distributed enrichments may be slightly more reactive. Therefore, the applicant added a small bias to the k-eff for these configurations. .

The applicant used a value of 10.686 g/cm^3 for UO_2 density. This is 97.5% of the theoretical density. This is a conservative value since it corresponds to a very high pellet density of 99% or more of the theoretical density.

The 9x9A, 10x10A, 10x10B, and 10x10G assembly classes have part-length rods (PLRs). The staff finds it acceptable that there is no minimum PLR requirement for these classes due to the following: for the 9x9A, 10x10A, and 10x10B assembly classes the applicant found that the fuel assembly was more reactive when they removed the PLRs all together. For the 10x10G, removing the PLRs was not conservative and therefore all rods are assumed full length. The applicant also determined that there was not an intermediate length of the PLRs for the 10x10G that was more reactive than full-length.

7.1.1.1 Non-Fuel Hardware

Non-fuel hardware is not authorized for storage in the MPC-68M.

7.1.1.2 Fuel Condition

The MPC-68M is designed to hold up to 16 DFCs in peripheral locations described in Section 1.III.2.3 and Figure 1.III.2 of the FSAR. Eight of these locations may contain fuel debris. For the 8x8F, 9x9E, 9x9F and 10x10G assembly classes the applicant states in Section 6.III.4.1 of the FSAR that the damaged fuel evaluations were performed using a reduced enrichment of 4.0%. For the 10x10F class, the applicant reduced the enrichment to 4.6%. The staff verified that the enrichment limit in the TS for the damaged fuel configuration for these fuel assembly classes was appropriately reduced.

The applicant models damaged fuel and fuel debris as fuel debris. The staff finds this is acceptable due to the following: The applicant models a bare array (i.e. no cladding or structural material of any kind) of various sizes within the confines of the channel to find the most conservative pitch and provides the results of the calculations in Table 6.III.4.1 of the FSAR. The applicant grouped all of the intact assembly classes into three groups based on enrichment limits and were modeled using a representative assembly from the group. The applicant found that the 10x10 and 11x11 arrays provide the most conservative results. The applicant did not vary the size of the bare rod array for the 10x10F array and performed the calculation using the 11x11 array and showed that it is bounded by the 10x10A array (4.8% enrichment group) calculation. This approach has also been used with previously approved MPCs and fuel assembly classes.

7.1.2 Model Specification

The staff verified that the applicant included manufacturing tolerances important to criticality safety in the drawings in Chapter 1 of the FSAR. The staff verified that the applicant considered the manufacturing tolerances of the basket when constructing their criticality model. This is discussed in Section 6.III.3 of the applicant's FSAR. The applicant uses the minimum possible basket wall width. The applicant performed calculations for various fuel types, including damaged fuel, and demonstrates that the tolerances chosen produce the most reactive configuration for all of the fuel types. The staff finds that the applicant has appropriately considered manufacturing tolerances.

The staff finds it acceptable that the applicant assumes that there are no gaps in the basket panels due to the following. The basket is manufactured from individual panels. These panels are expected to be in direct contact with each other. However since there is a possibility that there could be small gaps between the panels, the applicant analyzed this effect. The applicant performed an analysis showing that the difference is very small. The staff finds it acceptable that the applicant assumes that there are no gaps in the basket panels.

Temperature (moderator density) is the same as that assumed for the other previously approved MPCs. The staff finds that the previous MPC-68 calculations performed with respect to finding the most reactive temperature are applicable to the MPC-68M.

7.1.2.1 Configuration

The staff finds that there are not additional accident or off normal conditions that would be introduced by adding the MPC-68M and its fuel contents. The applicant included the basket deflection that could result from accident conditions. The staff finds that the applicant's criticality model adequately represents the MPC-68M and its fuel contents during normal, off-normal, and accident conditions.

The staff verified that the dimensions of the basket are consistent with the drawings in the supplied revised Chapter 1 of the FSAR.

The applicant models each fuel type according to the specifications in Tables 2.1.4 and 2.III.3 of the FSAR, and these specifications are consistent with the fuel specifications required by the TS. The applicant does not include gadolinium or other burnable absorbers in their fuel analysis models. Including gadolinium and burnable absorbers in the analysis model would decrease the reactivity of the assembly. This provides a conservative assumption, and therefore the staff finds that fuel rods containing gadolinium (an element that is added to BWR fuel that acts as a neutron absorber) are acceptable for storage. This is because a fuel assembly with burnable absorbers will always be less reactive than the same fuel assembly with no burnable absorbers (such as the analyzed condition). Additionally, the applicant neglects all minor structural material and replaces it with unborated water in the analysis model. The staff finds that replacing the structural material with unborated water is conservative assumption since the fuel assemblies then are undermoderated. This is a characteristic of the fuel that means that adding more water will make them more reactive and since structural material does not perform any significant neutron absorbing capability, adding more water instead of structural material is a more conservative assumption.

The applicant performed all evaluations assuming eccentric fuel positioning. This is where the fuel is placed closest to the center of the basket in each basket cell. The applicant provided justification for this assumption in the supplied revised Section 6.III.4.2 of the FSAR. In addition the applicant performed a calculation demonstrating that this is the most reactive configuration in Table 6.III.4.6. The staff finds this assumption acceptable because it is consistent with previously staff reviewed and accepted practices.

The applicant assumes full external and internal flooding for the basket and DFCs. The staff finds this assumption acceptable due to the following. The applicant provided justification for this assumption in Section 6.III.4.3 of the FSAR. In addition the applicant performed calculations demonstrating that this is the most reactive configuration. The applicant shows in Table 6.III.4.7 of the FSAR that the reactivity is independent of the external flooding. In Table 6.III.4.8, the applicant shows that for the 10x10A array, that the fully flooded condition is more reactive than partial flooding conditions.

7.1.2.2 Material Properties

The staff verified that the composition and density provided for the Metamic-HT material is that used for the new basket. This is in obtained from supplied revised Table 6.III.3.4 of the FSAR. All other basket materials are defined in the main body of the FSAR in Table 6.3.4 and have been previously evaluated and accepted by the staff.

B-10 is a neutron absorber used for criticality control because it has a large cross section (i.e. probability of absorption) of neutrons of the same energy that also cause thermal fission such as in LWR reactor fuels, therefore, leaving fewer neutrons available to cause fission. Once the neutron is absorbed the B-10 atom becomes B-11 and is no longer as effective of an absorber. Over time the amount of B-10 may be degraded (i.e. become B-11) as it absorbs neutrons. The staff finds the applicant's conclusion that there is no degradation of the B-10 for 60 years acceptable due to the following. The applicant performed an analysis that assumed that the B-10 in the absorber material received a constant neutron flux equivalent to the design basis fuel as determined in FSAR, Section 5.2 and found that the amount of B-10 was roughly the same after 60 years. Even though there is some absorption of neutrons and some conversion

of B-10 to B-11, the applicant's analysis shows that there is enough B-10 present that the design basis flux over 60 years is not enough to reduce its capability as an absorber. Since the design basis fuel described in FSAR, Section 5.2 is not appreciably different from the new contents the B-10 degradation would be similar. The staff finds that the differences would not cause a safety significant increase in B-10 degradation and that this meets the requirements of 10 CFR 72.124(b) which requires an analysis demonstrating that there would not be significant degradation of neutron absorbing materials.

The applicant assumes a B-10 content of 90% for the Metamic-HT. The staff verified that the applicant has performed appropriate acceptance testing and assessment of the material to ensure its continued acceptability. The staff reviewed the applicant's testing and certification program provided in the supplied proposed revision to Section 9.III.1.5.2 of the FSAR. The staff finds this acceptable because these programs are similar to the applicant's certification program for Metamic-HT used in the Holtec HI-STAR 180 Transportation System that the staff has previously evaluated and found acceptable. Additionally, periodic NRC inspections have evaluated these specific applicant programs and have found them acceptable.

7.1.3 Criticality Analysis

7.1.3.1 Computer Programs

The applicant states the calculations that support the MPC-68M and its contents were performed using the same computer codes as that of the main part of Chapter 6 that were previously reviewed and approved by the NRC. The staff finds that the use of these codes would also be appropriate for the MPC-68M and its contents and finds the use of these codes acceptable because of the similarity in design and materials and that there have been modeled since its previous approval.

7.1.3.2 Multiplication Factor

The staff examined the results of the k-eff calculations. These are in Tables 6.III.1.1 through 6.III.1.3 of the FSAR. Tables 6.III.1.1 of the FSAR shows maximum k-eff values using the transfer cask configuration, Table 6.III.1.2 in the FSAR has representative k-eff values for the storage cask configuration, and Table 6.III.1.3 in the FSAR shows the maximum k-eff values when considering DFCs allowed. These values represent the highest k-eff that might occur during normal, off-normal and accident conditions. They are evaluated with the worst combination of manufacturing tolerances and include the calculational bias, uncertainties, and calculational statistics.

The limiting criticality configuration is for the transfer cask (HI-TRAC 100) configuration since when it is flooded it has the highest reactivity conditions. The calculation performed with DFCs is the most limiting for this design. This is for the configuration with "all assembly classes except the 8x8F, 9x9E/F, and 10x10G" at a maximum planar average enrichment of 4.8% and provides a k-eff of 0.9408. The limiting configuration for the MPC-68M without DFCs is the 10x10G fuel with 4.6% planar average enrichment and the maximum k-eff value is 0.9393. The applicant calculated a "representative value" of k-eff for the storage cask (overpack) and presented the results in Table 6.III.1.2 of the FSAR. The staff notes that this result is for the 10x10A array. Although, the applicant made no effort to determine the most reactive assembly class for this configuration, the analysis results provide a large margin to criticality. Therefore, any other assembly class would also show a large margin to criticality based on the similar characteristics and calculated k-eff values for other assembly classes identified in Tables

6.III.1.1, 6.III.1.2, and 6.III.1.3 of the FSAR. Light Water Reactor fuels require a water as the moderating medium to become critical, and since storage conditions are dry (i.e. without water) the staff finds that this is conservative for all assembly classes. Therefore the staff finds this result acceptable and finds that it provides adequate assurance that the MPC-68M within the HI-STORM 100 overpack is subcritical.

All of the calculated k-eff values meet the sub-criticality criterion of k-eff<0.95 and therefore the staff finds them acceptable.

7.1.3.3 Independent NRC Staff Calculations

The staff performed independent calculations to verify the k-eff of the HI-STORM 100 with the addition of the MPC-68M. The staff constructed its model using design information found in the SAR. The staff used the KENO6 code with the 238-group cross section library derived from ENDFB-VI data. The staff performed calculations of the MPC-68M using the configuration with DFCs. The staff chose the configuration that the applicant found to be the most reactive. This is the 10x10A array for the intact fuel and a bare array for the DFCs. Both intact and damaged fuel had an enrichment of 4.8%. The staff's model did not include gadolinium or any other type of burnable absorber because absorbers decrease reactivity and a goal of the analysis is to provide a conservative model for safety evaluation.

The staff's model was based on the HI-TRAC transfer cask since this cask would see flooding and provides a more reactive condition. The staff used the 125 ton model of the HI-TRAC. The staff performed a sensitivity study reducing the shielding to simulate conditions of the 100 ton HI-TRAC, and found that the 125 ton model produces slightly more conservative results.

The staff's model assumed full flooding of the inside of the fuel assemblies, and basket, including pure water inside the gap between the fuel rod and the cladding. The staff's model also assumes full density water external moderator.

The staff used several simplifying assumptions similar to that of the applicant. The staff assumed no structural materials within the basket besides the fuel tubes. The staff assumed that there was no assembly structural material and modeled only the active fuel length of 150 inches as axially centered within the basket. For simplicity, the staff modeled the active length of the damaged BWR fuel assemblies the same as that of the intact fuel, 150 inches. The applicant's FSAR in Section 6.4.4.22 says that the damaged fuel was modeled using a length of 155 inches. The staff modeled the channel box right outside the fuel rod array to minimize the amount of material, and the staff also neglected the material in the water rods. The staff assumed that there was no extra material for the DFCs.

FSAR Section 6.4.4.22 states that the minimum, maximum, and typical radii were modeled for all allowed fuel rod types to determine the most reactive. The staff performed a sensitivity study with its model and found that the most reactive fuel rod type was the nominal fuel type, but that the differences in the three calculations were within the uncertainty of the calculation.

The results of the staff's calculations show that the k-eff of the MPC-68M with 10x10A intact fuel with damaged fuel assemblies is 0.9336. This compares to the applicant's result of 0.9408. The staff's value does not include any code or modeling biases. Also the staff's modeling has some non-conservative assumptions such as non-eccentric positioning of the fuel assemblies. Therefore the staff finds the differences between its value and that of the applicant's are expected. The staff finds that this helps to demonstrate that the system is subcritical, and the

features important to criticality are sufficiently described. The applicant has addressed the most reactive conditions, the reported k-eff is conservative, and the applicant has appropriately modeled the cask geometry and materials.

7.1.3.4 Benchmark Comparisons

Since the applicant used the same computer code that was used for previously approved baskets and contents did not perform additional benchmarking calculations or re-calculate a bias. The staff finds this acceptable because the experiments used to calculate the bias include enrichment limits that bound that of the fuel that is authorized to be loaded in the MPC-68M.

7.2 PWR Fuel Assembly Class

The applicant proposes to add a new PWR assembly class to the HI-STORM 100 Cask System and provided the description in its proposed TS submitted with the amendment request. The 15x15I array class is only authorized to be stored in the MPC-32. The applicant calculated the k-eff of the 15x15I array class and the results show that it is bounded by the 15x15B array class. The applicant calculated the k-eff for intact fuel using the minimum soluble boron concentrations for each enrichment limit (1800ppm at 4.1% enrichment and 2500ppm at 5.0% enrichment). The staff viewed the design parameters of the 15x15I array class and compared them to the 15x15B array class and found that the two are similar; the main difference being the number of guide tubes (15x15I array class has 9 and the 15x15B array class has 21) and that the 15x15I array class has mostly solid Zircaloy “guide bars.” The staff finds that the two assembly classes have similar behavior with respect to reactivity when varying certain parameters (manufacturing tolerances, flooding, etc.), and that the 15x15B array class bounds that of the 15x15I array class. Any differences would not be substantial enough to cause any safety concerns, especially given the conservative assumptions used in the analysis. Damaged fuel evaluations for the MPC-32 are documented in Rev. 7 (8/2008) of the FSAR. The 15x15B array class was used to represent the 15x15A, C, and G array classes and therefore the staff finds that the 15x15B array class is also appropriate to represent the 15x15I array class for damaged fuel evaluations and that the authorized parameters for damaged fuel apply to the 15x15I array class. The MPC-32 is limited to 8 DFCs in peripheral locations as specified by the TS.

7.3 Evaluation Findings

Based on the above statements, the staff has the following evaluation findings with respect to the criticality analysis:

F7.1 SSCs important to criticality safety are described in sufficient detail in Chapters 1, 2 and 6 (as supplemented in amendment request 1014-8) of the FSAR to enable an evaluation of their effectiveness.

F7.2 The cask and its spent fuel transfer systems are designed to be subcritical under all credible conditions.

F7.3 The criticality design is based on favorable geometry, and fixed neutron poisons. An appraisal of the fixed neutron poisons has shown that they will remain effective for the term requested in amendment request 1014-8 and there is no credible way for the fixed neutron poisons to significantly degrade during the requested term in amendment request 1014-8; therefore, there is no need to provide a positive means to verify their continued efficacy as required by 10 CFR 72.124(b).

F7.4 The analysis and evaluation of the criticality design and performance have demonstrated that the cask will enable the storage of spent fuel for the term requested in amendment request 1014-8.

The staff finds that the criticality design features for amendment request 1014-8 are in compliance with 10 CFR Part 72, and that the applicable design and acceptance criteria have been satisfied. The evaluation of the criticality design provides reasonable assurance that the HI-STORM 100 Cask System will allow safe storage of spent fuel. These findings are reached on the basis of a review that considered the regulation itself, appropriate regulatory guides, applicable codes and standards, and accepted engineering practices as provided in NUREG 1536, Rev. 1.

8.0 MATERIALS

Amendment request 1014-8 incorporates one change requiring materials evaluation, and that is the use of previously evaluated Holtec Metamic HT for the fuel basket structure in the MPC-68M. Previous HI-STORM 100 Cask System MPC fuel baskets have used austenitic (300-series) stainless steel for the fuel basket structure with classic Metamic neutron poison plates attached to the basket. The remaining materials used in the fabrication of the HI-STORM 100 Cask System are unchanged and have been evaluated in previous Holtec storage system MPCs licensing applications.

8.1 Metamic HT Spent Fuel Basket

Metamic HT is a Holtec proprietary aluminum-based material intended for dual purpose use in the Holtec HI-STORM 100 Cask System MPC fuel basket. Metamic HT is designed to be both a neutron poison for criticality control and also a load-bearing structural material. Previously, Metamic HT was not used in a dual purpose MPC, although it has been used in the HI-STAR 180 System basket design.

The composition and properties of Metamic HT are unique. It is a powder metallurgy material composed of aluminum combined with aluminum oxide and boron carbide (Holtec uses the terminology “metal matrix composite” to generically describe Metamic HT). The aluminum oxide is a finely dispersed second-phase which provides enhanced room temperature and elevated temperature (creep) strength. The boron carbide is the neutron poison used for criticality control. Since Metamic HT evolved from a previously reviewed, non-structural neutron poison (classical Metamic), with closely similar neutronic properties, the neutronic properties of the new Metamic HT were already well characterized.

Discussion and evaluation of Metamic HT principally involves its structural characteristics. The neutronic properties of Metamic HT are essentially identical to previously reviewed classical Metamic and are evaluated in other sections of this SER.

Since Metamic HT is a new structural material, a comprehensive test program was necessary to fully determine its physical properties and characteristics. The testing program emulated that typically employed to qualify an American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code (Code) structural material. Holtec provided extensive data from their testing program for staff review. The data and analyses were documented in the

comprehensive Holtec proprietary report; “Metamic HT Qualification Sourcebooks” in lieu of

excerpting detailed technical discussion from the applicants' documents.

The testing program, employing a variety of standard American Society for Testing Materials (ASTM) test methods, included: yield strength, tensile strength, elongation, reduction in area, Young's Modulus, Charpy impact strength, thermal conductivity, coefficient of thermal expansion, and emissivity. Additionally, isotropy was evaluated, thermal aging effects were examined, welding procedures developed, weld properties determined, irradiation effects evaluated, thermally induced microstructural alteration assessed, and accelerated creep testing was performed. Neutron attenuation was verified (as for previously accepted "classic" Metamic) and corrosion testing was conducted in a simulated borated pool water environment.

For each material property and evaluation, up to 30 samples were tested. Samples were taken from multiple lots of material to determine lot to lot variability, which was found to be minimal.

Overall, the staff found that the applicant's test program was comprehensive in scope and supported the wide variety of property data needs for characterizing Metamic HT per the guidance in section 8.4.2.1 of NUREG-1536, Rev. 1.

8.1.1 Mechanical Properties

Using the guidance of the Code Section II, Appendix 5, Holtec determined mechanical properties (as listed previously) at room temperature and also at several temperatures ranging from -40°C (-40°F) to 500°C (752°F). The test data was analyzed using statistical methods and minimum, average, and mean values of the various properties were determined. Additionally, a design value for the various properties was established which is called the Minimum Guaranteed Value (MGV). The MGV is an arbitrary value (for any given property) below the lowest measured value from the test data. The MGV is then demonstrated (guaranteed) to be exceeded for every manufactured lot of Metamic HT through lot testing. Any lot which fails to meet one (or more) MGV values is discarded. Lot testing sample size is controlled by a statistical sampling approach. If any lot of material fails to meet any one of the various MGV's, then an enhanced sampling plan is invoked for subsequent production lots.

The staff found the mechanical properties of Metamic HT to generally equal or exceed those of conventional high-strength aluminum alloys, especially with respect to high-temperature performance. The lab data demonstrated that the short and long-term high-temperature strength was significantly superior to any conventional aluminum alloy. These superior high temperature properties extended well beyond the normal Code temperature limit of about 204°C (400°F) for aluminum alloys. Metamic HT was shown to have acceptable engineering properties even at 500°C (932°F).

Since the strength of aluminum alloys is also adversely affected by long times at elevated temperatures, the creep performance of Metamic HT was studied by Holtec. As discussed in other sections of this SER, the creep properties of Metamic HT were demonstrated to be significantly superior to conventional aluminum-based materials.

A common property of high strength aluminum alloys is reduced ductility. Metamic HT also displays this attribute. The design consequence of using a low ductility material requires the overall stresses to remain well below yield and local stresses and strains to remain near the elastic limit. The design consequence of using a low ductility material requires the overall stresses to remain well below yield and local stresses and strains to remain near the elastic limit. Therefore, the suitability of the material must be considered in the context of the structural

analysis. The material is therefore acceptable only if it meets the appropriate criteria under structural performance. This was evaluated in Section 3.0 of this SER.

8.1.2 Low Temperature Effects

Since Metamic HT is an aluminum-based material, it is expected that it would not be susceptible to ductile-brittle transformation or brittle fracture issues at low temperatures. To verify this assumption, coupons were tested at -40°C (-40°F). The samples showed no degradation in tensile, elongation, or impact properties, compared to room temperature properties. Thus, it was demonstrated that no low temperature ductility issues affected Metamic HT.

8.1.3 Thermal Aging Effects on Mechanical Properties

All metals undergo changes in their mechanical properties when exposed to elevated temperatures. Aluminum-based materials typically exhibit a decline in properties at temperatures above about 93°C (200°F). These property changes are generally reversible after short duration exposure. Long duration, elevated temperature exposure usually results in permanent decreases in mechanical properties such as yield and tensile strength. Since the storage canister is designed for a 40-year license interval, the long-term elevated temperature performance of Metamic HT is of primary interest. This is especially so since the fuel basket is designed to initially operate in the temperature regime above the typical Code limit of approximately 204°C (400°F) for aluminum alloys.

To determine the long-term, elevated temperature performance of Metamic HT, the applicant exposed approximately 30 samples to temperatures up to 350°C (662°F) in order to age the material in an accelerated fashion. The accelerated aging technique is intended to duplicate the metallurgical and physical property changes that would occur in the material under design conditions, but on a faster time scale. This accelerated testing is done by exposing the material to a higher temperature (for a shorter time period) and using a mathematical model to equate this accelerated aging process to a lower temperature, longer duration exposure as would occur in normal service. After the thermal aging, the samples were tested and compared to as-produced (un-aged) material samples to determine if any permanent changes had occurred to the material properties.

Holtec asserted that since Metamic HT is not a heat-treatable material (very unlike typical high strength aluminum alloys), it should not exhibit any aging effects. The Holtec test data supported this assertion. Aged samples exhibited some small changes in properties when compared to un-aged (room temperature) samples. The staff judged the changes to be minor. Tensile and yield strength values dropped about 2-4% whereas elongation slightly increased and reduction in area slightly decreased. Charpy impact strength was virtually unchanged. This is a unique response, as all other high-strength aluminum-based Code materials exhibit some amount of aging effect, often severe. The fact that the aging temperature used for the Metamic HT samples was roughly 121°C (250°F) higher than the temperature limit for other aluminum alloys makes the Metamic HT performance more notable.

The staff finds that the applicants' data supports that thermal aging is not a significant factor at the design temperature. Thus, no long term, thermally induced, degradation would be expected in service conditions and meets the requirements of 10 CFR 72.236(a).

8.1.4 Thermal Aging plus Irradiation Effects on Mechanical Properties

Samples of Metamic HT were exposed to elevated temperatures plus radiation fluence levels beyond those to be found in service. The test results have not revealed any unexpected or significant adverse synergistic effects. Some increase in tensile and yield strength are noted along with a corresponding slight reduction in ductility and toughness. This result is consistent with the performance of other metals in similar service. The staff finds the results of the thermal plus irradiation exposure show some small effect which is consistent with the experience for other metals. Thus no unanticipated response to service conditions (heat plus radiation) would result. The staff further notes that the fluence levels for a storage cask are a number of orders of magnitude lower than the accumulated dose of similar metals in reactor operation. This provides an additional margin of safety against adverse effects (such as brittle fracture) in Metamic HT components in service conditions.

8.1.5 Thermal Aging Effects on Microstructure

Microstructural alteration is an important indicator of material property changes. To support the mechanical property data provided, the applicant microscopically examined as-extruded material and compared its microstructure to thermally aged samples. The thermally aged samples showed little or no microstructural alteration when compared to the as-extruded (e.g., new material) samples. This demonstrates a thermal stability not observed in other aluminum-based materials, especially at the long-term creep test temperature of 400°C (752°F).

The staff examined the photomicrographs of the as-extruded (new) and thermally aged materials and found no significant changes to the microstructure. Based upon this examination, the staff finds that Metamic HT is highly stable at the design temperature. This provides reasonable assurance that microstructural changes under service conditions will not occur, and thus the material properties of Metamic HT will remain unaffected during service.

8.1.6 Short-Term Temperature Excursions

To address the material performance at short-term temperature excursions beyond 400°C (752°F), additional test samples were exposed at temperatures up to 500°C (932°F). Several material properties were determined at both 450°C (842°F) and 500°C (932°F). This data was obtained to support analyses of short-term (accident condition) temperature excursions on the basket performance.

The staff finds accident conditions specified and analyzed to be acceptable based on the discussion above.

8.1.7 Anisotropy

A significant issue with many high performance aluminum materials is anisotropy of mechanical properties. Potential anisotropy in Metamic HT was explored as part of the materials characterization project. A series of samples, including as-manufactured, artificially aged, and aged and irradiated, were tested in accordance with the compressive test technique of ASTM E9-89, “Standard Test Methods of Compression Testing of Metallic Materials at Room Temperature.” Test results showed no significant variation in properties with respect to extrusion orientation. The applicant concluded that anisotropy in Metamic HT is negligible.

Based on a review of the data provided, the staff finds that anisotropy in Metamic HT is small

and may be neglected for the intended application.

8.1.8 Weld Properties

The welding of high performance aluminum alloys presents many problems, as the heat from welding adversely alters the microstructure of the alloys. The strengthening microstructure is obtained by special heat-treatments, which high strength aluminum alloys must undergo to attain their enhanced strength. Consequently, high strength aluminum alloy structures (such as aircraft) are always riveted, bolted, or bonded versus welded.

Metamic HT utilizes a strengthening mechanism ("oxide dispersion strengthening") which is different from that employed in most aluminum-base materials. It does not depend on a heat treatment to achieve its strength. However, the Metamic HT strengthening mechanism is adversely affected by weld-zone, temperature-induced, microstructural alteration. Consequently, any Metamic HT welds are expected to possess less strength than that of the original base material.

The applicant undertook a weld development program to optimize the strength of Metamic HT welds. They were able to achieve weld strength of approximately 60 percent of the original base material strength. This reduced weld performance obviously affects weld design and restricts weld utilization. To compensate for this loss of strength, the Metamic HT basket welds are restricted to areas where stresses are low.

Welds with strengths which are lower than the base material strength are not permitted in Code-based designs. The Metamic HT basket is not designed in compliance with the Code. To compensate for the adverse weld properties, a strain-controlled design method was employed for the basket design. The adoption of a strain-controlled design and the placement of the welds in a low stress (hence low strain) location somewhat compensates for reduced weld strength.

The staff finds that although Metamic HT is deficient with respect to accepted weld strength requirements, (e.g., the Code) the strain-limited design approach and the placement of the welds in a low stress (and strain) region of the basket assures adequate performance in this application.

8.1.9 Creep Properties

Since the fuel basket operates above the Code temperature limit of about 400°C (752°F) for aluminum-base materials, creep strength/performance was identified early in the Metamic HT characterization program as a key issue. Consequently, a creep test program was established to determine the creep strength of Metamic HT at the design operating temperature.

Seven long-term creep samples were tested. Samples were tested at 300°C (572°F), 350°C (662°F), and 400°C (752°F). Stress levels used in the tests ranged from 200 to 1000 psi. The test conditions were designed to provide data which can be extrapolated to a creep life at design conditions. Although employing a limited number of creep test samples, the applicant attempted to bound design conditions sufficiently to ensure a large margin of safety.

If the normal operation design conditions for the Metamic HT basket are conservatively

assumed to be the maximum permissible fuel cladding temperature of 400°C (752°F), with a dead-weight load of less than 100 psi, then the creep test samples clearly show adequate creep-life margins for an assumed service life of 40 years. This is because the creep test samples were all tested at temperature and/or stress levels significantly above the design maximums.

The creep samples accumulated varying test times, between about 13,500 hours and 20,000 hours at the various temperature and load conditions. When the testing was discontinued, no specimens had failed. Cumulative creep strains were reported to vary between about 0.07% and 0.24%, depending upon test conditions (time, stress, and temperature).

The staff verified that the creep testing temperatures and stresses bound the operating parameters of the Metamic HT service conditions. Typically, a wide difference between the accelerated test conditions and the design conditions is avoided due to metallurgical considerations. Typical creep tests are generally conducted at a slightly elevated temperature, but at the same stress as the maximum design stress. The reason is to avoid metallurgical phenomena which could distort the result and lead to a significantly understated (though conservative) creep life prediction. While this would be safely conservative, it could be unnecessarily so by leading to much shortened creep life predictions.

The applicant used the accumulated creep data in conjunction with an established creep strain equation to produce a cumulative creep strain versus operating time relation. The applicant then compared the creep equation predictions with the data from the creep test samples. In every case, it was shown that the creep equation over predicted the actual measured creep strain by a comfortable margin. The applicant asserts that this demonstrates their equation is conservative due to over predicting the cumulative creep strain.

In conjunction with the creep equation, a limiting creep strain of 0.4% was adopted by the applicant as the maximum allowable creep strain in service. This limit is based on a foreign construction code creep strain limit for aluminum components. Employing the applicants creep strain equation along with the 0.4% allowable creep strain limit yields a service life well beyond the normal 5-year license period in the foreign license code.

Although this method is logical, the limiting creep strain is not adequately supported by the available data, because none of the creep test specimens were tested to failure. This means the creep strain at failure is undetermined. Thus, any creep strain limit based on failure strain is not supported. The applicant has demonstrated what the creep strain rate is for the several specimens. Absent failure data, there is no way to predict either creep strain at failure or maximum creep life, meaning time to failure. Adoption of the 0.4% creep strain limit could be conservative, based upon general creep failure strain knowledge, but it is speculative absent failure strain data.

Another issue with the creep testing program involves the choice of test parameters (temperatures and stresses). It is recognized that a limitation of accelerated creep testing is the susceptibility to overstate the creep strain at failure, at design conditions. Higher temperature, or, especially, higher stress, accelerated creep testing is susceptible to yielding creep failure strains which are slightly larger than those which would be achieved under service conditions. The reasons are related to the activation of additional creep mechanisms due to the necessarily more severe conditions employed during testing. In the applicant's case, the test temperatures were significantly higher than what would be typically employed for accelerated creep testing. Thus, the staff should view the achieved creep strains as possibly overstated. However, the

test outcome, when the time (not strain) element is considered, is conservative with respect to predicting the useful creep life under design conditions. Thus, although the staff concludes that the measured creep strains are overstated, the creep life (time) performance of the material when in operation is conservative.

With respect to predicted creep life, as measured in hours instead of accumulated creep strain, the applicant applied a mathematical formula called the Larson-Miller (L-M) equation. The L-M equation is frequently employed to relate the time at accelerated test conditions to time at operating conditions. Again, as for the creep strain equation, the L-M equation cannot predict total creep life unless the creep data includes samples tested to failure. However, given that the test samples never failed under simulated operating conditions, there is assurance that a predicted service life based upon the test sample times will be conservative.

The applicant provided an L-M calculation, based on the least conservative data from one of the lower temperature and stress test samples, which showed that a 40-year continuous operating life was achievable. The staff, through an independent L-M calculation, verified that a 40-year operating life is supported by the applicants creep test data.

Despite some shortcomings of the test program, the staff finds that the creep testing has bounded the design conditions by a wide margin. The severity of the creep tests will tend to understate the predicted service life (as measured in hours), which is very comfortably conservative at a predicted continuous service life of greater than 40 years. With respect to creep strain, the staff finds that due to the very elevated test conditions (temperature, stress), and the lack of failure strain data, the validity of the proposed 0.4% creep strain limit is unknown. However, the shortcoming of strain results and predictions becomes immaterial when the service life, expressed in hours, is employed instead.

In conclusion, the staff finds that the creep tests have conclusively demonstrated the ability of Metamic HT to adequately perform continuously at design conditions for at least 40 years, which is significantly beyond the 5-year license interval from the foreign code. An added level of conservatism exists when it is recognized that no storage canister is ever likely to be loaded with a design basis heat load for the entire license period.

8.1.10 Corrosion Resistance

Metamic HT was tested for compatibility with borated water, as would be typical for cask loading and unloading conditions. Aluminum alloys are very slightly corroded by borated water and Metamic HT performed similar to other aluminum-based materials in immersion tests. The applicant concluded that Metamic HT will not undergo any significant corrosion during cask loading or drying operations, or during dry cask storage.

The staff finds that Metamic HT is not susceptible to significant chemical or galvanic reactions and will perform adequately in accordance with 10 CFR 72.120(d).

8.1.11 Conclusions–Metamic HT

The applicant has provided data to quantify the performance and characteristics of Metamic HT in a manner somewhat emulating that of an ASTM/ASME material, but with several deviations from normal Code practice (the staff uses the Code as a benchmark). One deviation from the Code is the establishment of the MGV for various properties. The MGV values adopted by the applicant differ from the “stress allowable” used by the Code. The Code “stress allowable” is

directly used for stress-based design calculations. The applicant's MGV data is primarily used as the Metamic HT material production QA/QC acceptance criteria. Consequently, a direct comparison of "stress allowable" and MGV's is not valid.

A second major deviation in Metamic HT's performance from normal Code requirements is the reduced weld strength. The Code does not permit welds of a lower strength than the base material. The applicant recognized this characteristic and specified weld use only in low stress areas.

The third deviation from Code practice involves the creep test data. Code practice is to creep test materials to failure. The Metamic HT creep testing program never took a sample to failure. Thus, the data is incomplete in the sense that failure creep strain and ultimate time to failure is not determined. However, the applicant has adequately demonstrated through accelerated testing of samples that the Metamic HT will perform without failure for a license period of 40 years. No safety issue results from the incomplete creep test data.

Despite the above-discussed deviations from normal creep testing practice, the staff finds the applicants program of testing to be adequate to characterize the properties and performance of Metamic HT in its use in the MPC-68M basket. The staff further finds that Metamic HT creep properties are acceptable for use in the MPC-68M basket.

The staff finds that the applicant has met the requirements of 10 CFR 72.124 because materials used for criticality control are adequately designed and specified to perform their intended function.

8.2 Other Materials of Construction

The balance of the HI-STORM 100 MPC is fabricated from materials which have all been previously evaluated by the staff for their suitability. The bill of materials in FSAR drawings and Chapter 2 provide details of each material type and specification. Since all the materials have been previously reviewed and employed for 10 CFR Part 72 storage designs, only a brief synopsis of materials related findings are summarized below.

8.2.1 Confinement Boundary

The fuel canister confinement is fabricated from one of several ASME grades of austenitic stainless steel, referred to by the applicant as "Alloy X". Alloy X assumes the least favorable property characteristics from among the several materials grades specified. These properties are use for all design calculations. The purpose is to allow for free interchange of the several grades of stainless steel. This provides the applicant with procurement flexibility while complying with all required design properties. This method of allowing for material substitution has been previously reviewed by the staff and found to be acceptable. The use of austenitic stainless steel also means that the fuel canister is immune to brittle fracture issues.

8.2.2 Gamma and Neutron Shield

The radiation shield (concrete overpack) is composed primarily of un-reinforced concrete with a carbon steel liner plate on the inside. Since the overpack has no structural role, the lack of reinforcing steel is not a detriment. The lack of reinforcing steel is a deliberate exclusion in order to avoid the possibility of interior voids in the concrete which would degrade the shielding performance. This type of construction has been approved by the staff previously and found to

be entirely satisfactory in service at numerous installations. For the design service conditions, there are no conditions which will result in a degradation of the materials performance for the duration of the license period (up to 40-years). Experience has shown that the materials should easily achieve 40 years of service with no loss of performance or adverse degradation.

8.2.3 Weld Material

All weld filler materials utilized in the welding of the confinement boundary comply with the provisions of the appropriate Code Subsection. All non-Code welds (e.g., not important to safety) will be made using weld procedures that meet the Code Section IX, AWS D1.1, D.1.2 or equivalent. All non-destructive examinations will comply with Section V of the Code, with acceptance criteria as specified by the code of construction for the specific component.

8.2.4 Chemical, Galvanic, or Other Reactions

The canister and contents (fuel basket, fuel payload) are all fabricated from corrosion resisting materials which have previously been evaluated and employed in storage canister service. During wet loading, there may be a slight chemical or galvanic reaction from the Metamic HT basket material. This reaction proceeds at such a slow rate that it has no measurable effect on the basket structure or performance. After loading, the cask is dried and helium backfilled to eliminate any credible corrosion from moisture and oxidizing gasses. Therefore, chemical, galvanic or other reactions involving the cask materials are minimal.

The staff finds that the applicant has met the requirements of 10 CFR 72.236(h) because the HI-STORM 100 Cask System MPCs employ materials that are compatible with wet and dry SNF loading and unloading operations and facilities. The staff finds there is no significant adverse chemical or galvanic reaction that would impact the safety or performance of the canister or payload as components are composed of materials with a service proven history of use.

8.3 Conclusion-Materials of Construction

The revised FSAR Chapter 8 adequately describes the materials used for SSCs important to safety and the suitability of those materials for their intended functions is sufficient detail to evaluate their effectiveness.

The applicant has met the requirements of 10 CFR 72.122(a). The material properties of SSCs important to safety conform to quality standards commensurate with their safety function.

The applicant has met the requirements of 10 CFR 72.122(h)(1) and 72.236(h). The design of the DSS and the selection of materials adequately protects the SNF cladding against degradation that might otherwise lead to damaged fuel.

The applicant has met the requirements of 10 CFR 72.236(h) and 72.236(m). The material properties of SSCs important to safety will be maintained during normal, off-normal, and accident conditions of operation so the SNF can be readily retrieved without posing operational safety problems.

The applicant has met the requirements of 10 CFR 72.236(g). The material properties of SSCs important to safety will be maintained during all conditions of operation so the SNF can be safely stored for the minimum required years and maintenance can be conducted as required.

8.4 Evaluation Findings

F8.1 The staff concludes the material properties of the structures, systems, and components of the HI-STORM 100 Cask System remain in compliance with 10 CFR Part 72, and that the applicable design and acceptance criteria have been satisfied. The evaluation of the material properties provides reasonable assurance the cask will allow safe storage of SNF for a licensed life of at least 40 years. This finding is reached on the basis of a review that considered the regulation itself, appropriate regulatory guides, applicable codes and standards, and accepted engineering practices.

9.0 OPERATING PROCEDURES EVALUATION

There were no requested operating procedures changes requiring evaluation.

10.0 ACCEPTANCE TESTS AND MAINTANANCE PROGRAM EVALUATION

A change to the confinement boundary leakage testing was previously addressed in section 5 of this SER.

11.0 RADIATION PROTECTION EVALUATION

There were no requested radiation protection changes requiring evaluation.

12.0 ACCIDENT ANALYSIS EVALUATION

There were no requested accident analysis procedures changes requiring evaluation. Relevant accident analyses for the MPC-68M are addressed in previous sections of this SER. Other accidents are bounded by previous analyses for other systems. Structures, systems, and components of the MPC-68M are adequate to prevent accidents, and to mitigate the consequences of accidents and natural phenomenon events that do occur.

13.0 TECHNICAL SPECIFICATIONS

13.1 Review Objective

The objectives of this review were to ensure that the changes to the operating controls and limits or the TS for the HI-STORM 100 Cask System continue to meet the requirements of 10 CFR Part 72. The evaluation is based on information provided by the applicant in this amendment request, a review of the FSAR, as well as consideration of accepted practices. Specifically, the proposed changes were reviewed to ensure that they acceptably supported the equipment changes requested by the applicant. The technical and safety aspects of these changes were evaluated by the staff in previous sections of this SER and were found to be acceptable. The applicant proposed technical and editorial TS changes. Additionally, the staff proposed several editorial TS changes.

Equipment changes and additions that required TS change evaluations were as follows:

1) Addition of a new MPC – 68M to the approved models presently included in CoC No. 1014 with two new BWR fuel assembly/array classes. Two new BWR fuel assembly array/classes are added for exclusive use in the MPC-68M. The MPC-68M provides the following improvements:

a. Fuel assemblies with planer average initial enrichments up to 4.8 weight percent ^{235}U can be loaded without burnup or gadolinium credit.
b. A supplemental cooling system is not required during short-term loading operations to keep fuel cladding temperatures below regulatory limits.

2) Addition of a PWR fuel assembly array/class to CoC 1014 for loading into the MPC-32.

The applicant requested a TS change to Appendix A, Table 3-1, "MPC Cavity Drying Limits" to correct errors that were issued in CoC No. 1014, Amendment No. 5. CoC No. 1014, Amendment No. 5, approved the revised TS Table 3-1, but the approved changes were not incorporated in the approved TS changed pages due to an administrative oversight. This created inconsistencies between Table 3-1 and TS Limiting Condition for Operation 3.1.1. The staff reviewed the request and determined inclusion of the previously approved, but omitted table is acceptable.

The staff identified the following additional CoC and TS changes:

1) CoC 1014, Condition 5. Revise Condition 5 to add "if applicable" after the reference to Section 3.5 of Appendix B, "Cask Transfer Facility (CTF);" to clarify that the CTF is an optional facility.

2) Appendix A, TS, Definitions. Revise the CTF definition to clarify that the CTF is an optional facility that could be used in lieu of 10 CFR Part 50 controlled structures for cask transfer evolutions.

These changes are consistent with the staff's original SER evaluation of Section 1.1. The language states that the "MPC transfer between the transfer cask and overpack can be performed inside or outside a 10 CFR Part 50 controlled structure (e.g., a reactor building)." The CTF is a Part 72 structure.

The staff provided these proposed revisions to Holtec for review and comment. In a December 15, 2011, letter to the NRC, Holtec agreed to these NRC initiated revisions. Therefore, the staff finds the changes acceptable.

13.2 Findings

F13.1 The staff finds that the conditions for use for the HI-STORM 100 Cask System continue to identify necessary TS to satisfy 10 CFR Part 72 and that the applicable acceptance criteria of 10 CFR 72.234(a) and 10 CFR 72.236 have been satisfied. The proposed TS changes provide reasonable assurance that the HI-STORM 100 Cask System will continue to allow safe storage of SNF. This finding is based on the regulation itself, appropriate regulatory guides, applicable codes and standards, and accepted practices.

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August 21, 2011