UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

SAFETY EVALUATION REPORT

DOCKET NO. 72-1014
HOLTEC INTERNATIONAL
HI-STORM 100 CASK SYSTEM
CERTIFICATE OF COMPLIANCE NO. 1014
AMENDMENT NO. 9, REVISION NO.1

SUMMARY

By letter dated July 1, 2014, Holtec International (Holtec) submitted a revision request to the U.S. Nuclear Regulatory Commission (NRC) for the HI-STORM 100 Certificate of Compliance (CoC) No. 1014, Amendment No. 9. The proposed changes include the following:

(1) Change Burnup/Cooling Time limits for thimble plug devices (TPDs),
(2) Change Metamic-HT testing requirements,
(3) Change Metamic-HT minimum guaranteed values (MGVs),
(4) Update fuel definitions to allow boiling water reactor (BWR) fuel affected by certain corrosion mechanisms within specific guidelines to be classified as undamaged fuel.

The staff has provided an additional CoC condition that allows a general user up to 180 days to implement any changes and to update their 10 CFR 72.212 evaluations required by implementation of the revision. Holtec agreed with this change in correspondence dated September 24, 2015 (ADAMS Accession No. ML15267A234).

This revised CoC, when codified through rulemaking, will be denoted as Amendment No. 9, Revision No. 1, to CoC No. 1014. As a revision, the CoC will supersede the previous version of the CoC and TS, effective March 11, 2014, in its entirety. The applicant has requested a revision in lieu of a new amendment utilizing the following justifications.

- Equipment for CoC No. 1014, Amendment No. 9, has been placed in service by several general licensees, all of whom were made aware of the revision request and supported it;
- No new canisters are being requested to be added to CoC No. 1014, Amendment No. 9;
- No new systems, components, or structures (SSCs) are requested to be added to CoC No. 1014, Amendment No. 9;
- The requested changes have minor field and administrative implementation impact on general licensees;
- The requested changes are applicable only to CoC No. 1014, Amendment No. 9.

This safety evaluation report (SER) documents the review and evaluation of the proposed revision. The NRC staff (staff) followed the guidance of NUREG-1536, Revision 1, “Standard Review Plan for Dry Cask Storage Systems,” Interim Staff Guidance (ISG) -11 “Cladding Considerations for the Transportation and Storage of Spent Fuel,” ISG-21 “Use of Computational Modeling Software” in performing its regulatory evaluation, and

ISG-23 “Application of ASTM Standard Practice C1671-07 when performing technical reviews of spent fuel storage and transportation packaging licensing actions.”

The staff's evaluation is based on a review of Holtec’s application and whether it meets the applicable requirements of 10 CFR Part 72 for independent storage of spent fuel. The staff’s evaluation focused only on modifications requested in the revision 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. 9.

1.0 GENERAL DESCRIPTION

The objective of this chapter is to review the requested design changes made to CoC No. 1014, Amendment No. 9 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. The specific changes are described and evaluated in later sections of this SER.

1.1 Findings

F1.1 The staff concludes that the information presented in the proposed FSAR pages satisfies the requirements for the general description under 10 CFR Part 72. This finding is reached on the basis of a review that considered the regulation itself, Regulatory Guide 3.61, and accepted practices. The staff concludes that the applicant’s information is sufficiently detailed to allow reviewers to familiarize themselves with the pertinent features of the system and the changes requested

2.0 PRINCIPAL DESIGN CRITERIA EVALUATION

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

3.0 STRUCTURAL EVALUATION

The staff has reviewed the requested revision to CoC No. 1014, Amendment No. 9, which requests two changes relevant to structural performance:

(1) Changes to testing requirements for Metamic HT and
(2) Changes to Minimum Guaranteed Values (MGVs) for Metamic HT

The first proposed change, as described by the applicant in their revision request, does not have any direct bearing on structural performance. Further review of the change request letter and the FSAR change pages (Section 1.III.2.4 Qualification of Metamic-HT, (e) Welding of Metamic – HT) shows that the applicant intends to employ a new qualified welding process called Friction Stir Welding (FSW), for external basket joints. Allowing the use of friction stir welding of the Metamic HT basket does not change the safety basis as evaluated by the staff in

HI-STORM 100, Amendment No. 9, with respect to basket structural performance. Since the basket corners utilize the same welded joint configuration specified in amendment No. 9 and prior amendments, the primary consideration is that of weld process and qualification, rather than structural performance of the weld itself.

Based upon a review of the application, the staff determined that the methods employed to structurally qualify the weld joint were sufficiently robust to demonstrate comparable structural performance to the welding method described in previous amendments. The staff's conclusion is based upon the fact that the loading conditions and the fully supported boundary conditions (via shims) of the peripheral basket panels result in joint stresses significantly below their full capacity. This results in significant margin which should account for any differences in welding procedures, should they arise in the future. These conclusions only apply to the basket corner welds and shim arrangement defined by this revision.

The second part of the change request addresses changes to the MGVs for Metamic HT. The MGVs for Metamic HT are used in calculations to demonstrate that the structural components will satisfy engineering requirements such as stress limits or deflection limits. By providing MGVs, all calculations performed with those values will represent a bounding calculation for a given engineering requirement. The applicant referenced its engineering change order (ECO) process supported by a 10 CFR 72.48 evaluation to make changes to the MGVs, but elected to additionally submit the proposed changes in those values to the NRC for review and approval through the revision process. A review of the material properties submitted by the applicant indicates an average reduction in MGVs of approximately 10%, for material yield stress, ultimate strength, and Young's modulus. A reduction of 20% of the MGV was reported by the applicant for the reduction in area criteria measured during a tensile test. The applicant applied these changes to structural calculations (stress strain curve development for finite element analysis) and determined that positive margin remains for basket performance criteria. The positive margins include the areas of peak stress criteria, maximum deflection criteria, and crack propagation criteria. The staff reviewed these results and, because positive margin remains for basket performance criteria even with the reduced MGVs, finds this acceptable.

3.1 Evaluation Findings

Based on evaluation of the supporting documentation and calculation for CoC No. 1014, Amendment No. 9, Revision No. 1, the staff finds that the revision acceptably meets the review criteria identified in NUREG-1536, REV. 1. Specifically, the staff finds:

F3.1 The structural properties of the CoC No. 1014, Amendment No. 9, SSCs remain in compliance with 10 CFR Part 72, and the applicable design and acceptance criteria have been satisfied. The evaluation of the structural properties provides reasonable assurance that the CoC No. 1014, Amendment No. 9, Revision No. 1, will allow safe storage of spent nuclear fuel (SNF). 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.

4.0 THERMAL EVALUATION

4.1 Summary

The objective of the thermal evaluation is to review and evaluate the thermal impact of the following applicable proposed change in the revision to CoC No. 1014, Amendment No. 9.

1) Reduced cooling time limit for Thermal Plug Devices.

The applicant proposed to reduce the cooling time limits for TPDs. The applicant stated that the TPDs are not considered in any of the thermal analyses of CoC No. 1014, Amendment No. 9, but noted that, in order to be in compliance with CoC No. 1014, Amendment No. 9, general licenses must perform an evaluation per 10 CFR 72.212 to ensure the maximum fuel storage location decay heat limits evaluated in the FSAR and captured in TSs are met. The staff reviewed the FSAR and finds that there is no impact on the internal pressure and fuel cladding temperatures of the MPC from the proposed change. However, as stated in the application, general licensees are required to evaluate and assure the cell heat loads per canister remain below the applicable limits as listed in FSAR and TS prior to loading.

2) Removing Metamic-HT thermal conductivity testing requirement

The applicant proposed to remove the thermal conductivity testing of the Metamic-HT during fabrication because there is little variability in thermal conductivity of Metamic-HT when fabricated according to the manufacturing manual.

The staff evaluated the applicant's proposal and agrees with the applicant's conclusion that the thermal conductivity of Metamic-HT is stable to 180 w/m-°K for temperature from 200°C to 500°C (normal operating temperatures) (Reference Metamic HT Qualification Sourcebook, ADAMS Accession No. ML090340598). The staff, therefore, finds removal of the Metamic-HT thermal conductivity testing to have no impact on any of the previously approved staff evaluations and is therefore acceptable.

4.2 Evaluation Findings

F4.1 The staff evaluated the application and concludes that the proposed changes in the cooling time limit of the TPDs and in the removal of Metamic-HT thermal conductivity testing requirement are in compliance with 10 CFR Part 72 with no negative impact to the safe storage of spent fuel for a licensed life. 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.

5.0 CONFINEMENT EVALUATION

There were no requested changes requiring evaluating confinement integrity related to important to safety SSCs to ensure compliance with the relevant general criteria established in 10 CFR Part 72.

6.0 SHIELDING EVALUATION

6.1 Summary

The objective of this review is to verify that Revision No. 1 to CoC No. 1014, Amendment No. 9, meets the external radiation requirements of 10 CFR Part 72 under normal conditions of transport (NCT) and hypothetical accident conditions. The staff reviewed the proposed changes to the burnup and cooling time limits for TPDs to ensure that CoC No. 1014, Amendment No. 9, as revised, continues to meet the shielding and radiation protection regulations in 10 CFR Part 72. TPDs are a form of non-fuel hardware used in pressurized water reactor fuel assemblies. During power operations, TPDs are inserted into the assembly guide tubes for uncontrolled assemblies. Currently, cooling times for TPDs exposed to typical burnups are very long, preventing many TPDs from being stored in MPCs. The applicant proposes to substantially reduce the required cooling times, so that they can store a larger population of TPDs and provide greater flexibility to general licensee users.

As described in more detail below, the staff evaluated the requested changes in conjunction with the findings from previous staff analysis to determine that CoC No. 1014, Amendment No. 9, continues to provide adequate shielding protection from the radioactive contents within. When performing this evaluation, the staff reviewed the methods and calculations employed by Holtec to determine the expected gamma dose rates at locations near the cask surface and at specific distances away from the cask.

6.2 Shielding Design Description

6.2.1 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.

6.3 Applicant’s Radiation Source Model

As explained by the applicant, TPDs are made of stainless steel and contain a small amount of Inconel. The applicant assumed the Cobalt-59 impurity level to be 4.7 gm/kg. These devices extend down into the plenum region of the fuel assembly but typically do not extend into the active fuel region. Also, TPDs are not used in all assemblies in a reactor core but are reused from cycle to cycle. The maximum activity for each TPDs is 141 curies. The applicant determined the TPD’s Co-60 activity as a function of burnup for a cooling time of five years. The difference between TPDs and burnable poison rod assemblies (BPRAs) are related to the amount of Co-60 in the four region of the fuel assembly. For TPDs, Co-60 is found in the upper end fitting, gas plenum spacer and gas plenum spring. On the other hand, for BPRAs, Co-60 is found in all regions as TPDs and also in-core. The applicant calculated the activity for each region by multiplying 141 curies by the Cobalt mass fractions. For BPRAs, the maximum activity was assumed to be 895 curies.

The applicant performed calculations at various distances from the various configurations of arrays of HI-STORM 100S, Version B, overpack containing the MPC-24 and concrete density of 2.24 gm/cc (140 lb/cuft) in the body and the lid. The HI-STORM 100S, Version B, overpack was implemented acceptably in the HI-STORM cask system FSAR, revision 2, utilizing 10 CFR 72.48 provisions. This MPC-24 with a HI-STORM 100S, Version B, overpack design is used for analysis in CoC No. 1014, Amendment No. 9. The normal analyzed configuration is with the lid on.

The applicant also calculated the dose rates from the HI-STORM 100S, Version B, overpack with the MPC-68 and MPC-32 using allowable burnup and cooling times from the proposed Revision No. 1 to CoC No. 1014, Amendment No. 9. Dose rates were calculated for both uniform and regionalized loading and compared to the TS contained in Amendment No. 9 to CoC No. 1014. All dose rates were less than the TS allowable values. The staff finds the applicant's evaluation to be in accordance with NUREG 1536, Rev. 1 guidelines, and, therefore, finds it acceptable.

6.4 Applicant's Shielding Model

The applicant described the HI-STORM 100S, Version B, overpack model in the revision application to Amendment No. 9 to CoC No. 1014.

6.5 Staff Shielding Evaluation

The applicant supplied a series of qualitative analyses to show the bounding shielding evaluation that was applied in approving Amendment No. 9 to CoC No. 1014 is still applicable to the HI-STORM 100S, Version B, overpack for the revision for Amendment No. 9 to CoC No. 1014. The applicant references the previously reviewed shielding analysis which states that the maximum Co-60 activity of TPDs is 141 curies. Fuel assemblies with BPRAs containing 895 curies of Co-60 were used to calculate the dose rates at various configurations and demonstrate that the results were still bounding. The staff reviewed the applicant's new shielding calculations that were based on five years cooling time and 63,000 MWD/MTU. The applicant's results determined the dose from the BPRAs bounds the dose from the TPDs. The applicant used SAS2H and ORIGEN-S to calculate a radiation source term for the TPDs and BPRAs. In the ORIGEN-S calculations the cobalt-59 impurity level was conservatively assumed to be 0.8 gm/kg for stainless steel and 4.7 gm/kg for inconel. Table J-2 of the application identifies a Co-60 source for TPDs with a five year cooling time and burnups less than or equal to 63,000 MWD/MTU is 141 curies. This is lower than the maximum Co-60 activity of the TPDs, which is 240 curies. The applicant selected the amount of 141 curies Co-60 for each TPD as the design basis Co-60 activity for each TPD. By selecting 141 curies of Co-60 for each TPD as the design basis activity, any TPD can be stored in a HI-STORM MPC so long as the TPD has a cooling time of five years or greater and a burnup of less or equal to 63,000 MWD/MTU, as required by the TSs. The applicant's analyses are consistent with the review guidelines of NUREG 1536, Rev. 1 and, therefore, the staff finds applicant's shielding evaluation acceptable.

6.6 Evaluation Findings

F6.1 Based on the information provided by the applicant the staff has determined that sections provided in Holtec Report No. 2033974, "HI-STORM 100S Version B Shielding Analysis," describe the shielding assumptions important to safety in sufficient detail to allow evaluation of their effectiveness.

F6.2 The staff concludes that the proposed changes to CoC No. 1014, Amendment No. 9, are in compliance with 10 CFR Part 72. The evaluation of the shielding system design provides reasonable assurance that the HI-STORM 100S Version B overpack will allow safe storage of spent fuel with TPDs with lower cooling time. The staff reached this conclusion considering the regulation itself, appropriate regulatory guides, applicable codes and standards, and accepted engineering practices.

7.0 CRITICALITY EVALUATION

7.1 Summary

The applicant requested that low enriched channeled BWR fuel that may have indeterminable minor damage to the cladding be considered as undamaged for storage in the MPC-68M without having to put these fuels into Damaged Fuel Containers (DFCs). As described in the FSAR, some older fuel assemblies (e.g., Copper Induced Localized Corrosion [CILC] fuel) may have corrosion-induced damage to the cladding, but the cladding has not been grossly breached. Currently, these fuels, in order to be considered undamaged, must have an undamaged channel that is attached normally to the fuel assembly and the maximum planar average enrichment of the assembly must be less than 3.3 wt% ^{235}U.

7.2 Criticality Design Criteria and Features

The applicant has made no changes to the design criteria and features of the HI-STORM 100 cask system.

7.3 Fuel Specification

There are no new fuels requested as part of this amendment. The applicant has utilized fuel designs that have been previously approved by the NRC with new requirements that will allow channeled low enriched BWR fuel of all classes to be considered as undamaged provided they meet three specific criteria. These criteria are:

1. The channel must be present and attached to the fuel assembly in the standard fashion:
2. The channel is essentially undamaged: and
3. The maximum planar average enrichment of the assembly must be less than or equal to 3.3 wt% ^{235}U.

7.4 Applicant Model Specifications

The previous amendments to CoC No. 1014 evaluated the storage of both undamaged and damaged fuels. Since the applicant has requested that channeled low enriched BWR fuel be considered as undamaged, staff requested examples of the applicant's MCNP computer modeling code in order to evaluate the methodology used for the inclusion of low enriched channeled BWR fuel with an indeterminate cladding condition. The applicant provided MCNP codes used for the 8x8, 10x10, and 12x12 cases, and used an approach similar to one that had been previously found acceptable to staff that has been used to evaluate the criticality safety of damaged fuel and fuel debris. The low enriched channeled BWR fuel was modeled under optimum moderation conditions. According to the applicant, since the channel ensures that the questionable fuel rods are confined within the fuel assembly in their potentially damaged but unbreached state, coupled with the low level of maximum enrichment of the rods, the overall reactivity of the fuel remains below regulatory limits.

7.5 Applicant Computer Programs

The applicant used the MCNP three-dimensional Monte Carlo code with continuous energy cross-sections. The MCNP code was developed by the Los Alamos National Laboratory for performing criticality analyses and is considered to be appropriate for this particular design and these fuel types.

7.6 Criticality Evaluation Summary

The applicant used three-dimensional calculation models in their criticality analyses based on engineering drawings in the FSAR and previously approved models submitted as part of the supplemental calculations. The staff evaluated the applicant's calculations to ensure that sound engineering practices and methodologies were used to evaluate the criticality safety of channeled low enriched BWR fuel of all classes to be considered as undamaged.

7.7 Staff Evaluation Findings

The staff reviewed the information provided as part of this revision and determined that it is in compliance with the requirements of 10 FR 72.24, 10 CFR 72.40, 10 CFR 72.124, and 10 CFR 72.236(c). The staff also determined that the criticality results associated with the evaluation of channeled low enriched BWR fuel of all classes to be considered as undamaged, as described in this application remain less than regulatory limits. The applicant used previously approved fuel designs that incorporated conservative assumptions and evaluated the fuel over a range of bounding credible scenarios. As a result, based on this assessment, staff finds that the HI-STORM 100 Cask System, as revised was adequately modeled with sound engineering practices and appropriate methodologies, and that the inclusion of low enriched channeled BWR fuel classified as undamaged as described above in the MPC-68M will continue to meet the criticality safety requirements of 10 CFR Part 72. 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. Specifically, the nuclear criticality safety evaluation demonstrates that the HI-STORM 100 Cask System will continue to meet the relevant regulatory requirements and the staff finds the following:

F7.1 Structures, systems, and components important to criticality safety are described in sufficient detail in 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 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 the revision application.

8.0 MATERIALS EVALUATION

The staff reviewed the information provided by the applicant and evaluated the four changes requested in this revision as follows:

(1) Change Burnup/Cooling Time limits for TPDs,
(2) Change Metamic-HT testing requirements,
(3) Change Metamic-HT MGVs, and
(4) Update fuel definitions to allow BWR fuel affected by certain corrosion mechanisms within specific guidelines to be classified as undamaged fuel.

The remaining materials used in the fabrication of the HI-STORM 100 Cask System are unchanged and have been used in all previous Holtec storage system MPC's.

The staff notes that Metamic-HT is a Holtec proprietary aluminum-based material intended for dual purpose use in the 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. Metamic-HT is a powder metallurgy material composed of aluminum combined with aluminum oxide and boron carbide. The aluminum oxide is a finely dispersed second-phase particle which provides enhanced room temperature and elevated temperature (creep) strength. The boron carbide is the neutron poison used for criticality control. The Metamic-HT evolved from a previously reviewed, non-structural neutron poison (Metamic), with closely similar neutronic properties. The neutronic properties of the Metamic-HT have been previously acceptably evaluated by the staff in CoC No. 1014, Amendment No. 8.

8.1 Changes to Burn-up and Cooling Time Limits for TPD:

The applicant requests modification of TS Appendix B, Table 2.1-8 by changing the Burn-up/Cooling time limits for TPDs. The applicant stated that currently, cooling times for TPDs exposed to typical burn-up are very long, preventing many TPDs from being stored in MPCs. The proposed change substantially reduces the required cooling times, so that a larger population of TPDs can be stored, providing greater flexibility to the users.

The staff notes that TPDs are non-fuel hardware irradiated by neutron flux during reactor operation and have become radioactive as a result. The staff finds storing of TPDs to be acceptable because there is no risk of non-fuel rupturing and releasing fission products, fission product gases or any material considered detrimental to the internals of the cask and/or preventing retrievability.

8.2 Changes to Metamic-HT Testing Requirements:

The applicant stated that changes to the Metamic-HT testing requirements were made to remove testing using a 1-inch beam; remove fabrication testing of Charpy V-notch and lateral expansion; revise fabrication testing requirements of weld test procedures and remove the thermal conductivity testing requirement; change failed MGV re-testing criteria by requiring only the failed property to be re-tested (not all MGVs), and add the ability to conduct 100% testing of an MGV property within a lot if it fails re-testing. According to the applicant, the changes to the Metamic-HT testing requirements are to improve testing processes or ease undue burden. The applicant asserts that some testing requirements were overly conservative and created a lengthy testing process, while others did not impact the safety analysis. The applicant proposes to relocate the reference in the CoC Appendix B to the FSAR with an explicit list of testing requirements.

The requirement for the use of a 1-inch beam is based on ISG-23 which concludes that a beam between 1 cm and 2.54 cm is acceptable for qualification and acceptance testing of neutron absorbing materials. However, the Charpy V-notch is a measure of a given materials notch toughness and acts as an impact loading tool to study temperature-dependent ductile-brittle transition. The ductile-to-brittle transition is characterized by a sudden and dramatic drop in the energy absorbed by a metal subjected to impact loading. As temperature decreases, a metal's ability to absorb energy of impact decreases; therefore, its ductility decreases. At some temperature the ductility may suddenly decrease to almost zero. However, this transition is essentially unidentified in metals possessing a face-centered cubic (fcc) crystal structure. Because Metamic-HT is an aluminum (fcc) based metal matrix composite, the staff therefore concludes, the Charpy V-notch is not a necessary test for the Metamic-HT.

8.3 Changes to MGV for Metamic-HT:

The applicant requested changes to certain MGVs of the thermal-physical properties of Metamic-HT used for the MPC-68M basket. Some of these revised values are used in the applicant’s structural analyses of the baskets. The applicant also reorganized fragmented text matter on Metamic-HT in the FSAR to clarify the role of each thermo-physical property. MGV requirements of tensile strength, yield strength, area reduction and elongation were revised lower by approximately 10%, necessitated to address limited global supply of Aluminum powder and the need to increase the range for surface area while maintaining overall requirements. The property values are guaranteed to be met by the imposition of a sampling test plan based on the standards for critical service parts. This is the same change as that made by Holtec to the HI-STORM 100 Flood/Wind (FW) Multipurpose MPC Storage System, CoC - No. 1032 using an acceptable 10 CFR 72.48 evaluation. It has been included here as a change submitted for NRC review for completeness of the Metamic-HT characterization.

Using the guidance of the American Society of Mechanical Engineers (ASME) Section II, Mandatory Appendix 5, “Guideline on the Approval of New Materials Under the ASME Boiler and Pressure Vessel Code,” the applicant determined mechanical properties at ambient/room temperature and various other higher/lower temperatures. The test data was analyzed using statistical methods and minimum, average, and mean values of the various properties were determined. In addition, the design value MGV for the various properties was established. The MGV is an arbitrary value for any given property below the lowest measured value from the test data. The MGV is then demonstrated or guaranteed to be exceeded for every manufactured lot of Metamic HT through lot testing.

The staff finds the mechanical properties of Metamic HT to equal or exceed those of conventional high-strength aluminum alloys, especially with respect to high-temperature performance. The mechanical property testing program adequately characterized the performance of Metamic HT for its intended use as a structural material in terms of identifying physical/mechanical properties, characteristics and testing. The properties have been quantified by a testing program that follows the qualification standard of an ASTM material. The reduced primary mechanical properties tensile strength, yield strength, young’s modulus and percent area reduction are properties that directly bear upon the structural strength of the basket. Therefore, these reduced MGVs required structural evaluation that was provided in SER Section 3.

8.4 Update to Fuel Definitions/Classifications:

The applicant stated that adding the definition of undamaged fuel assemblies, grossly breached spent fuel rod, and repaired/reconstituted fuel assembly provides further clarity to the user and consistency with ISG-1, Revision 2, guidance on classifying fuel. In addition, the applicant contends these definitions will serve some BWR users who have older, low enriched, channeled BWR fuel with potential cladding defects that they wish to load without placement in a damaged fuel container.

8.4.1 Undamaged Fuel Assembly

The applicant proposed adding the new definition “UNDAMAGED FUEL ASSEMBLY.” The applicant proposed that this is: a) a fuel assembly without known or suspected cladding defects greater than pinhole leaks or hairline cracks and which can be handled by normal means; or b) a BWR fuel assembly with an intact channel, a maximum planar average initial enrichment of 3.3 wt% U-235, without known or suspected GROSSLY BREACHED SPENT FUEL RODS, and which can be handled by normal means. An UNDAMAGED FUEL ASSEMBLY may be a REPAIRED/ RECONSTITUTED FUEL ASSEMBLY.”

The applicant stated that “for channeled BWR fuel, inspections to classify the fuel cladding as undamaged in accordance with the currently approved definition may be prohibitive from a cost, ALARA, or safety perspective. A particular subset of fuel, as described and analyzed in proposed FSAR Subsection 6.III.4.4, [low enriched, channeled BWR fuel] is shown to remain subcritical even if there was significant cladding damage and rearrangement of the fuel rods inside the channel. Therefore, if it can be determined that this fuel does not have gross cladding breaches, can be

handled by normal means, and has enrichment less than or equal to 3.3 wt%, then it does not require a damaged fuel container nor is it limited to certain basket locations in the MPC-68M." A specific example of this type of application is BWR fuel that has a condition crud induced localized corrosion (CILC) which is copper induced; this has the potential to provide corrosion-induced damage to the cladding, characterized by pitting, but does not cause grossly breached spent fuel rods.
The staff's evaluation in this area considered that, in BWRs, two types of crud deposits (plant corrosion products) can be found. The most frequently observed crud is low density, loosely adherent crud with good heat transfer capability. The other is a rarely observed tightly adherent crud of high density, through which lamination can lead to low heat transfer capability. Scale type crud contains high concentrations of copper, and heavy copper bearing crud was observed to be sandwiched between the zirconium metal and naturally occurring zirconium oxide at the cladding surface. This specific type of CILC has been discovered exclusively in BWR plants with copper containing condenser tubes.

Damage to spent fuel rods can be minor without causing gross cladding breaches or major that does cause gross cladding breaches. The applicant is proposing to classify SNF rods that do not have gross breaches, along with the other specified conditions, as undamaged.

ISG-1, Rev. 2, provides guidance for classifying fuel as either undamaged or damaged. Undamaged fuel may contain some cladding defects if the fuel is safeguarded from high temperatures/oxidation, and does not contain gross cladding breaches. The HI-STORM 100 Cask System MPCs are backfilled with helium, shown to keep the peak cladding temperature of the fuel below the guidance limits in ISG-11, Rev. 3; therefore fuel is protected during storage from temperatures that would lead to gross ruptures. As long as the fuel does not already contain a gross breach, ISG-1, Rev. 1, concludes that there is no means to release fragments during storage. In addition, fuel that contains an assembly defect may be considered undamaged per ISG-1, Rev. 1, if it can still meet fuel-specific and system related functions; therefore repaired and/or reconstituted assemblies, as proposed in the definition as part of this change, are considered undamaged.

Therefore, the staff finds that the applicants proposed definition of undamaged fuel complies with the specified conditions described above and is consistent with ISG-1, Rev. 1, and is therefore acceptable.

8.4.2 Repaired/Reconstituted Fuel Assembly:

The applicant has proposed the following definition for "fuel assembly": "Spent nuclear fuel assembly which contains dummy fuel rod(s) that displaces an amount of water greater than or equal to the original fuel rod(s) and/or which contains structural repairs so it can be handled by normal means." The applicant stated that this definition is provided for clarification purposes and is a subset of "Undamaged Fuel." It is a common practice for nuclear fuel assemblies to be repaired by the removal of a damaged fuel rod and replaced with a dummy rod. This allows the fuel assembly to be returned to the reactor core. The NRC has approved this use in specific applications, and has provided guidance to 10 CFR Part 50 licensees to ensure that this is performed within the requirements of the licensee's 10 CFR Part 50 TS without creating an unreviewed safety question. A repaired/reconstituted fuel assembly is restored to a condition within the bounds of its original design and safety analysis. The staff, therefore, finds this type of assembly to be a subset of “undamaged fuel” and the applicant’s proposed definition consistent with ISG-1, and therefore acceptable.

8.5 Materials Evaluation Summary:
FSAR Chapter 8 adequately describes the materials used for SSCs important to safety and the suitability of those materials for their intended functions in sufficient detail to evaluate their effectiveness.

8.6 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.

F8.2 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.

F8.3 The applicant has met the requirements of 10 CFR 72.122(h)(1) and 72.236(h). The design of the dry cask storage system and the selection of materials adequately protects the SNF cladding against degradation that might otherwise lead to damaged fuel.

F8.4 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.

F8.5 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.

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.

9.0 OPERATING PROCEDURES EVALUATION

There were no requested changes requiring an operating procedures evaluation to ensure compliance with the relevant general criteria established in 10 CFR Part 72.

10.0 ACCEPTANCE TESTS AND MAINTANANCE PROGRAM EVALUATION

There were no requested changes requiring an acceptance tests and maintenance program evaluation for the principal design criteria related to the SSCs important to safety to ensure compliance with the relevant general criteria established in 10 CFR Part 72.

11.0 RADIATION PROTECTION EVALUATION

There were no requested changes requiring a radiation protection evaluation for the principal design criteria related to the SSCs important to safety to ensure compliance with the relevant general criteria established in 10 CFR Part 72.

12.0 ACCIDENT ANALYSIS EVALUATION

There were no requested changes requiring an accident analysis evaluation for the principal design criteria related to the SSCs important to safety to ensure compliance with the relevant general criteria established in 10 CFR Part 72.

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 in CoC No. 1014, Amendment No. 9, Revision No. 1, continue to meet the requirements of 10 CFR Part 72. The evaluation is based on information provided by the applicant in this revision 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. Equipment changes and additions that required TS change evaluations were as follows:

(1) Change Burnup/Cooling Time limits for TPDs,
(2) Change Metamic-HT testing requirements,
(3) Change Metamic-HT MGVs,
(4) Update fuel definitions to allow BWR fuel affected by certain corrosion mechanisms within specific guidelines to be classified as undamaged fuel, and

The staff has provided an additional CoC condition that allows a general user up to 180 days to implement any changes and to update their 10 CFR 72.212 evaluations required by implementation of the revision. There are currently several general licensee users of licensee users of The staff has determined that because there are existing general licensee users of HI-STORM 100, Amendment No. 9, and these users have voluntarily agreed to accept the changes required by Revision 1. The staff has determined to incorporate a condition in the CoC to allow the current Amendment No. 9 users up to 180 days transition time to incorporate any hardware or procedural changes required by Revision 1. This also allows the general users 180 days to revise its 10 CFR 72.212 report. This implementation period has no effect on the health and safety of the public and is consistent with other implementation times granted by the NRC for similar license actions.

The corresponding CoC and TS changes are

(1) CoC; Added condition 13.

(2) TS Appendix B; Table 2.1-8 is modified by changing the Burnup/Cooling time limits for TPDs.
(3) TS Appendix B; Section 3.2.9: Neutron Absorber Tests: Removed reference to Section 9.111.2.2 of the HI-STORM 100 FSAR and added revised testing requirements for Metamic-HT.
(4) Fuel definition changes.
   A. Appendix A, Section 1.1 Definitions; a definition of GROSSLY BREACHED SPENT FUEL ROD is added to the table. (ISG-1)
   B. Appendix A, Section 1.1 Definitions; a definition of REPAIRED/RECONSTITUTEDFUEL ASSEMBLY is added to the table.
   C. Appendix A, Section 1.1 Definitions; the definition of UNDAMAGED FUEL ASSEMBLY is added to the table.
   D. Appendix B, Section 2.1.1.a, "UNDAMAGED FUEL ASSEMBLIES" is added to the paragraph.
   E. Appendix B, Section 2.1.3, "or UNDAMAGED FUEL ASSEMBLIES" is added to the paragraph.
   F. Appendix B, Table 2.1-1, Section VI; "INTACT FUEL ASSEMBLIES" is changed to UNDAMAGED FUEL ASSEMBLIES" in all places.
   G. Appendix B, Table 2.1-3; Note 16 is modified. "INTACT FUEL ASSEMBLIES" is changed to "UNDAMAGED FUEL ASSEMBLIES" in all places.
   H. Appendix B, Table 2.1-3; Note 19 is added.
   I. Appendix B, Table 2.4-1; "or Undamaged" is added to the column label for "Intact Fuel Assemblies".
   J. Appendix B, Section 2.4.2; "or Undamaged" is added to the parenthetical statement "(Intact Fuel only)".
   K. Appendix B, Section 3.2.7; "or undamaged" is added to the paragraph.

13.2 Findings

F13.1 The staff finds that CoC No. 1014, Amendment No. 9, continues to identify necessary TS to satisfy 10 CFR Part 72 and that the applicable criteria of 10 CFR 72.236 have been satisfied. The proposed TS changes provide assurance that the HI-STORM 100 Cask System will continue to allow safe storage of spent nuclear fuel.

14.0 CONCLUSIONS

Based on its review of the revision request to CoC No. 1014, Amendment No. 9, the staff has determined that there is reasonable assurance that: (i) the activities authorized by the amended certificate can be conducted without endangering the health and safety of the public and (ii) these activities will be conducted in compliance with the applicable regulations of 10 CFR Part 72. The staff has further determined that the issuance of the revision will not be inimical to the common defense and security. Therefore, the revision should be approved.

March 21, 2016