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FINAL SAFETY EVALUATION REPORT

DOCKET NO. 72-1004
TN AMERICAS LLC
CERTIFICATE OF COMPLIANCE NO. 1004
STANDARDIZED NUHOMS® SYSTEM
AMENDMENT NO. 15

SUMMARY

This safety evaluation report (SER) documents the U.S. Nuclear Regulatory Commission (NRC) staff's review and evaluation of the amendment request to amend Certificate of Compliance (CoC) No. 1004 for the Standardized NUHOMS® System. By letter dated March 28, 2017 (Agencywide Documents Access and Management System (ADAMS) Accession Number ML17094A714), as supplemented on July 18, 2017 (ADAMS Accession Number ML17202Q145), and December 14, 2017 (ADAMS Accession Number ML17363A276), and March 22, 2018 (ADAMS Accession Number ML18088A180), TN Americas LLC, from here on referred to as the "applicant", requested that NRC amend the CoC to include the following changes:

1. Unify and standardize the fuel qualification tables for four pressurized water reactor (PWR) systems (32PT, 24PTH, 32PTH1 and 37PTH) in order to simplify the Technical Specifications (TS). The standardized fuel qualification tables (FQTs) provide for minimum required cooling times, as low as two years, as a function of enrichment and burnup (BU) for all the heat loads described in the various heat load zoning configurations (HLZCs) for these four PWR systems. Further, the FQTs are generated for three different metric tons of uranium (MTU) loadings per fuel assembly (FA) and allow for interpolation between MTU loadings and to establish cooling times for FAs that fall into the unanalyzed regions of the FQTs. For this purpose, the source term, dose rate, occupational exposure and site dose analyses have been revised for the four PWR systems described above. The TS and Updated Final Safety Analysis (UFSAR) Appendices M, P, U and Z have been revised accordingly.
2. For the 32PT System, add a new HLZC #4 to allow for the loading of FAs with decay heat up to 2.2 kW corresponding to a 2-year cooled fuel. The TS and UFSAR Appendix M have been revised to incorporate this new HLZC.
3. For the 32PT System, increase the maximum assembly average BU from 55 gigawatt-days per metric ton of uranium (GWd/MTU) to 62 GWd/MTU. The TS and UFSAR Appendix M have been revised to incorporate this change.
4. For the 32PT System, allow for the loading of damaged fuel assemblies confined within top and bottom end caps and failed fuel assemblies loaded within individual failed fuel canisters. Provide for a basket option to increase the number of poison plates from 24 to 32 resulting in an increase in the allowable enrichment of the authorized contents. Expand the definition of the poison rod assemblies (PRAs) to include rod cluster control assembly (RCCA) materials, specifically silver neutron absorber. A clarification of the definition for damaged fuel for all DSCs was also made in the UFSAR sections and TS

Enclosure

tables. Additionally, the TS now has a separate definition for intact fuel. The TS and UFSAR Appendix M have been revised to incorporate this change.

5. For the 32PT System, include other zirconium alloy cladding materials such as ZIRLO and M5. The TS and UFSAR Appendix M have been revised to incorporate this change.
6. For the 24PTH System, add a new HLZC #6 to allow for the loading of FAs with decay heat up to 2.5 kW corresponding to a 2-year cooled fuel, and a total heat load of 35 kW per basket. The TS and UFSAR Appendix P have been revised to incorporate this new HLZC and editorial changes are made to the TS for the descriptions of basket types.
7. For the 24PTH System, the OS197 is added as an authorized transfer cask (TC) for the transfer of the 24PTH-S-LC dry shielded canister (DSC) in addition to the standardized TC. UFSAR Chapters P.1, P.2 and P.4 have been revised to incorporate this change.
8. For the 61BTH System, revise the existing HLZC #10 to allow loading FAs with decay heat up to 1.2 kW corresponding to a 2-year cooling time. GNF-2 and ATRIUM-11 FA designs are also added as authorized contents. Additionally, the FQTs with minimum cooling times of two years are generated for MTU loadings of 0.180 and 0.198 per fuel assembly at a decay heat of 1.2 kW and to establish cooling times for FAs that fall into the unanalyzed regions of all the FQTs. The TS and UFSAR Appendix T have been revised to incorporate these changes.
9. For the 32PTH1 System, add new HLZC #5 to allow for the loading of FAs with decay heat up to 1.1 kW for a total heat load of 35.2 kW per basket and HLZC #6 to allow for loading of FAs with decay heat up to 1.3 kW for a total heat load of 37.6 kW per basket. This is applicable for Type 1 DSCs using solid aluminum rails only. The TS and UFSAR Appendix U have been revised to incorporate these changes.
10. Provide a description in the UFSAR for the solar shield currently described in the TS for the TC during transfer operations. UFSAR Chapter 10 has been revised to incorporate this change.
11. Update Technical Specification 4.3.3 Item 11 to add flexibility to general licensees in verifying compliance regarding the storage pad location and the soil-structure interaction, which may affect the response of loaded horizontal storage modules (HSMs).

The amended CoC, when codified through rulemaking, will be denoted as Amendment No. 15 to CoC No. 1004. This SER documents the review and evaluation of the proposed amendment. The staff followed the guidance of NUREG-1536, Revision 1, “Standard Review Plan for Spent Fuel Dry Storage Systems at a General License Facility”; Interim Staff Guidance (ISG) -11, “Cladding Considerations for the Transportation and Storage of Spent Fuel”; ISG-21, “Use of Computational Modeling Software”; 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 the applicant’s application and whether it meets the applicable requirements of 10 CFR Part 72 for dry storage of spent nuclear fuel. The staff’s evaluation focused only on modifications requested in the amendment as supported by the submitted revised UFSAR (see ADAMS Accession Nos. ML17094A720, ML17202Q145, ML17363A280 and ML18088A180) and did not reassess previous revisions of the UFSAR nor previous amendments to the CoC.

1.0 GENERAL DESCRIPTION

The objective of this chapter is to review the changes requested to CoC No. 1004 for the Standardized NUHOMS® System to ensure that the applicant provided an adequate description of the pertinent features of the storage system and the changes requested in the application. The specific changes requested by the applicant are described and evaluated in the following sections of this SER.

2.0 PRINCIPAL DESIGN CRITERIA EVALUATION

The applicant did not propose any changes that affect the staff's principal design criteria evaluation provided in previous safety evaluations for CoC No. 1004, Amendments No. 1 through 11, 13 and Amendment No. 14. Therefore, the staff determined that a new evaluation was not required.

3.0 STRUCTURAL EVALUATION

The staff reviewed the proposed changes to Amendment No. 15 to the Standardized NUHOMS® System to ensure that TN had performed adequate structural evaluation of the system to demonstrate its acceptance. In reviewing the eleven proposed changes summarized in Enclosure 7 of the supplement to the application dated March 22, 2018 (ADAMS Accession Number ML18088A180), the staff performed a screening assessment of the structural performance of the fuel basket associated with the DSC emplacement of fuel assemblies in different heat load zoning configurations (HLZCs). This assessment was done by noting that loading fuel in different HLZCs would result in minute differences in the basket inertia load distribution effect, and the temperature dependent basket assembly material moduli of elasticity would remain essentially unchanged. Therefore, on the basis of engineering judgement, the staff has reasonable assurance to conclude that the previously calculated basket stresses are applicable for evaluating the stress performance of the basket. Additionally, since the basket temperature profiles, which affect the component stress allowables determination and thermal stress analysis, are also bounded by those evaluated previously, the staff concludes that the basket stress margins of safety and the thermal stresses continue to be bounded by the previous analyses.

On this basis of the above assessment, the staff determined that the changes pertaining to further structural evaluation consist primarily of two changes. The two changes are the addition of:

1. Change #8, the new GNF2 and ATRIUM 11 fuel types to be stored in the 61BTH DSC System, and
2. Change #11, the Technical Specifications design features clarification to provide flexibility to general licensees in verifying compliance regarding the storage pad location site specific parameters evaluation requirements.

The staff's structural evaluation with respect to the proposed changes is as follows:

<u>GNF2 and ATRIUM 11 Fuel - Cask Corner Drop Accident</u>: Section T.3.5.5 of Appendix T to the FSAR evaluates the cladding structural performance of the GNF2 and ATRIUM 11 fuel assemblies for the 80-inch transfer cask corner drop handling accident.

The applicant compared physical attributes of the fuel rod and constraints accorded by the fuel compartment walls for the GNF2 fuel to those of the ABB-10-2 fuel and determined that the latter is bounding for the 10x10 fuel category. This determination indicates that the bounding ABB-10-2 10x10 fuel type evaluated in Appendix Y, Section Y.3.5.3 for the cask handling corner drop accident for the for 61BTH DSC fuel assemblies is applicable for the GNF2 fuel assembly 80-inch corner drop condition in the 61BTH basket. Therefore, no further corner drop accident evaluation is needed for the GNF2 fuel.

For the ATRIUM 11 fuel evaluated in Section T.3.5.5, the applicant noted that there was no change introduced to the methodology, assumptions, and model attributes, including the time history loading and boundary and initial conditions, used in the previous finite element analyses of the fuel assemblies. By using the ATRIUM 11 specific fuel geometry and fuel compartment data together with the same initial rod internal pressure as before, the applicant again computed the maximum cladding principal strain and maximum stress. The staff reviewed the results and found them well below the at-temperature material yield strain and yield stress. Thus, the staff has reasonable assurance to conclude that the fuel rod would retain its elastic behavior and no permanent fuel rod deformation would need to be considered for evaluating the criticality safety to meet the 72.236(b) and (c) requirements.

<u>GNF2 and ATRIUM 11 Fuel - Damaged Fuel Cladding Integrity Assessment</u>: Section T.3.6.2.3 of Appendix T to the application performs a damaged fuel structural integrity assessment for off-normal loads for the GNF2 and ATRIUM 11 fuel types. The assessment was done by applying a bounding fuel rod inertia load exerted separately along the transfer cask lateral and vertical downward directions during the fuel handling operation. By following the same approaches used for other bounding 7x7, 8x8, 9x9, and 10x10 fuel types, the applicant determined the maximum cladding bending stress for the ATRIUM 11 fuel. Correspondingly, the applicant evaluated the maximum cladding bending stress of the previously approved 10x10 ATRIUM fuel and determined it to be bounding and applicable to the 10x10 GNF2 fuel types. Furthermore, the calculated fuel cladding bending stresses were also used in a fracture mechanics evaluation to demonstrate that, for the two geometries of cladding defects greater than pinhole leak or hairline cracks, the extent of cladding damage for the fuel assembly is to be limited for the off-normal loading conditions.

In Section T.3.6.3.2 of Appendix T to the UFSAR, the applicant evaluated fracture behavior for Zircaloy cladding tubes for two fracture geometries. The fracture Geometry #1 is a through-wall circumferential crack and fracture Geometry #2 is a crack emanating from a circular hole in the tube simulating the burst opening defect in a fuel rod. Considering the same equation used previously for fracture Geometry #1 for other approved cladding material, the applicant calculated the Mode I stress intensity factor, K_{IC} to be comparable to those for the respective 7x7, 8x8, 9x9 and 10x10 bounding fuel types. Similarly, for the applicable equation for Geometry #2, the applicant calculated the Mode I stress intensity factor, K_{IC} to demonstrate that it is also comparable to those for the bounding fuel types.

On the basis of the above review, the staff concludes that the stresses calculated for all off-normal loadings are less than the yield stress of the Zircaloy-2 cladding material. Since the corresponding stress intensity factors are well below that for crack initiation under the off-normal loadings, the staff has reasonable assurance to conclude that GNF-2 and ATRIUM 11 fuel types can be safely handled as damaged fuel assemblies in the 61BTH DSC System in meeting the 72.236(a) and (b) requirements.

<u>Technical Specifications Clarification - Site Parameters and Analysis</u>

Section 4.3.3, "Site Specific Parameters and Analyses," of the Technical Specifications provides that the potential Standardized NUHOMS® System user (general licensee) shall perform the verifications and evaluations in accordance with 10 CFR 72.212 before the use of the system under the general license. The proposed Amendment No. 15 requested that, under item No. 11, the site parameter reconciliation analysis provision be revised to read, "The storage pad location shall be evaluated for the effects of soil-structure interaction which may affect the response of the loaded HSMs. *Seismic responses at the location of the HSM center of gravity (CG) may be obtained from the soil-structure interaction analyses* (added change italicized)." The staff reviewed the item No. 11 provision and concludes that the added clarification language will provide flexibility to general licensees in performing a 10 CFR 72.212(b)(5) site parameter reconciliation evaluation associated with the storage pad location. The staff finds the added change provides Technical Specifications clarity, yet recognizes the need for consideration of the effects of soil-structure interaction that may amplify the design basis seismic response of loaded HSMs under the site-specific seismic design parameters.

3.1 Findings

F3.1 On the basis of the review of the statements and representations in the application, the staff concludes that the proposed changes to the structural design and operations for Standardized NUHOMS® System continues to meet the requirements of 10 CFR Part 72.

4.0 THERMAL EVALUATION

The objective of the staff's review of the applicant's thermal evaluation for Amendment No. 15 to the Standardized NUHOMS® System is to verify that the cask and fuel material temperatures will remain within the range of allowable values or criteria for normal, off-normal, and accident conditions. Specifically, the staff analyzed whether the temperatures of the fuel cladding will meet regulatory requirements throughout the storage period and will protect the cladding against degradation that could lead to gross rupture. The applicable regulatory requirements are found in 10 CFR 72.236(b), 72.236(f), 72.236(g), and 72.236(h).

The staff reviewed the information provided in the amendment request to determine whether the Standardized NUHOMS® System continues to fulfill the acceptance criteria listed in Chapter 4 of NUREG-1536, Rev. 1, "Standard Review Plan for Spent Fuel Dry Storage Systems at a General License Facility."

The changes requested by the applicant relevant to the thermal evaluation do not involve design changes to the major components of the Standardized NUHOMS® System and are related to the authorized contents. The following changes in the application that involve thermal consideration are described below:

1. Change #2, 32PT Added Heat Load Zoning Configuration (HLZC) #4 (UFSAR Appendix M & TS)
2. Change #6, 24PTH Added HLZC #6 (UFSAR Appendix P & TS)
3. Change #8, 61BTH Revised HLZC #10 (UFSAR Appendix T & TS)
4. Change #9, 32PTH1 Added HLZC #5 and HLZC #6 (UFSAR Appendix U & TS)
5. Change #4, 32PT Damaged Fuel Assemblies (UFSAR Appendix M & TS)
6. Change #4, 32PT Failed Fuel Assemblies (UFSAR Appendix M & TS)
7. Change #4, 32PT 32 Single Poison-Plate Configuration (UFSAR Appendix M & TS)
8. Change #7, OS197 Transfer Cask added as authorized Transfer Cask (TC) for 24PTH (UFSAR Appendix P & TS)

4.1 Standardized NUHOMS® New Heat Load Zoning Configurations

4.1.1 Standardized NUHOMS® 32PT DSC with HLZC #4 (Change 2)

Section M.4.12.3 of the application describes the thermal evaluation of the 32PT DSCs with HLZC #4. The HLZC #4 has a maximum decay heat of 30.8 kW per DSC, however, the 32PT DSC is limited to 24kW.

The thermal analysis of HLZC #4 was based on a sensitivity analysis of HLZC #1 (24 kW), which was determined to be the bounding case. This analysis of HLZC #4 utilized the same input parameters and model as HLZC #1 with the exception of the heat load pattern. The staff reviewed the analysis results of the sensitivity study and finds that the fuel cladding temperatures and all but one DSC maximum component temperatures for HLZC #4 remain below the reported bounding temperatures for HLZC #1, as asserted by the applicant. The rail temperature values exceed the bounding value by 20°F. Section M.3.6.1.3.2 presents an evaluation of this temperature excursion and asserts that it is not safety significant. Staff agrees with this conclusion since the component temperatures are still below allowable limits.

On the basis of the sensitivity study, the applicant also concluded that the temperature of the 32PT DSC fuel cladding will remain at or below the limits for all normal, off-normal, and accident conditions for HLZC #4. This conclusion was also applied to the transfer time limits of HLZC #1, which will bound transfer time limits for HLZC #4.

The applicant also reported that the average helium temperature and internal canister pressure for HLZC #4 exceeds that of the bounding value of HLZC #1. Since the average helium temperature in the DSC for HLZC #1 was 9°F lower than the average helium temperature in the DSC for HLZC #4, staff finds this temperature excursion will not have a safety significant effect on the internal pressure.

4.1.2 Standardized NUHOMS® 24PTH Type 1 DSC with HLZC #6 (Change 6)

Section P.4.10 of the application describes the thermal evaluation of the 24PTH Type 1 DSC with HLZC #4. The HLZC #6 has a maximum decay heat of 35.2 kW per DSC. The thermal analysis of HLZC #6 was based on a sensitivity analysis of HLZC #1 (40.8 kW), which was determined to be the bounding case. This analysis of HLZC #6 utilized the same input parameters and model as HLZC #1 with the exception of the heat load pattern. The staff reviewed the analysis results of the sensitivity study and concludes that the fuel cladding temperatures and DSC maximum component temperatures for HLZC #6 remain below the reported bounding temperatures for HLZC #1, as asserted by the applicant.

On the basis of the sensitivity study, the applicant also concluded that the temperature of the 24PTH DSC fuel cladding will remain at or below the limits for all normal, off-normal, and accident conditions for HLZC #6. This conclusion was also applied to the transfer time limits of HLZC #1, which will bound transfer time limits for HLZC #6.

The applicant also concluded that the average helium temperature and internal canister pressure remains bounded by HLZC #1. Since the average helium temperature in the DSC for HLZC #1 was larger than the average helium temperature in the DSC for HLZC #6, staff finds this conclusion acceptable.

4.1.3 Standardized NUHOMS® 61BTH DSC with revised HLZC #10 (Change 8)

Section T.4.6.10.2.2 of the application describes the thermal evaluation of the 61BTH DSC for the revised HLZC #10, which provides an alternate option for this HLZC. The maximum decay heat for HLZC #10 remains 31.2 kW for the 61BTH Type 2 DSC, however the maximum individual fuel assembly heat load for Zone 3 changes from 0.9 kW to 1.2kW. The maximum decay heat for HLZC #10 is equivalent to the decay heat for the 61BTH DSCs reported by the applicant in Amendment No. 14.

The thermal analysis of alternate HLZC #10 was based on a sensitivity analysis of HLZC #5 to HLZC #8 (31.2 kW), which was determined to be the design basis case. This analysis of HLZC #10 utilized the same input parameters and model as the design basis HLZCs with the exception of the heat load pattern. The staff reviewed the analysis results of the sensitivity study and concludes that the fuel cladding temperatures and DSC maximum component temperatures for the alternate HLZC #10 remain at or below the reported bounding temperatures for the design basis HLZC, as asserted by the applicant. In two cases, for the R45/R90 rails and the Top Grid, the applicant reported component temperatures 3°F and 2°F above the design basis temperatures, respectively. The NRC does not find this temperature excursion safety significant.

On the basis of the sensitivity study, the applicant also concluded that the temperature of the 61BTH Type 2 DSC fuel cladding will remain at or below the design basis limits for all normal, off-normal, and accident conditions for alternate HLZC #10. This conclusion was also applied to the transfer time limits of HLZC #7, which will bound transfer time limits for alternate HLZC #10.

The applicant also concluded that the average helium temperature and internal canister pressure remains bounded by the design basis HLZC. Since the average helium temperature in the DSC for the design basis HLZC was the same as the average helium temperature in the DSC for HLZC #10, staff finds this conclusion acceptable.

4.1.4 Standardized NUHOMS® 32PTH1 Type 1 DSC with HLZC #5 and HLZC #6 (change #9)

Section U.4.11 of the application describes the thermal evaluation of the 32PTH1 DSCs with HLZC #5 and HLZC #6. The HLZC #5 has a maximum decay heat of 35.2 kW per DSC and HLZC #6 has a maximum decay heat of 37.6 kW per DSC. The applicant opted to perform a sensitivity analysis for the two proposed HLZCs because no design changes were made to the 32PTH1 canister. The maximum decay heat for HLZC #6 was determined to be bounding for this sensitivity evaluation since both the total heat load and individual fuel assembly heat loads were higher than for HLZC #5.

The staff reviewed the analysis results of the sensitivity study and observed that the fuel cladding temperatures and all DSC maximum component temperatures for the HLZC #6 remain at or below the reported bounding temperatures for the design basis HLZC #1, as asserted by the applicant for the normal condition case utilizing 106°F ambient temperature. Since there is no instance where peak cladding temperature or component temperatures for normal conditions exceed those for the bounding HLZC #1, the applicant assumed that the temperatures for off-normal and accident conditions would also bound those for HLZC #5 and HLZC #6. The staff finds this assumption reasonable. This conclusion was also applied to the transfer time limits of HLZC #1, which will bound transfer time limits for alternate HLZC #5 and HLZC #6

The staff does note however that the analysis neglected to assign thermal insolation to exposed surfaces which will produce unconservative results. In this instance, the staff does not believe that the absence of solar insolation is a sufficient enough omission to overcome the reported margin of safety of 48°F for the peak cladding temperature, therefore no further evaluation or corrections will be requested as part of this licensing action.

4.2 Storage of up to 28 Damaged PWR Fuel Assemblies in Standardized NUHOMS® 32PT DSC (change #4)

Section M.4.12.1 of the application described loading up to 28 damaged PWR fuel assemblies to be stored in the 32PT DSC. In addition, based on Table 1-1e of the Technical Specifications, there could be anywhere from 4 to 32 intact fuel assemblies with up to 28 damaged fuel assemblies (such that the total of intact and damaged fuel assemblies adds up to 32). The applicant's analyses of thermal performance of intact fuel for normal and off-normal conditions remains valid for damaged fuel.

For the hypothetical drop accident, the applicant performed a sensitivity study considering numerous configurations of both intact and damaged fuel. The most limiting configuration of this sensitivity study considered the cladding temperature of a single intact fuel assembly surrounded by rubblized fuel. This value of cladding temperature for the intact fuel assembly is below the recommended ISG-11 value of 1058°F, therefore the staff finds this acceptable.

The reported bounding average helium temperature for the 32PT loaded with damaged fuel assemblies was less than the bounding average helium temperature of 705°F for intact fuel assemblies. As such the internal pressure for the 32PT DSC loaded with damaged fuel will be less than the reported value associated with the average helium temperature of the 32PT DSC with fully intact fuel. In both cases, the internal pressure is lower than the DSC cavity allowable pressure of 125 psig, therefore the staff finds these results acceptable.

4.3 Storage of up to 8 Failed PWR Fuel Assemblies in Standardized NUHOMS® 32PT DSC (change #4)

Section M.4.12.2 of the application described loading a 32 PT DSC with a heat load configuration containing up to 8 failed PWR fuel assemblies in failed fuel cans. Loaded failed fuel cans have a maximum heat load of 0.8kW.

The applicant performed a sensitivity study to determine the maximum component temperatures with 8 failed fuel assemblies assumed as rubble during the bounding accident conditions. The configurations considered were 24 intact/8 failed and the loading pattern is illustrated in Figure M.4.12.1-1 of the FSAR, with the 8 failed fuel assemblies in Position A. These configurations were assessed using HLZC #2 with a maximum heat load of 24kW. The results of the sensitivity study demonstrated that the maximum calculated intact fuel cladding temperature was less than the temperature of 863°F for the bounding design basis values generated for 32 intact fuel assemblies using HLZC #2. Further, this value for the intact fuel assembly is below the recommended ISG-11 value of 1058°F, therefore the staff finds this acceptable.

The reported bounding average helium temperature for the 32PT loaded with failed fuel assemblies was less than the bounding average helium temperature or 705°F for intact fuel assemblies. As such the internal pressure for the 32PT DSC loaded with failed fuel will be less than the pressure for the average helium temperature of the 32PT DSC with fully intact fuel.

4.4 Storage of 32 Single Poison-Plate Configuration in Standardized NUHOMS® 32PT DSC (change #4)

Section M.4.12.4 of the application described the thermal evaluation for a configuration of 32 Single Poison Plates to be use in the 32PT DSC. This change is limited to replacing 8 aluminum chevrons with 8 single poison plates. The applicant determined that since there were no other design changes, a sensitivity study would be appropriate using a bounding HLZC. Evaluation by inspection illustrated that HLZC #3 was the bounding case as it had the highest heat load per fuel assembly in the inner portion of the basket.

The results of the sensitivity study demonstrated that the maximum calculated intact fuel cladding temperature was 723°F, which is 3°F greater than the temperature of 720°F for the bounding design basis HLZC #3. Since a bounding case was used and the reported cladding temperature is below the recommended limit of 752°F in ISG-11, the increase of 3°F is not safety significant for normal storage and short term transfer. Furthermore, during bounding accident conditions, the maximum reported temperature for the fuel cladding has a margin of 195°F. The staff finds that lowering the margin by 3°F is not safety significant.

4.5 Addition of OS197 as Authorized Transfer Cask for 24PTH System (change #7)

Section P.4.11 of the application describes the thermal evaluation of the 24PTH-S-LC DSC in the OS197 Transfer Cask. The applicant used a similarity argument between the 24PTH-S-LC and the 32PT canisters, rather than performing an explicit two-step 2D analysis for the 24PTH-S-LC. The similarity approach took advantage of the geometric, material, and heat load similarities between the two canister systems to allow for the assumption that the canister surface temperatures for the 24PTH-S-LC would be nearly identical to the previously evaluated 32PT for normal, off-normal, and accident conditions, inside the OS197 Transfer Cask. The applicant further argued that since the normal and off-normal shell temperatures for the 24PTH-S-LC in the Standardized Transfer Cask exceed those of the 32PT in the OS197, it could be concluded that the previously approved Standardized Transfer Cask analysis results for the 24PTH-S-LC bound those that would be obtained from the OS197 in a two-step 2D analysis.

In the case of accident conditions, the OS197 Transfer Cask surface shell temperature of 600°F exceeded the shell surface temperature of 487°F determined in the Standardized TC. However, this temperature of 600°F is bounded by the blocked vent accident condition of the 24PTH-S-LC DSC in HSM Model 102, which was analyzed based on a shell temperature of 613°F that produced a peak cladding temperature of 821°F which was significantly below the accident temperature limit of 1058°F and consistent with the guidance of ISG 11, Rev 3.

Similarly for the canister pressure calculations, the applicant used the average internal helium temperature (618°F) for the aforementioned blocked vent case to calculate the internal pressures in the 24PTH-S-LC. The results of the calculation demonstrated that the internal pressure value of 85.81 psig was below the allowable limit of 90 psig.

4.6 Changes in Technical Specifications

The staff reviewed the proposed changes to the Technical Specifications for the Standardized NUHOMS® System to implement the requested changes in fuel types and fuel load configurations requested in the amendment application. The staff's evaluation of these proposed changes to the Technical Specifications is found in Chapter 13 of this SER.

4.7 Findings

F4.1 The staff has reasonable assurance that the structures, systems, and components (SSCs) important to safety are described in sufficient detail in Appendices M, P, T, U, and Z of the SAR to enable an evaluation of their thermal effectiveness. Cask SSCs important to safety remain within their safe operating temperature ranges.

F4.2 The staff has reasonable assurance that the Standardized NUHOMS® 32PT, 24PTH, 61BTH, 32PTH1 DSCs within the HSM or HSM-H systems are designed with a heat-removal capability having verifiability and reliability consistent with its importance to safety. The casks are designed to provide adequate heat removal capacity without active cooling systems.

F4.3 The staff has reasonable assurance that the spent fuel cladding is protected against degradation which could lead to gross ruptures. The cladding temperature will be maintained below maximum allowable limits in a helium gas environment in the cask cavity under normal, off-normal, and accident storage conditions. In addition, protecting the cladding against degradation is expected to allow ready retrieval of spent fuel for processing or disposal at a later time.

F4.4 The staff concludes that the thermal design of the Standardized NUHOMS® 32PT, 24PTH, 61BTH, 32PTH1 DSCs within the HSM or HSM-H systems complies with 10 CFR Part 72, and furthermore, that the applicable design and acceptance criteria are satisfied. The staff finds the thermal design provides reasonable assurance that the Standardized NUHOMS® 32PT, 24PTH, 61BTH, 32PTH1 DSCs within the HSM or HSM-H systems will allow safe storage of spent fuel for a licensed life of 40 years. The staff reached this finding after 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 confinement design changes requested in the amendment application requiring additional evaluation of the confinement criteria related to the structures, systems, and components important to safety to ensure compliance with the relevant general criteria established in 10 CFR Part 72. Internal pressures changes were investigated as part of the thermal evaluation and found to be either bounding, or non safety significant for all cases, therefore no further confinement evaluation was necessary.

6.0 SHIELDING EVALUATION

The applicant requested to amend CoC No. 1004 for the Standardized NUHOMS® System design to allow for higher decay heat loads for the 61BTH and four PWR canisters (32PT, 24PTH, 32PTH1 and 37PTH) and loading of damaged and failed fuels. The objective of this shielding review is to evaluate the proposed shielding features of the Standardized NUHOMS® System design as amended to determine if the design with the proposed changes will continue to provide adequate protection to workers and the public from radiation from the dry storage system contents. This review seeks to ensure that the shielding design is reasonably capable of meeting the operational dose requirements of 10 CFR 72.104 and 72.106 in accordance with 10 CFR 72.236(d).

The staff reviewed the applicant's safety analyses for the requested changes to the CoC following the guidance provided in NUREG-1536, "Standard Review Plan for Dry Cask Storage Systems," Revision 1. This safety evaluation report documents the staff's evaluation of the following proposed changes related to shielding design:

1. Change #1, the FQTs for four PWR canisters (32PT, 24PTH, 32PTH1 and 37PTH)
2. Change #2, the 32PT canister with a new HLZC #4 to allow for the loading of FAs with decay heat up to 2.2 kW corresponding to a 2-year cooled fuel,
3. Change #3, the 32PT canister with increased maximum assembly average burnup from 55 GWd/MTU to 62GWd/MTU.

4. Change #4, the 32PT canister to allow for the loading of damaged FAs confined within top and bottom end caps and failed FAs loaded within individual failed fuel canisters and increase the number of poison plates from 24 to 32 resulting in an increase in the allowable enrichment of the authorized contents. The applicant also requested authorization for storage of up to 28 damaged fuels or up to 8 failed fuels in 32 PT canister.
5. Change #5, Evaluation of the 32PT canister to include other zirconium alloy cladding materials such as ZIRLO and M5™.
6. Change #6, the 24PTH canister to add a new HLZC #6 ·to allow for the loading of FAs with decay heat up to 2.5 kW corresponding to a 2-year cooled fuel, and a total heat load of 35 kW per basket.
7. Change #7, the 24PTH canister, to add the OS197 as an authorized TC for the transfer of the 24PTH-S-LC DSC in addition to the standardized TC.
8. Change #8, the 61BTH canister to revise the existing HLZC #10 to allow loading FAs with decay heat up to 1.2 kW corresponding to a 2-year cooling time. Also add the GNF-2 and ATRIUM-11 FA designs as authorized contents.
9. Change #9, the 32PTH1 canister to add new HLZC #5 to allow for the loading of FAs with decay heat up to 1.1 kW for a total heat load of 35.2 kW per basket and HLZC #6 to allow for loading of FAs with decay heat up to 1.3 kW for a total heat load of 37.6 kW per basket.

Unified / Standardized Fuel Qualification Tables

The applicant proposed new consolidated/standardized FQTs for the 32PT, 24PTH, 32PTH1 and 37PTH PWR systems to provide minimum cooling times, enrichment, and burnup for all heat loads described in the various heat load zoning configurations (HLZCs) for the four PWR systems. Additionally, the FQTs were generated for three different uranium loadings per FA. The FQTs allow interpolation between uranium loadings per FA. The applicant revised the source term, dose rate, occupational exposure and site dose analyses for the four PWR systems. The design basis source term was previously determined for the 0.490 MTU/FA using the SAS2H/ORIGEN-S modules of SCALE 4. 4. The proposed new design basis source term is determined using the SCALE6.0/ORIGEN-ARP.

New FQTs were generated for the 32PT, 32PTH1, 24PTH, and 37PTH DSCs for three uranium loadings of 0.490, 0.475, and 0.380 MTU per FA, burnups, maximum assembly average initial U-235 enrichments, and minimum cooling times. A new heat zone HLZC #4 is proposed to be added to the 32 PT, new HLZC #6 is proposed to be added to the 24 PTH, and two new heat zones HLZC #5 and HLZC #6 are proposed to be added to the 32PTH1. The existing HLZC *#10* is proposed to be revised for the 61BTH canister. The applicant provided revised source term, dose rate, occupational exposure and site dose analyses for the 32PT, 24PTH, 32PTH1 and 37 PTH systems and determined the new bounding HLZCs. The changes to the above systems are described below. The technical specification and SAR were revised to reflect the new FQTs generated.

Standardized NUHOMS® 37PTH DSC

The 37PTH system is a modular canister based spent fuel storage and transfer system with two alternate configurations depending on the canister length. The 37PTH DSC is designed for a maximum heat load of 30.0 kW and is designed with solid aluminum rails for support and to facilitate heat transfer.

The 37PTH DSC is authorized to accommodate up to 37 intact (or up to 4 damaged and balance intact) PWR fuel assemblies with or without control components (CCs). The CCs include burnable poison rod assemblies (BPRAs), control rod assemblies (CRAs), rod cluster control assemblies (RCCAs), thimble plug assemblies (TPAs), axial power shaping rod assemblies (APSRAs), orifice rod assemblies (ORAs), vibration suppression inserts (VSIs), neutron sources, and neutron source assemblies (NSAs).

Design Features and Criteria:

The applicant did not propose any changes to the shielding design features and criteria for the 37PTH DSC.

Source term:

The source term for the 37PTH DSC is presented in Section Z.5.2 of SAR Appendix Z. The applicant performed gamma and neutron source term calculations using the *ORIGEN-ARP* modules of the SCALE 6.0 computer code. The fuel types considered in this application are listed in Table Z.2.2. The B&W 15x15 assembly types was selected as the bounding fuel assembly for the shielding analyses because it has the highest initial heavy metal loading and Co-60 content compared to other fuel assemblies listed in Section Z.2 of the SAR.

Shielding analysis

The applicant used MCNP 5 to model the 37PTH DSC in the HSM with a new source term using the 0.490 MTU/FA loading. The same model was re-run for 0.380 MTU/FA loading to determine the effect of uranium loading on the dose rates. The applicant calculated scaling factors by comparing the dose rates at the same positions along the canister for the 0.490 and 0.380 MTU/FA loading. For example, dose rates calculated at different locations along the side of HSM for 0.380 uranium loading are divided by the dose rates calculated at the same corresponding positions for the 0.490 uranium loading. The highest ratio factor is then used to calculate new dose rates for the 0.380 MTU loading by multiplying dose rates on the side of HSM by the highest factor. This results in the calculated dose rates for 0.380 MTU/FA loading to be more conservative than the results obtained from MCNP calculation for 0.490 uranium loading. These scaling factors are shown in the footnotes of the Tables Z.5-1(Dose Rates for HSM-H Containing NUHOMS®-37PTH DSC), Z.5-2 (Dose Rates during Transfer Operations), and Z.5-3 (Dose Rates during Decontamination and Welding Operations).

Staff evaluation

The staff reviewed the technical specifications section with respect to shielding and radiation protection issues for the 37PTH. Staff independently evaluated the source term, dose rate, occupational exposure and site dose analyses and concluded that the information described in the technical specifications provide reasonable assurance that DSC 37PTH will allow for the safe transfer and storage of the spent fuel according to the regulatory requirements for providing adequate radiation protection to licensee personnel and members of the public include 10 CFR Part 20, 10 CFR 72.104(a), 10 CFR 72.106(b), 10 CFR 72.212(b), and 10 CFR 72.236(d).

Standardized NUHOMS® 32PT DSC

The 32PT DSC is authorized to safely store 32 intact standard PWR fuel assemblies with or without control components (CCs). The applicant requested authorization to store up to 28 damaged, or up to 8 failed fuel cans in the 32 PT DSC. The system is designed for a maximum heat load of 24 kW per canister and maximum 2.2 kW per assembly. The proposed damaged fuel canisters would have top and bottom end caps to assure retrievability of the fuels. The proposed failed fuels would be stored at outside corner cells of the 32PT DSC in the failed fuel canisters. Each canister has a welded bottom closure and removable top closure and is built from metal sheet, to facilitate lifting. There are screens at the top and bottom to capture fuel debris from filling and drainage of water during loading operation.

Shielding Design Features and Design Criteria

The applicant requested to add a new HLZC #4 to support allowable contents with a maximum decay heat of 2.2 kW per assembly and a maximum heat load of 24 kW per canister with a 2-year cooled fuel. The applicant also requested to increase the maximum assembly average burnup from 55 GWd/MTU to 62GWd/MTU, and the loading of damaged fuel assemblies confined within top and bottom end caps and failed fuel assemblies loaded within individual failed fuel cans. Additionally, the applicant requested authorization of cladding materials such as ZIRLO and M5. Since the density and thickness of alloy cladding materials such as ZIRLO and MS are not different from what used in shielding calculations, therefore, this change doesn't have any effect on the shielding feature of the 32PT cask. These proposed changes do not result in any change to the structural design of the canister or any changes to basket geometry.

The heat load configurations are shown in Figure M.2-1 through Figure M2-3a of the SAR.

Source terms

The source specification for the 32PT DSC is presented in Section M.5.2 of SAR Appendix M. Source terms are calculated for each burnup/enrichment combination and are listed in the Section M.5 of the SAR. The radiation source is modeled as an explicit basket with smeared fuel compositions within the basket cells. Conservative material compositions and axial peaking factors are applied in the modeling of the source term. The gamma and neutron source term calculations were performed with the ORIGEN-ARP modules of the SCALE 6.0/ENDFB-VI computer code.

The fuel types considered in this application are listed in Table M.2-2. The B&W 15x15 assembly type was chosen as the design basis fuel assembly because it has the highest initial heavy metal loading. The proposed new FQTs were generated to cover a larger range of burnup, enrichments, and heavy metal loading using the ORIGEN-ARP module of Scale 6.0. These FQTs are shown in the Technical Specifications Tables 1-3a through 1-3p.

Source Term Confirmatory Analyses

The staff performed confirmatory source term evaluations using the SCALE 6.1 computer code with the ORIGEN-ARP isotopic depletion and decay sequence with the 44-GROUP ENDF/B-VI cross section library. The staff used the same irradiation parameter assumptions used in the applicant's evaluation and obtained calculated source terms that were similar to those determined by the applicant. Therefore, the staff finds the applicant's source term evaluation acceptable.

For the proposed changes, the HLZC #4 with 2.2 kW corner fuel assemblies is determined to be the bounding configuration for dose rate analysis since the higher decay heat correlated to higher gamma source.

Dose Rate Scaling Factor Determination for Standardized NUHOMS® 32PT DSC

Dose rate calculations were performed using 3-D MCNP5 models for all bounding external dose rate calculations. The MCNP code is a standard in the nuclear industry for performing neutron and photon shielding analyses. Dose rate calculations using design basis source terms from Section M.5.2.7 of the SAR were performed to determine the contributions from the bottom, in core, plenum and top regions, as appropriate, from 32 PWR fuel assemblies and at various locations on and around the TC and HSM.

The dose rates for the 32PT/OS197TC and 32PT/HSM were also evaluated using MCNP 5 model and shown in Tables M5-3, M5-4, M5-5, and M5-23 of the SAR. The applicant used the 1-D ANISN code to determine the HSM and TC dose rates for each burnup, enrichment, cooling time (BECT) combination given in the FQTs.

The applicant used the ANSI/ANS Standard 6.1.1-1977 flux-to-dose rate conversion factors in the MCNP models to calculate dose rates, which is consistent with the SRP (NUREG 1536). Additional simplifications and bounding assumptions that reduce the amount of actual shielding were used in the shielding model and are discussed in the SAR. The staff finds that these assumptions add extra conservatisms in the shielding calculations and are acceptable.

The applicant used the response function for the three-zone 32PTH1 DSC transfer cask in Table U.5-15 of the previous SAR in calculating the dose rate of HLZC #4. This is conservative because HLZC #4 is a more heterogeneous loading configuration in comparison to the previously bounding loading pattern HLZC #2, and the response function for HLZC #4 bounds the HLZC # 2 because the hotter fuels in the inner zones are shielded more effectively by the fuels in the outer zone. This methodology is the same as used in previous amendments.

An updated source term for 0.490 MTU loading was run using MCNP 5 to determine dose rates at different positions along the TC and HSM-H for normal and hypothetical accident conditions. The same model was re-run for 0.380 MTU loading to determine the effect on the dose rates. Similar to the discussion above, the applicant used a comparison of the dose rates results for the 0.490 and 0.380 MTU loadings to determine scaling factors to be applied to the dose rates.

The results are shown in Tables U.5-28 through U.5-33 with maximum scaling factors determined to be for the 0.380 MTU/FA loading.

The dose rates in the Tables M.5.3, M.5.4, and M.5.23 for 32PT TC and 32 PT (HSM) for 0.491 MTU /fuel assembly are multiplied by higher scaling factors to obtain dose rates for the HLZC #4.

The scaling factors were applied to previous DORT analysis results to generate dose rates for the bounding HLZC #4 and the FQTs for HLZC #4 are shown in the Technical Specifications Tables 1-3a through 1-3p. The effect of uranium loading (380 kgU per assembly) on the shielding analysis and revised dose rates are summarized in Section M.5.4.16 of the SAR. The heat zone in the upper hemisphere of HLZC #4 is higher than the bottom hemisphere due to asymmetric loading in HLZC #4. Only the upper quadrant of this heal zone is modeled in MCNP with reflective boundaries on the west and south surface. This results in heat load of 30.4 KW which is much higher than 24 kW of the HLZC #4. The dose rates were scaled up using the methodology above to consider the heat zone HLZC #4, source term evaluated in Section M.5.2.7 of the SAR and 0.381 MTU fuel with compare with 0.475 MTU fuel assemblies used in the DORT analysis.

Confirmatory Calculations

The staff performed independent confirmatory analyses of selected dose rates using the MCNP 6.0/ ENDF/B-VI code system. The staff based its evaluation on the design features and model specifications presented in the drawings shown in SAR Appendix M and the limiting fuel characteristics, and the burnup and cooling time as included in the Technical Specifications. The staff's calculated dose rates were in reasonable agreement with the SAR values or were generally lower due to the applicant's conservative loading assumptions. The staff found that the SAR has adequately demonstrated that the 32PT DSC is designed to meet the criteria of 10 CFR 72.104(a) and 72.106.

Standardized NUHOMS® 24PTH DSC

The 24PTH DSC is authorized to safely store 24 intact standard PWR fuel assemblies with or without CCs. The 24PTH DSC consists of a cylindrical shell, canister bottom and top cover plates and shield plugs or shield plug assemblies and a basket assembly. The B&W 15x15 assembly types was selected as the bounding fuel assembly for shielding because it has the highest initial heavy metal loading (0.492 MTU) and Co-60 content as approved in the previous amendments. Fuel assemblies can be stored in five alternate heat zoning configurations in the 24PTH DSCs as shown in Figure P.2-1 through Figure P.2-4 and Figure P.2-9 of the SAR.

Shielding Design Features and Design Criteria

The applicant requested to add a new HLZC #6 to the allowable contents to allow loading of fuel assemblies with decay heat up to 2.5 kW, 2 years cooling and a total heat load of 35kW per basket. The applicant also requested authorization for the use of Transfer Cask OS197 for the transfer of the 24PTH-S-LC DSC in addition to the standardized TC of the 24PTH DSC. These changes do not result in any revisions to the shielding design features and design criteria.

Source terms

The SAR list the calculated source term for each burnup/enrichment combination. The proposed new FQTs were generated over a larger range of burnup, enrichments, and heavy metal loading using ORIGEN-ARP module of Scale 6.0. These FQTs are shown in the Technical Specifications Tables 1-3a through 1-3p. The HLZC #2 has larger heat load and hottest fuel assemblies in the peripheral zone, but the proposed new HLZC #6 has the hottest fuel assemblies (2.5kW). The HLZC #2 is the bounding configuration for the current heat zone 1, 3, 4, and 5. Therefore, the applicant developed source terms for both HLZC #2 and #6, zones. The ANISN transfer cask and HSM storage response functions in the Section P.5.2 of the SAR and as described above were used to evaluate the source terms for each FQT dose combination burnup, enrichment, and cooling time (BECT). The BECT that resulted in a higher source term was selected as the bounding design basis source term.

Dose Rate Scaling Factor Determination for Standardized NUHOMS® 24PTH DSC.

The methodology used to determine the dose rate scaling factor is the same as described above for the 32PT DSC. The applicant performed MCNP 5 model using updated source term and updated material specifications to show the effect of a reduction in fuel loading (0.380 MTU) on the dose rate for the bottom, in core, plenum, and top region at various location on and around the 24PTH cask in the HSM and TC. The dose rate results of these evaluations were compared with previous dose rate results for the loading of 0.490 MTU per assembly. Six scaling factors were derived based on the updated results using the same methodology described above for 32PT DSC. The dose rates for 0.380 MTU loading are obtained from the dose rates of the 0.490 MTU loading fuel assemblies by multiplying using the highest scaling factors. This provides more conservative results than the MCNP5 Model for 0.380 MTU per assembly. The scaling factors are shown in the footnote of Table P.5-1 through Table P.5-5, Table P.5-21, Table P.5-22, Table P.5-24, and Table P.5-26 for dose rate results.

Confirmatory Calculations

The staff performed independent confirmatory analyses of selected dose rates using the MCNP 6.0/ ENDF/B-VI code system for both HSM and OS197 transfer cask. The staff based its evaluation on the design features and model specifications presented in the drawings shown in SAR Appendix P. The fuel characteristics, burnups and cooling times, that are listed in the Technical Specifications were used by staff to determine the dose rates around the TC and the HSM. The staff's calculated dose rates were in reasonable agreement with the SAR values or were generally lower due to the applicant's conservative loading assumptions. The staff found that the applicant has adequately demonstrated reasonable assurance that an ISFSI loaded with the 24PTH Limiting fuel DSCs will meet the criteria of 10 CFR 72.104(a) and 72.106.

Standardized NUHOMS® 61BTH DSC

The 61BTH DSC consists of a shell assembly (cylindrical shell, canister bottom and top cover plates and shield plugs or shield plug assemblies) and a basket assembly that can be used to store up to 61 intact or, 61 BWR damaged fuel assemblies. The GE 7x7 fuel assemblies were approved in the previous amendments as bounding conditions. The 61BTH has two types (Type 1 and Type 2). The maximum heat load for Type 1 is 22.0 kW. The Type 2 DSC incorporates the fixed top grid assembly design in lieu of the top hold down ring and accommodate the higher DSC heat loads of up to 31.2 kW.

Shielding Design Features and Design Criteria

The applicant requested to add an alternative loading option to HLZC #10, and allow loading of fuel assemblies with decay heat up to 1.2 kW, 2 years cooling in the zone 3 of the previous configuration in the type 2 DSC. The total heat load is the same as in the previous loading configuration. In addition the applicant requested adding two new fuel assemblies for the GNF-2 and ATRIUM-11 FA designs as authorized contents. These changes do not result in any revisions to the shielding design features and design structural.

Source terms

The source terms for the shielding analysis are calculated using design basis heavy metal loadings of 0.198 MTU per assembly as previous amendments. However, the FQTs are developed for both heavy metal loadings between 0.170 MTU to 0.198 MTU. The results are shown in Section T.5.2 and Table T.5-30 of the SAR.

The source term for NUHOMS® 61BTH DSC is determined using the geometry and material from Section T.5.2 of the SAR. The bounding source is based on the GE 7x7 with maximum Co-59 content for each hardware regions.

Dose Rate Determination for Standardized NUHOMS® 61BTH DSC

The shielding analysis results are shown in Table T.5-4 of the SAR which shows that more than 60% of the dose rates on the transfer cask under normal conditions is due to contributions from gamma radiation and the dose rates near the horizontal storage module (HSM) is due to gamma radiation. The cobalt content in the hardware of the generic fuel design is about 16 grams per assembly. The HLZC #10 is the same as previously analyzed with the only difference being the loading of contents with decay heat up to 1.2 kW, 2 years cooling in HLZC #3 in the Type 2 DSC.

The revised HLZC #10 in HSM is bounded by HLZC #5 and the OS197 transfer cask is bounded by HLZC #3 as described in the previous amendments. The GNF-2 and ATRIUM 11 are newer assembly designs and have lower cobalt content of about 8 gram per assembly. Therefore the source term and dose rates are bounded by the generic BWR fuel design.

Confirmatory Calculations

The staff performed independent confirmatory analyses of selected dose rates for the HSM and transfer cask using the MCNP 6.0/ ENDF/B-VI code system and for source term calculation using the ORGEN-ARP of the SCALE 6.1. The staff based its evaluation on the design features and model specifications presented in the drawings shown in SAR Appendix T. Limiting fuel characteristics, burnups and cooling times that are listed in the Technical Specifications were used by the staff to determine the dose rates around the TC and the HSM. The staff's calculated dose rates were in reasonable agreement with the applicant's values or were generally lower due to the applicant's conservative loading assumptions. The staff found that the applicant adequately demonstrated reasonable assurance that an ISFSI using the 61BTH DSC is capable to meet the criteria of 10 CFR 72.104(a) and 72.106.

Standardized NUHOMS® 32PTH1 DSC

The 32PTH1 DSC is designed to safely store a maximum of 32 intact PWR assemblies, or up to 16 damaged and 16 intact assemblies (for a total of 32 assemblies), with an initial enrichment of up to 5.0 weight percent. Allowable fuel types evaluated include the WE 17x17, CE 16x16, BW 15x15, CE 15x15, WE 15x15, CE 14x14, and WE 14x14 as well as accompanying Control Components. The most radioactive intact fuel design evaluated is the B&W 15x15 Mark B-10 assembly because this assembly contains the maximum heavy metal among the other fuel designs. For the dose rate calculation, the B&W 15x15 fuel assemblies was used for bounding conditions consistent with previously approved amendments

Shielding Design Features and Design Criteria

The applicant requested to add a new HLZC #5 to allow loading of fuel assemblies with decay heat up to 1.1 kW for a total heat load of 35.2 kW per basket, and HLZC #6 to allow for loading of FAs with decay heat up to 1.3 kW for a total heat load of 37.6 kW per basket. These changes do not result in any revisions to the shielding design features and design structure.

Source terms

The source terms for each burnup/enrichment and cooling time combination and are listed in Section U.5.2.6 of the SAR. MCNP5 models for 32PTH DSC, HSM-H and OS200 input files were rerun using SCALE6. ORIGEN-ARP design basis source terms to determine the impact on the dose rates. The burnup, enrichment and cooling time that results in the higher source term was selected as the bounding design basis source term.

The new FQTs were generated over a larger range of burnup, enrichments, and heavy metal loading using ORIGEN-ARP module of Scale 6.0. These FQTs shown in the Technical Specifications Tables 1-3a through 1-3p. The HLZCs are shown in Figures U.2-1 through U.2-3 and U.2-5 through U.2-7. The applicant modeled the four center compartments with a decay heat of 0.8 kW and all other compartments as 1.3 KW decay heat, with total decay heat of 42.8 kW. This bounds the decay heat of 40.8 kW for the DSC. The original 490 kg U/FA FQTs for the 32PTH1 DSC have been replaced with FQTs for 380 kgU, 475 kgU, and 492 kg U/FA developed using source terms developed by SCALE6.0/ORIGEN-ARP. These FQTs are documented in Section U5.2.6.

Dose Rate Scaling Factor Determination for Standardized NUHOMS® 32PTH1 PTH DSC.

The methodology used is the same as described above for the 32PTCask. The applicant used MCNP 5 calculation with updated source term and material specification to show the effect of reduction in uranium loading on dose rate for the bottom, in core, plenum, and top region at various locations on and around the 32PTH1 cask in the HSM and TC. The results were compared with results of the 0.490 MTU dose rates to determine the scaling factors. The scaling factors are shown in the footnote of Table U.5-1 through Table U.5-5, Table U.5-21, Table U.5-22, Table U.5-24, and Table U.5-26.

Confirmatory Calculations

The staff performed independent confirmatory analyses of selected dose rates for HSM and transfer cask using the MCNP 6.0/ ENDF/B-VI code system and ORGEN-ARP of the SCALE 6.1 for source term calculation. The staff based its evaluation on the design features and model specifications presented in the drawings shown in SAR Appendix U. The staff's calculated dose rates were in reasonable agreement with the applicant's values or were generally lower due to the applicant's conservative loading assumptions. The staff found that the Standardized NUHOMS® System design with the proposed amendment provides reasonable assurance the DSC will continue to meet the regulatory criteria of 10 CFR 72.104(a) and 72.106.

6.1 Findings

F6.1 Section 5 of the SAR, sufficiently describes shielding SSCs important to safety in sufficient detail to allow evaluation of their effectiveness.

F6.2 Radiation shielding is sufficient to meet the radiation protection requirements of 10 CFR Part 20, 10 CFR 72.104, and 10 CFR 72.106.

F6.3 The design of the radiation protection system of the 32PTH, 24PTH, 61BTH, and 32PTH1 DSCs, when used with the appropriate HSM and TC, are in compliance with 10 CFR Part 72, and that the applicable design and meets the acceptance criteria as prescribed in SRP [Ref. 3]. The radiation protection system design provides a reasonable assurance that the 32PTH, 24PTH, 61BTH, and 32PTH1 DSCs will provide safe storage of spent fuel in an ISFSI. This finding is based on a review that considered the regulation itself, the appropriate regulatory guides, applicable codes and standards, the applicant's analyses, the staff's confirmatory analyses, and acceptable engineering practices.

6.2 References

U.S. *Code of Federal Regulations*, Standards for Protection against Radiation, Title 10, Part 20.

U.S. *Code of Federal Regulations*, Licensing Requirements for the Independent Storage of Spent Nuclear Fuel, High-Level Radioactive Waste, and Reactor - Related Greater Than Class C Waste, Title 10, Part 72.

U.S. Nuclear Regulatory Commission, Standard Review Plan for Dry Cask Storage Systems at a General Licensee Facility, NUREG-1536, July 2010.

7.0 CRITICALITY EVALUATION

The staff's objective in reviewing the applicant's criticality evaluation of Amendment No. 15 to the Standardized NUHOMS® System design is to verify that the spent fuel contents remain subcritical under the normal, off-normal, and accident conditions of handling, packaging, transfer, and storage. The applicable regulatory requirements are those in 10 CFR 72.24(c)(3), 72.24(d), 72.124, 72.236(c), and 72.236(g).

The staff reviewed the information provided in the amendment request to determine whether the Standardized NUHOMS® System continues to fulfill the acceptance criteria listed in Section 7 of NUREG 1536, "Standard Review Plan for Dry Cask Storage Systems at a General License Facility."

7.1 Criticality Design Criteria and Features

The applicant provided a summary of the proposed changes to the Standardized NUHOMS® System in Enclosure 2 to the letter dated March 8, 2017 (Areva Document E-46881, "Application for Amendment 15 to Standardized NUHOMS® Certificate of Compliance No. 1004 for Spent Fuel Storage Casks, Revision 0"). This section of the safety evaluation report discusses the proposed changes which affect criticality safety of the storage system. These proposed changes are as follows:

1. Change #4, Revision to UFSAR Appendix M for the 32PT DSC to store up to 28 damaged CE 14x14, WE 14x14, and WE 17x17 or CE 15x15 class fuel assemblies in cells confined with top and bottom end caps in the Type A1 or A2 basket.
2. Change #4, Revision to UFSAR Appendix M for the 32PT DSC to store up to eight (8) failed fuel assemblies in failed fuel canisters (FFCs) in the Type A1 or A2 basket.
3. Change #4, Revision to UFSAR Appendix M for the 32PT DSC to add a basket option with 32 poison plates, and corresponding increased assembly average initial enrichments for intact WE 17x17 and CE 15x15 class spent fuel assemblies.
4. Change #4, Revision to UFSAR Appendix M for the 32PT DSC to include a higher enrichment limit and soluble boron loading requirement option for intact WE 17x17 class assemblies stored with boron carbide (B_4C) poison rod assemblies (PRAs) in Type B and C baskets, with Type 1 or 2 poison plates;
5. Change #4, Revision to UFSAR Appendix M for the 32PT DSC to provide an option to use silver-indium-cadmium (AIC) neutron absorber material as a substitute for B_4C PRAs in several configurations; and

6. Change #8, Revision to UFSAR Appendix T for the 61BTH DSC to allow two additional boiling water reactor (BWR) fuel types: GNF2 and ATRIUM-11.

7.2 Fuel Specification

The applicant stated that none of the proposed changes for the 32PT DSC that affect criticality safety involved a change in the fuel types allowed for storage. The changes involved either adding a new basket poison plate configuration for existing pressurized water reactor (PWR) fuel types, changing the number and configuration of damaged or failed fuel assemblies allowed to be stored in the 32PT DSC, or adding criticality control components similar to those previously approved for this and other DSCs. The applicant revised the definitions of intact, damaged, and failed fuel (see Standardized NUHOMS® System Technical Specifications Table 1-1e), and concluded that these revisions do not affect criticality safety of the 32PT DSC, as the damaged and failed fuel configurations assumed in the applicant's criticality analysis bound the revised definitions.

The applicant revised Appendix M of the Standardized NUHOMS® System UFSAR to add a basket option with 32 poison plates, with two different poison plate ^{10}B areal densities (Type A1-32 and Type A2-32), and corresponding increased assembly average initial enrichments for WE 17x17 and CE 15x15 class intact spent fuel assemblies. Table 1-1g1 of the Standardized NUHOMS® System Technical Specifications provides enrichment limits for intact WE 17x17 class fuel assemblies, with and without control components, and intact CE 15x15 class fuel assemblies, without control components, for two soluble boron loadings (2500 and 2800 ppm).

The applicant also revised Appendix M of the Standardized NUHOMS® System UFSAR to add a higher enrichment and soluble boron loading (2800 ppm) option for Type B1, B2, and C1 baskets, for WE 17x17 class intact spent fuel assemblies without control components. The applicant also revised this appendix to provide an option to use silver-indium-cadmium (AIC) neutron absorber in PRAs in the Type B1-r, B2-r, C1-r, or C2-r baskets, for WE 17x17 class intact spent fuel assemblies without control components. Table 1-1g1 of the Standardized NUHOMS® System Technical Specifications provides enrichment limits for intact WE 17x17 class fuel assemblies stored in: 1) the Type B1 or B2 baskets with four B_4C PRAs; 2) the Type C1 basket with eight B_4C PRAs; 3) the Type B1-r or B2-r baskets with four AIC PRAs; or 4) the Type C1-r or C2-r baskets with eight AIC PRAs. The applicant provides specifications for the B_4C and AIC PRAs in Figure M.1-2 of the UFSAR. Required PRA locations for the 32PT DSC configuration with four and eight PRAs are given in Standardized NUHOMS® System Technical Specifications Figures 1-5 and 1-6, respectively.

The applicant also revised Appendix M of the Standardized NUHOMS® System UFSAR to increase the allowable numbers of damaged fuel assemblies to: 1) 28 damaged WE 17x17 or CE 15x15 fuel assemblies in a Type A1 or A2 basket with 32 poison plates; 2) 28 damaged WE 17x17, CE 15x15, WE 14x14, or CE14x14 fuel assemblies in a Type A1 or A2 basket with 24 poison plates; and 3) 16 damaged WE 14x14 or CE14x14 fuel assemblies in a Type A1 or A2 basket with 24 poison plates. Table 1-1g2 of the Standardized NUHOMS® System Technical Specifications provides enrichment limits for each of the fuel and basket combinations described above, for soluble boron loadings between 1800 and 2800 ppm. Figure 1-4b of the Standardized NUHOMS® System Technical Specifications provides the required locations of damaged fuel assemblies for each maximum number of damaged fuel assemblies in the 32PT DSC.

The applicant also revised Appendix M of the Standardized NUHOMS® System UFSAR to include up to eight failed fuel assemblies in FFCs in Type A1 or A2 baskets with 24 poison plates, or in Type A1-32 or A2-32 baskets with 32 poison plates. The FFC configuration is shown in drawing number NUH-32PT-1007-SAR. Type A1 or A2 baskets may be used to store up to eight failed WE 17x17, WE 14x14, CE 14x14, or CE 15x15 class fuel assemblies, with the remainder intact. Type A1-32 or A2-32 baskets may be used to store up to eight failed WE 17x17 or CE 15x15 class fuel assemblies, with the remainder intact. Intact WE 17x17, WE 14x14, and CE 14x14 class fuel assemblies in these configurations may be stored with or without control components, while CE 15x15 class fuel assemblies must be stored without control components. Table 1-1g3 of the Standardized NUHOMS® System Technical Specifications provides enrichment limits for each of the fuel and basket combinations described above, for soluble boron loadings between 1800 and 2800 ppm. Figure 1-4b of the Standardized NUHOMS® System Technical Specifications provides the required locations of failed fuel assemblies in FFCs in the 32PT DSC.

The applicant revised the fuel specification for the 61BTH DSC to include two new boiling water reactor (BWR) fuel types: the GNF2 10x10 fuel assembly and the ATRIUM-11 11x11 fuel assembly. The GNF2 fuel assembly is similar to the previously approved GE12 and GE14 fuel assembly, and therefore the previously determined initial enrichment limits for intact, damaged, and failed fuel are the same for these fuel assemblies. The ATRIUM-11 fuel assembly is limited to the 61BTH Type 2F basket, with a maximum lattice average initial enrichment of 4.45 weight percent ^{235}U, and is limited to no more than four damaged fuel assemblies. Failed ATRIUM-11 fuel is not authorized for storage in the 61BTH DSC. Tables 1-1v, 1-1w, and 1-1w1 of the Standardized NUHOMS® System Technical Specifications provide the enrichment limits for intact, damaged, and failed fuel, respectively, in the 61BTH DSC.

7.3 Model Specification

7.3.1 Configuration

The applicant modeled the Standardized NUHOMS® System with the 32PT and 61BTH DSCs using the most reactive configuration of the DSC basket and fuel assemblies determined from analyses in amendments previously approved by the staff. For the basket geometry, the configuration consisted of the most reactive combination of basket guide tube wall and neutron absorber plate thicknesses, and eccentric positioning of fuel assemblies. For the fuel assemblies, the configuration consisted of the most reactive combination of tolerances for the pellet diameter, outer clad diameter, and clad thickness, as well as the maximum active fuel length (or infinite length, in cases where this was modeled). The applicant modeled the fuel material as UO_2 at a stack density of at least 96% theoretical density, without allowance for dishing or chamfer, which conservatively increases the fuel material in the assembly. Additionally, the applicant conservatively assumes that the gap is filled with full density moderator.

For damaged fuel assemblies, the applicant assumes that the pitch can vary to optimum, and that rods may be removed to achieve an optimum moderated condition. For failed fuel assemblies, the pitch for each lattice evaluated is allowed to vary to find the most reactive value. Additionally, failed rods are conservatively modeled without cladding.

For all new configurations in the 32PT and 61BTH DSCs, the applicant also varied the internal moderator density, in order to find the most reactive condition. While this water density is always 100% for the 61BTH with fresh water in the basket, the most reactive density varies for each 32PT configuration, depending on the configuration and the soluble boron concentration in the moderator. This analysis of moderator density captures varying densities which may occur in the DSC during draining and drying operations.

The staff concludes that the configurations of the 32PT and 61BTH DSCs analyzed in the criticality analysis are consistent with previously approved analyses of the Standardized NUHOMS® System and similar spent fuel storage systems. The staff finds that the applicant has determined the most reactive configuration of each DSC basket and fuel type.

7.3.2 Material Properties

The 32PT and 61BTH DSC fuel and basket materials have not changed from the previously approved design, and are described in the associated appendix for each DSC. These descriptions include the composition of the major components of each DSC, including UO_2 fuel, steel and aluminum structural components, and neutron absorber panels and poison rod assemblies. The criticality analyses conservatively assume no more than 75% of the neutron absorber manufacturer's minimum specified ^{10}B content for Boral® panels, as well as for B_4C poison rod assemblies. For borated aluminum and aluminum/B_4C metal matrix composite neutron absorbers, the criticality analyses assume no more than 90% of the minimum specified ^{10}B content, due to the more robust material verification requirements for these materials.

The applicant also provided materials specifications for the AIC PRA neutron absorber material, used in Types B1-r, B2-r, C1-r, and C2-r baskets of the 32PT DSC with WE17x17 class intact fuel assemblies. The AIC PRA consists of an alloy of 15% indium, 5% cadmium, and 80% silver. None of the indium and cadmium, and less than 50% of the silver expected to be present (based on the nominal density of AIC material and the required silver content per unit length of absorber rod) is credited in the criticality analysis. Additionally, the silver content is reduced to 75% of its required value in the 32PT DSC criticality analysis, consistent with the recommendation for neutron absorber credit in NUREG-1536.

The staff concludes that the materials properties assumed in the 32PT and 61BTH DSC criticality analyses are consistent with those assumed in previously approved analyses of the Standardized NUHOMS® System and similar spent fuel storage systems. The staff finds that the applicant has determined conservative materials properties for each DSC basket and fuel type.

7.4 Criticality Analysis

The applicant performed a criticality analysis for the requested new configurations using the criticality model and material properties described above. The resulting maximum initial enrichments, minimum required soluble boron for loading and unloading, and associated k_{eff} values are reported in Appendixes M and T of the UFSAR, for the 32PT, and 61BTH DSCs, respectively.

For the 32PT DSC, the applicant calculated the maximum system k_{eff} for intact WE 17x17 class fuel assemblies in Type A1-32 and A2-32 baskets with no PRAs and 32 poison plates, with 2500 and 2800 ppm minimum required soluble boron loadings in the water. The applicant varied the internal moderator density to find the optimum to produce the highest k_{eff}, for fuel assemblies with and without control components. The k_{eff} results for this analysis are reported in Table M.6-63 of the UFSAR. The applicant performed similar analyses for CE 15x15 class intact fuel assemblies, the results of which are reported in Table M.6-64 of the UFSAR.

Also for the 32PT DSC, the applicant calculated the maximum system k_{eff} for intact WE 17x17 class fuel assemblies in Type B1, B2, B1-r, B2-r, C1, C1-r, and C2-r baskets. The applicant performed all of these evaluations with the required 2800 ppm soluble boron concentration in the moderator, and without control components in the guide tubes of each assembly. The applicant modeled the Type B1 and B2 configurations with the required four B_4C PRAs, and the C1 configuration with eight B_4C PRAs. The applicant modeled the B1-r and B2-r configurations with the required four AIC PRAs, and the C1-r and C2-r configurations with eight AIC PRAs. The applicant varied the internal moderator density to find the optimum to produce the highest system k_{eff}. The k_{eff} results for this analysis are reported in Table M.6-63 of the UFSAR.

Additionally for the 32PT DSC, the applicant calculated the maximum system k_{eff} for several new damaged fuel configurations. For WE 17x17 and CE 15x15 fuel classes, the applicant evaluated 32 damaged fuel assemblies in the Type A1 and A2 basket, at 2500 and 2800 ppm soluble boron loading in the moderator. Additionally, the applicant evaluated 32 damaged WE 17x17 and CE 15x15 class fuel assemblies in the Type A1-32 and A2-32 baskets, at 2500 and 2800 soluble boron loading in the moderator. Although the applicant demonstrates that the 32 damaged fuel assembly configurations described above are subcritical, the applicant conservatively limits the 32PT DSC to 28 damaged fuel assemblies, with the center four assemblies required to be intact. This is conservative, as intact assemblies consistently produce lower system reactivities than optimally reconfigured damaged fuel assemblies. The applicant varied the internal moderator density for each of these configurations to find the optimum which produces the highest system k_{eff}. The k_{eff} results for this analysis are reported in Table M.6-68 of the UFSAR.

Similar to the above analysis, the applicant calculated the maximum system k_{eff} for WE 14x14 and CE 14x14 fuel assembly classes, for 16 or 28 damaged fuel assemblies in Type A1 and A2 baskets with 24 poison plates. The locations of damaged fuel assemblies in the 32PT DSC are as given in Figure 1-4b of the Standardized NUHOMS® System Technical Specifications. The applicant performed these evaluations at five different soluble boron moderator loadings: 1800, 2100, 2300, 2500, and 2600 ppm. The applicant varied the internal moderator density for each of these configurations to find the optimum which produces the highest system k_{eff}. The k_{eff} results for this analysis are reported in Tables M.6-69 and M.6-70 of the UFSAR.

For failed fuel in the 32PT DSC, the applicant evaluated WE 17x17 class fuel assemblies, with and without control components, and CE 15x15 class fuel assemblies, without control components, with failed fuel in FFCs in eight corner locations (defined in Figure 1-4b of the Standardized NUHOMS® System Technical Specifications). The applicant performed these evaluations in the Type A1 and A2 baskets with 24 poison plates, and in the Type A1-32 and A2-32 baskets, with 32 poison plates. Inside the FFC, the applicant modeled several different array sizes of unclad fuel rods, to find the optimum pitch and number of rods. In these evaluations, the applicant considered soluble boron loadings of 2500 and 2800 ppm. The applicant varied the internal moderator density for each of these configurations to find the optimum which produces the highest system k_{eff}. The k_{eff} results for this analysis are reported in Table M.6-71 of the UFSAR.

Similar to the above analysis, the applicant evaluated WE 14x14 and CE 14x14 class fuel assemblies, with and without control components, with failed fuel in FFCs in eight corner locations (defined in Figure 1-4b of the Standardized NUHOMS® System Technical Specifications). The applicant performed these evaluations in the Type A1 and A2 baskets with 24 poison plates. For WE 14x14 class fuel assemblies, the applicant considered four soluble boron loadings in the moderator: 1800, 2100, 2300, and 2500 ppm. For CE 14x14 class fuel assemblies, the applicant considered a single soluble boron loading of 2600 ppm. Inside the FFC, the applicant modeled the optimum pitch and number of unclad fuel rods to produce the highest system k_{eff}. The applicant varied the internal moderator density for each of these configurations to find the optimum which produces the highest system k_{eff}. The k_{eff} results for this analysis are reported in Tables M.6-72 and M.6-73 of the UFSAR.

For the analysis of GNF2 BWR fuel in the 61BTH DSC, the applicant performed a comparison of system reactivity with GNF2 fuel to that with GE12 fuel, determined to be the bounding fuel type for enrichment limit determination in an analysis supporting a previously approved amendment to the Standardized NUHOMS® System. The applicant performed criticality analyses of the 61BTH DSC with GNF2 fuel, in the Type 1 and 2 DSC with A, C, and F poison loadings, for both the damaged and failed fuel assembly configurations. The results of these analyses demonstrate that k_{eff} of the 61BTH DSC with GNF2 fuel is statistically the same as that with GE12 fuel. Therefore, the applicant determined that the enrichment limits previously determined using the bounding GE12 fuel design are applicable to the new GNF2 fuel type. The staff finds the applicant's conclusion acceptable, since the k_{eff} results for the GNF2 fuel type are statistically the same (i.e., within the calculation Monte Carlo uncertainty) as those for the GE12, and since the GNF2 fuel assembly is structurally and materially similar (i.e., same fuel material, same pitch, similar pellet diameter) to the GE12 fuel assembly. The k_{eff} results for the GNF2 analysis are reported in Tables T.6-26 and T.6-27 of the UFSAR for intact fuel, Tables T.6-29 and T.6-30 of the UFSAR for up to 4 damaged fuel assemblies, Tables T.6-32 and T.6-33 of the UFSAR for up to 16 damaged fuel assemblies, Table T.6-34 of the UFSAR for up to 4 failed fuel assemblies, and Table T.6-35 for 4 failed and 12 damaged fuel assemblies.

The applicant performed an analysis of ATRIUM-11 fuel in the 61BTH Type 2 DSC with F poison loading. The applicant evaluated both the DSC with only intact fuel assemblies, as well as the configuration with up to four damaged fuel assemblies. From the results of this analysis, the applicant concluded that the ATRIUM-11 fuel assembly can be stored in the Type 2 61BTH DSC with F poison loading, provided the maximum lattice average maximum lattice average initial enrichment is limited to 4.45 weight percent ^{235}U. The k_{eff} results for this analysis are reported in Tables T.6-28 and T.6-31 of the UFSAR.

7.4.1 Computer Programs

The applicant used the CSAS5 sequence of the SCALE 6.0 code system with the KENO V.a three-dimensional Monte Carlo neutron transport program and the 44-group ENDF/B-V cross section library for all k_{eff} calculations for this amendment. The SCALE code system is a standard in the nuclear industry for performing Monte Carlo criticality safety and radiation shielding calculations.

The staff performed confirmatory calculations using the CSAS5 sequence of the SCALE 6.2 code system, with the KENO V.a three-dimensional Monte Carlo neutron transport program and the continuous-energy ENDF/B-VII.1 cross section library.

7.4.2 Multiplication Factor

The applicant demonstrated that k_{eff} values for the additional storage configurations for the 32PT and 61BTH DSCs are all below the Upper Subcritical Limits (USL) calculated for each DSC in the benchmarking analysis for the SCALE 6.0 code and 44-group ENDF/B-V cross section library used in the criticality analysis. Therefore, the Standardized NUHOMS® System with the 32PT and 61BTH DSCs will remain subcritical under normal, off-normal, and accident conditions, meeting the criticality safety requirements of 10 CFR 71.124.

The staff performed confirmatory criticality evaluations of the Standardized NUHOMS® System with additional fuel configurations in the 32PT and 61BTH DSCs. Using assumptions similar to the applicant's, the staff calculated k_{eff} values for select configurations which were within the margin of error of those calculated by the applicant, and confirmed that the storage system will meet the criticality safety requirements of 10 CFR Part 72.

7.4.3 Benchmark Comparisons

As the applicant's requested revisions to the 32PT involve the same fuel types and similar basket, poison (with the exception of AIC PRAs), and fuel configurations, the applicant determined that the previously approved benchmarking analysis performed for the code and cross section data used in this amendment is still applicable. There are no significant deviations from the type or concentrations of fuel, moderator, and absorber material used in previously approved evaluations, and all of the parameters of interest to the criticality calculation remain within the area of applicability of the previous benchmarking analysis. Therefore, the staff finds that, for storage configurations that do not include AIC PRAs, the previously approved USL for the 32PT DSC modeled with SCALE 6.0 and the 44-group ENDF/B-V cross section library is appropriate.

For 32PT DSC configurations with AIC PRAs, the applicant credits some of the silver in the absorber, which is significantly different from the boron-based neutron absorbers considered in the applicant's previously approved benchmarking analysis. The previous benchmarking analysis for SCALE 6.0 and the 44-group ENDF/B-V cross section library did not include experiments containing silver as a neutron absorber, and the applicant did not modify the benchmark analysis to include such experiments. However, since the applicant ignores the significant neutron absorbing isotopes of indium and cadmium present in the absorber material, and credits less than 50% of the silver present in the AIC material (with 75% of that amount credited in the criticality analysis), the staff finds the existing benchmarking analysis for the 32PT DSC to be acceptable.

For the 61BTH DSC, the applicant's requested revisions involve the same fuel types and similar basket, poison, and fuel configurations. Therefore, the applicant determined that the previously approved benchmarking analysis performed for the code and cross section data used in this amendment is still applicable. The staff concludes that there are no significant deviations from the type or concentrations of fuel, moderator, and absorber material used in previously approved evaluations, and all of the parameters of interest to the criticality calculation remain within the area of applicability of the previous benchmarking analysis. Therefore, the staff finds that the previously approved USL for the 61BTH DSC modeled with SCALE 6.0 and the 44-group ENDF/B-V cross section library is appropriate.

7.5 Findings

F7.1 Structures, systems, and components important to criticality safety are described in sufficient detail in Chapters M.2, M.6, T.2, and T.6 of the UFSAR 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, fixed neutron poisons, and soluble poisons of the spent fuel pool. An appraisal of the fixed neutron poisons has shown that they will remain effective for the term requested in the CoC application and there is no credible way for the fixed neutron poisons to significantly degrade during the requested term in the CoC application; 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 the CoC application.

The staff concludes that the criticality design features for the Standardized NUHOMS® System design 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 Standardized NUHOMS® System design will allow safe storage of spent fuel. 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.

8.0 MATERIALS EVALUATION

The staff reviewed the information provided by the applicant and evaluated the eleven changes requested in Amendment No. 15 as follows:

1. Change #1, unify and standardize the fuel qualification tables for four PWR systems in order to simplify the Technical Specifications.
2. Change #2, for the 32PT System, add a new HLZCs #4 to allow for the loading of FAs with decay heat up to 2.2 kW corresponding to a 2-year cooled fuel.
3. Change #3, for the 32PT System, increase the maximum assembly average BU from 55 Gwd/MTU to 62 Gwd/MTU.
4. Change #4, for the 32PT System, allow for the loading of damaged fuel assemblies confined within top and bottom end caps and failed fuel assemblies loaded within individual failed fuel canisters. Provide for a basket option to increase the number of poison plates from 24 to 32 resulting in an increase in the allowable enrichment of the authorized contents. Expand the definition of the PRAs to include RCCA materials, specifically silver neutron absorber.
5. Change #5, for the 32PT System, include other zirconium alloy cladding materials such as ZIRLO and M5.
6. Change #6, for the 24PTH System, add a new HLZC #6 to allow for the loading of FAs with decay heat up to 2.5 kW corresponding to a 2-year cooled fuel, and a total heat load of 35 kW per basket.
7. Change #7, for the 24PTH System, the OS197 is added as an authorized TC for the transfer of the 24PTH-S-LC DSC in addition to the standardized TC.
8. Change #8, for the 61BTH System, revise the existing HLZC #10 to allow loading of BWR FAs with decay heat up to 1.2 kW corresponding to a 2-year cooling time. GNF-2 and ATRIUM-11 FA designs are also added as authorized contents.
9. Change #9, for the 32PTH1 System, add new HLZCs #5 for the loading of FAs with decay heat up to 1.1 kW for a total heat load of 35.2 kW per basket and HLZC #6 to allow for loading of FAs with decay heat up to 1.3 kW for a total heat load of 37.6 kW per basket.
10. Change #10, provide a description in the UFSAR for the solar shield currently described in the Technical Specifications for the TC during transfer operations.
11. Change #11, Technical Specification 4.3.3 Item 11 is changed to add flexibility to general licensees in verifying compliance regarding the storage pad location and the soil-structure interaction.

The changes proposed in the application (CoC No. 1004 Amendment No. 15), including the new HLZCs for the 32PT and 24PTH, did not result in changes to the maximum temperatures and pressures for the Standardized NUHOMS® System. The operating environmental conditions are unchanged for the Standardized NUHOMS® System DSCs, TCs and horizontal storage modules (HSMs).

As noted in proposed change #4, the definition of the PRAs is modified to include silver neutron absorber. As noted in proposed change #5, the allowed contents of the 32PT System are modified to include fuels with cladding materials other than Zircaloy including ZIRLO and M5. As noted in proposed change #8, GNF-2 and ATRIUM-11 FA designs are also added as authorized contents. Other than these changes, the allowable contents for the Standardized NUHOMS® System are unchanged.

The applicant provided the updated technical specifications and an UFSAR to support the proposed Amendment No. 15 changes. The applicant also provided an evaluation of the impact of these structures, systems, and components included in the application on CoC No. 1004 renewal.

The staff reviewed and evaluated the application using the guidance in Chapter 8 of NUREG-1536, Revision 1, to reach reasonable assurance of adequate materials performance under normal, off-normal, and accident-level conditions. Staff review of the application identified a limited number of changes associated with the materials evaluation areas listed in NUREG-1536, Revision, 1 Section 8.2 as follows:

1. General
    a. Cask Design/Materials
    b. Environmental Conditions
    c. Engineering Drawings
2. Materials Selection
    a. Applicable Codes and Standards and Alternatives to the Code
    b. Neutron Poison Materials for Criticality Control
    c. Mechanical Properties and Creep Analyses
3. Corrosion
    a. Galvanic/Chemical/Radiolytic Reactions of Fuel with Canister Internals
4. Cladding Integrity/Fuel
    a. Fuel Burn-up
    b. Cladding Temperature Limits
    c. Damaged Fuel Definition

The proposed changes in the application did not result in changes that required materials evaluation for operational Issues, examination and testing, or code case acceptability. As such, these areas are not included in the materials review of the application.

In addition to the materials evaluation areas listed above, the staff evaluated the impact of these structures, systems, and components included in the application on the CoC No. 1004 renewal. As stated in NUREG-1927, Revision 1, amendment applications submitted after a CoC has been renewed should either (1) show that SSCs described in the amendment are already encompassed in the aging management activities associated with the CoC renewal or (2) revise or propose new aging management activities for new SSCs proposed in the amendment. The staff review of the effect of the amendment on the CoC renewal is provided in the subsequent sections of this chapter.

8.1 Cask Design/Materials, Engineering Drawings and Applicable Codes and Standards

The applicant description of the proposed changes includes the addition of the loading of damaged fuel assemblies confined within top and bottom end caps and failed fuel assemblies loaded within individual failed fuel canisters (FFC) in the 32PT DSC. The applicant provided engineering drawings for the damaged fuel endcaps and the FFCs. The top and bottom end caps are provided with screens at the bottom and top to contain fuel debris and allow filling/drainage of water during loading operations. The FFCs are constructed of sheet metal and provided with a welded bottom closure and a removable top closure, which allows lifting of the FFC with the enclosed damaged assembly/debris. The FFC is provided with screens at the bottom and top to contain fuel debris and allow filling/drainage. No other design or material changes are included in the application.

The staff reviewed the proposed design changes, design drawings, material specifications, and safety classifications of the components. The staff determined that quality category “A” components are designed and constructed in accordance with ASME Section III Subsection NF using ASME SA-240 Type 304 stainless steel. This includes the DSC basket end caps for damaged fuel and the FFCs for failed fuel. The design and materials of construction for the damaged fuel end caps and the FFCs are similar to previously approved end caps and FFCs used in other CoC No. 1004 DSCs. The staff determined that the materials of construction for the damaged fuel end caps and the FFCs were adequate because (1) the design includes screens at the bottom and top to contain fuel debris and allow filling/drainage and (2) these components are constructed in accordance with the ASME Code Section III subsection NF and using ASME code approved materials.

8.2 Environmental Conditions

The applicant did not provide an analysis of the potential degradation due to irradiation of the stainless steel DSC, carbon steel components of the Standardized NUHOMS® System, or aluminum components of the DSC. Previous assessment of neutron fluence have been conducted for dry storage systems. For dry storage systems, a neutron flux of 10^4–10^6 n/cm^2-s [6.5×10^4–6.5×10^6 n/in^2-s] is typical (Sindelar et al., 2011). At these flux levels, the accumulated neutron fluence after 60 years is about 10^{13}–10^{15} n/cm^2 [6.5×10^{13}–6.5×10^{15} n/in^2].

The Draft MAPS Report (NRC, 2017) includes an assessment of the effects of neutron radiation on stainless steels, carbon steels and aluminum alloy materials. For stainless steels, Gamble (2006) found that neutron fluence levels greater than 1×10^{20} n/cm^2 [6.5×10^{20} n/in^2] are required to produce measureable degradation of the mechanical properties. Caskey et al. (1990) also indicates that neutron fluence levels of up to 2×10^{21} n/cm2 [1×10^{22} n/in^2] were not found to enhance SCC susceptibility. For carbon and alloy steels, neutron irradiation has the potential to increase the tensile and yield strength and decrease the toughness of carbon and alloy steels (Nikolaev et al., 2002). Neutron fluence levels greater than 10^{19} neutrons/square centimeter (n/cm^2) [6.5×10^{19} n/in^2] are required to produce a measureable degradation of the mechanical properties (Nikolaev et al., 2002; Odette and Lucas, 2001). For aluminum alloys, Farrell and King (1973) showed that pure aluminum had increased strength but decreased ductility after being irradiated to fast fluences in the range of 1 to 3 × 10^{22} n/cm^2 [6.5 to 18 × 10^{22} n/in^2] from a research reactor for 8 years. Alexander (1999) showed that irradiation at 10^{22} n/cm^2 [6.5 × 10^{22} n/in^2] simulating reactor conditions affected the mechanical properties of aluminum alloy 6061-T651.

To verify the conservatism of the previous estimate of accumulated neutron fluence by Sindelar et al. (2011), the NRC staff performed an independent calculation of the maximum potential accumulated neutron fluence on the dry storage system components (NRC, 2017). The staff considered components most directly exposed to the radiation source (middle of the fuel basket) and assumed fuel is loaded immediately after it is removed from the reactor vessel and stored for 100 years. To further provide a bounding estimate, the staff assumed a cask design that uses 40 Westinghouse 17×17 PWR fuel assembles with an average burnup of 70 Gwd/MTU and 4.0 fuel enrichment. The staff calculated the neutron source term for neutrons with energy at or greater than 1 MeV using the Origen/Arp computer code of the SCALE 6.1 computer code system. At this location, the total accumulated neutron fluence after 100 years of storage was calculated to be 2.63 × 10^{16} n/cm^2 [1.70 × 10^{17} n/in^2]. This worst-case estimate is greater than that calculated using the flux levels reported in Sindelar et al. (2011), however, the NRC determined the fluence level is still three orders of magnitude below the levels reported to degrade the fracture resistance of carbon and alloy steels, stainless steels, and aluminum alloys (NRC, 2017). Therefore, the staff concluded that the changes proposed in in the application are acceptable because the neutron fluence is insufficient to result in a degradation of material properties of the storage system components.

8.3 Neutron Poison Materials for Criticality Control

The applicant stated that for the 32PT System, the proposed changes include a basket option to increase the number of poison plates from 24 to 32 resulting in an increase in the allowable enrichment of the authorized contents. The design changes affect the number of neutron absorber plates used in the 32PT System.

The staff reviewed the proposed design changes, design drawings, material specifications, and safety classifications of the neutron absorber materials. The staff determined that the neutron absorber plates are quality category A components that are a non-code material. The staff determined that the neutron absorber plates are enriched borated aluminum alloy or metal matrix composite (MMC) which were previously evaluated and approved for use in the 32PT DSC as well as other dry storage systems. The staff determined that the changes proposed in the application are acceptable because the change is to increase the number of neutron absorber plates and does not alter the previously approved material used for the neutron absorber.

8.4 Mechanical Properties and Creep Analyses

The applicant stated that for the 24PTH, 32PT, 32PTH1, and 61BTH DSCs system, the proposed changes include the loading of 2 year cooled fuel. The applicant revised the material to be stored in the 32PT DSC increasing the maximum assembly burnup from 55 to 62 Gwd/MTU and to allow for the storage of damaged or failed fuel in the 32PT DSC. The applicant proposed new HLZC for these DSCs to accommodate fuel that had been cooled for a minimum of 2 years and increased the maximum allowable per assembly decay heat. However, the applicant did not increase the maximum heat load for any DSC. The applicant provided analyses of the component temperatures to support the new HLZC for the 32PT, 24PTH and 32PTH1 DSCs, the revised HLZC for the 61BTH DSC and the storage of damaged and failed fuel in the 32PT DSC.

The staff reviewed the calculated component temperatures including the DSC shell, basket and the neutron absorber for each of the new HLZCs for 32PT, 24PTH and 32PTH1 DSCs and the revised HLZC for the 61BTH DSC. The staff determined that the changes in the DSC component temperatures resulting from the new or revised HLZC in the application are minimal. The staff determined that the proposed new and revised HLZC are acceptable because the temperatures of the DSC components remain below their respective temperature limits for normal, off-normal and accident conditions.

The staff reviewed the calculated component temperatures including the DSC shell, basket and the neutron absorber for the storage of damaged and failed fuel in the 32PT DSC. The staff determined that the storage of damaged and failed fuel in the 32PT DSC is acceptable because temperatures of the DSC components remain below their respective temperature limits for normal, off-normal and accident conditions.

The staff reviewed the calculated temperatures for the aluminum alloy basket transition rails for each of the new HLZCs for 32PT, 24PTH and 32PTH1 DSCs and the revised HLZC for the 61BTH DSC. The staff also reviewed the calculated temperatures for the aluminum alloy basket transition rails for the storage of damaged and failed fuel in the 32PT DSC. The staff reviewed the potential for changes to mechanical properties and creep of the aluminum alloy basket transition rails. The staff used the guidance included in the Draft MAPS Report (NRC, 2017) to assess the effect of temperature on thermal aging of the aluminum alloy basket transition rails (MAPS Report Section 3.2.3.7) and creep (MAPS Report Section 3.2.3.5). The staff determined that the changes in the temperatures for the aluminum alloy basket transition rails resulting from the new or revised HLZC in the application are minimal. However, the staff determined that the temperatures of the aluminum alloy basket transition rails is sufficiently high to result in over-aging of the alloy resulting in a loss of strength over time. The staff note that the applicant assumed properties of annealed material for the material properties of the aluminum alloy basket transition rails to account for the reduced strength associated with over-aging. The staff determined the approach of using the annealed properties is consistent with the stipulation in ASME Section II part D to use time dependent mechanical properties when a heat treated aluminum alloy is used at temperatures where the mechanical properties can be altered. The staff determined that the proposed new and revised HLZC are acceptable because the mechanical properties of the aluminum alloy basket transition rails are assumed to be equivalent to the fully annealed material to account for the effects of temperature.

8.5 Galvanic/Chemical/Radiolytic Reactions of Fuel with Canister Internals

The application included two changes to the proposed contents including (i) an expanded definition of the poison rod assemblies (PRAs) to include rod cluster control assemblies (RCCA) materials, specifically silver neutron absorber for the 32PT DSC, and (ii) the addition of GNF-2 and ATRIUM-11 fuel assembly designs as authorized contents for the 61BTH System.

The staff reviewed the additional contents proposed in the application for compatibility with the DSC materials under the range of environmental conditions including loading and drying operations. The staff followed the guidance in NUREG-1536, Revision 1, Section 8.4.8 Galvanic/Corrosive Reactions to assess the potential for galvanic, chemical and/or radiolytic reactions of the fuel or contents with canister internals. The staff's review noted the previous assessment of these materials in NUREG-1536, Revision 1, Section 8.4.8 states the following:

> *The staff has found the following materials to be acceptable for storage when the canister is constructed of stainless steel with stainless steel and aluminum basket components: Neutron source materials composed of stainless steel or zirconium alloy cladding containing: antimony-beryllium, americium-beryllium, plutonium-beryllium, polonium-beryllium, and californium. Exposure of these various contents to the wet loading and dry storage environment was assessed and found to be satisfactory. Control elements composed of zircaloy or stainless steel cladding containing: boron carbide, borosilicate glass, silver-indium-cadmium alloy, or thorium oxide. Exposure of these various contents to the wet loading and dry storage environment was assessed and found to be satisfactory.*

Because these materials have previously been evaluated for potential galvanic, chemical and/or radiolytic reactions of the fuel or contents with canister internals, the staff determined that the materials included in the application are acceptable.

8.6 Fuel Burn-up

The applicant stated that the allowable maximum assembly average burnup for the 32PT DSC was increased from 55 Gwd/MTU to 62 Gwd/MTU. The overall heat load of the 32PT System, the initial fuel enrichment, and the maximum cladding temperature limits for normal and off-normal conditions were not changed in the application.

The staff reviewed the proposed increased burn-up limit for the 32PT Systems. The staff determined that the increase in the maximum assembly average burnup from 55 Gwd/MTU to 62 Gwd/MTU for the 32PT DSC is acceptable because it is consistent with NUREG-1536, Revision 1, Section 8.4.17.1 (NRC, 2010). The NRC Office of Nuclear Reactor Regulation (NRR) has previously approved PWR fuel burnup up to 62 Gwd/MTU. This approval was supported by the technical basis provided in a reactor license amendment request and the finding that there are no significant adverse environmental impacts associated with extending peak-rod fuel burnup to 62 Gwd/MTU (Ramsdell, Jr. et al., 2001). The NRC has also previously approved PWR fuel with a maximum burnup of 62 Gwd/MTU for dry storage systems that followed the guidance in NUREG-1536, Revision 1, Sections 8.4.17 and 8.4.18.

8.7 Cladding Temperature Limits

The applicant proposed several additional changes that required review of cladding temperature limits. These changes included:

1. Change #1 including changes to the fuel qualification tables for four PWR systems (32PT, 24PTH, 32PTH1 and 37PTH) in order to simplify the Technical Specifications. The standardized fuel qualification tables (FQTs) provide for minimum required cooling times, as low as two years, as a function of enrichment and burnup (BU) for all the heat loads described in the various heat load zoning configurations (HLZCs) for these four PWR systems.
2. Change #2 which included the addition of a new HLZC #4 to the 32PT System to allow for the loading of FAs with decay heat up to 2.2 kW corresponding to a 2-year cooled fuel.
3. Change #3 to increase the maximum assembly average BU from 55 Gwd/MTU to 62 Gwd/MTU.
4. Change #6 which included the addition of a new HLZC #6 for the 24PTH System to allow for the loading of FAs with decay heat up to 2.5 kW corresponding to a 2-year cooled fuel, and a total heat load of 35 kW per basket.
5. Change #7 which included the addition of the OS197 as an authorized transfer cask for the 24PTH System.
6. Change #8 which included the revision of the existing HLZC #10 for the 61BTH System to allow loading FAs with decay heat up to 1.2 kW corresponding to a 2-year cooling time.
7. Change #9 which included the addition of a new HLZC #5 for the 32PTH1 System to allow for the loading of FAs with decay heat up to 1.1 kW for a total heat load of 35.2 kW per basket and HLZC #6 to allow for loading of FAs with decay heat up to 1.3 kW for a total heat load of 37.6 kW per basket.

The applicant stated that the changes were evaluated for structural, thermal, shielding, confinement and criticality adequacy, as applicable, and has concluded that these changes to the Standardized NUHOMS® System have no significant effect on safety.

The staff reviewed the proposed changes included in the application along with the guidance on cladding temperature limits included in NUREG-1536, Revision 1, Section 8.4.17.1. As noted in NUREG-1536, Revision 1, there are three considerations for cladding temperature limits:

- For high burn-up fuel, defined as any fuel with a burn-up greater than 45Gwd/MTU, the maximum allowable cladding temperature limit is 400°C,
- During loading operations, repeated thermal cycling (repeated heatup/cooldown cycles) may occur but should be limited to less than 10 cycles, where cladding temperature variations are not more than 65°C (117°F) each, and
- For off-normal and accident conditions, the maximum cladding temperature should not exceed 570°C (1058°F).

The staff confirmed that the maximum fuel cladding temperature limit of 400°C (752°F) is applicable to normal conditions of storage and all short term operations from spent fuel pool to ISFSI pad including vacuum drying and helium backfilling for the 32PT, 24PTH, 32PTH1 and 37PTH and the 61BTH DSC. The staff confirmed that the operational specifications do not permit thermal cycling of the fuel cladding with temperature differences greater than 65° C (117°F) during DSC drying, backfilling and transfer operations. The staff confirmed that for off-normal and accident conditions, the maximum cladding temperature does not exceed 570°C

(1058°F). The staff determined that the proposed changes in the application are acceptable with respect to fuel cladding because the proposed changes in the application result in cladding temperatures and temperature cycles that follow the guidance in NUREG-1536, Revision 1, for normal, off-normal and accident conditions.

8.8 Damaged Fuel Definition

The applicant stated that Change #4 was introduced for the 32PT System to allow for the loading of (1) damaged fuel assemblies confined within top and bottom end caps to ensure retrievability and (2) failed fuel assemblies loaded within individual failed fuel canisters. The applicant defined damaged and failed fuel as follows:

- Damaged assemblies are assemblies containing missing or partial fuel rods, fuel rods with known or suspected cladding defects greater than hairline cracks or pinhole leaks. The extent of damage in the fuel assembly, including noncladding damage, is to be limited such that a fuel assembly is able to be handled by normal means and retrievability is ensured following normal and off-normal conditions.

- Failed fuel is defined as ruptured fuel rods, severed fuel rods, loose fuel pellets, or fuel assemblies that cannot be handled by normal means. Fuel assemblies may contain breached rods, grossly breached rods, and other defects such as missing or partial rods, missing grid spacers, or damaged spacers to the extent that the assembly cannot be handled by normal means. Fuel debris and fuel rods that have been removed from a fuel assembly and placed in a rod storage basket are also considered as failed fuel. Loose fuel debris, not contained in a rod storage basket must be placed in a failed fuel can for storage, provided the size of the debris is larger than the failed fuel can screen mesh opening.

In response to a request for additional information (ML17363A276), the applicant clarified that damaged fuel assemblies are required to be handled by normal means, which refers to the use of the crane and grapple to handle and load damaged fuel assemblies, and the damaged fuel assemblies are confined to their respective compartments by means of top and bottom end caps. The applicant also clarified that from the standpoint of NUREG 1536, Revision 1, the damaged fuel assemblies for the 32PT are more similar to the “undamaged” fuel assemblies where their geometry is still in the form of intact bundles. The applicant clarified that the “failed” fuel assemblies for the 32PT are more similar to the “damaged” fuel assemblies per NUREG 1536, Revision 1.

The applicant also stated that the fuel compartment and the top and bottom end caps together form the “acceptable alternative, per NUREG-1536, Revision 1” for confinement of damaged fuel assemblies. The applicant stated that the top and bottom end caps provide for the confinement of gross fuel particles to a known volume in the event that any fuel particles smaller than a pellet that are released from the damaged assembly. The applicant stated that the bottom end cap is designed to be removable and contains socket for the use of a handling tool which allows gross fuel particles to be retrieved.

The staff reviewed the application and the guidance included in NUREG-1536, Revision 1, Section 8.4.17.2 for fuel classification. The applicant has limited the storage of damaged fuel with the use of basket endcaps to fuel assemblies that can be handled by normal means after normal and off-normal events. The applicant's approach, which included the use of endcaps to contain debris for damaged fuel and the use of individual failed fuel canisters, provided the size of the debris is larger than the failed fuel can screen mesh opening, is consistent with the guidance in NUREG-1536, Revision 1, and previously approved damaged and failed fuel storage in the DSCs included in the CoC No. 1004. The staff determined that the application was acceptable because the content of the application with respect to fuel classification was consistent with the guidance in NUREG-1536, Revision 1, Sections 8.4.17.2 and 8.6.C.

The staff determined that for damaged fuel in the 32PT DSC using damaged fuel endcaps, the functions the applicant has imposed on the damaged fuel assemblies and the damaged fuel endcaps by fuel specific and system-related functions meet a regulatory requirement for storage. Specifically, the staff determined that the applicant's specifications for damaged fuel and the functions of the damaged fuel endcaps meet the regulatory requirements of 10 CFR 72.236(h) and (m) and allow the system users to meet the regulatory requirements of 10 CFR 72.122(h)(1) and (h)(5). The thermal, shielding and criticality evaluations of the 32PT DSC for the storage of damaged fuel is included in sections 4, 6 and 7 of this SER respectively.

8.9 Evaluation of Amendment No. 15 on the CoC No. 1004 Renewal

The applicant provided an assessment of the Standardized NUHOMS® System CoC No. 1004 renewal scoping evaluation, aging management review (AMR), and fuel retrievability review for the SSC's determined to be within the scope of the application. The applicant identified that the proposed changes resulted in the addition of failed fuel cans to the 32PT DSC basket. The applicant provided a table to update the 32PT DSC AMR due to this change and determined that there were no aging effects requiring management and no aging management activities required for any of the added subcomponents. The applicant concluded that (1) no UFSAR changes were identified due to Amendment No. 15 that caused any additions to the Standardized NUHOMS® System CoC No. 1004 renewal UFSAR changes, and (2) no technical specification changes were identified due to Amendment No. 15 that caused any additions to the Standardized NUHOMS® System CoC No. 1004 renewal technical specification changes.

The staff reviewed the evaluation of the proposed Amendment No. 15 changes on the CoC No. 1004 renewal and determined that the applicant's assessment is acceptable because the failed fuel cans and the top and bottom basket endcaps for damaged fuel added to the 32PT DSC are the only added structures, systems or components that require aging management review. The staff determined that for the basket endcaps for damaged fuel and the failed fuel cans added to the scope, no unique aging effects were identified that require a UFSAR/Technical Specifications change because these components are constructed from materials that will not react in loading operations or degrade in storage.

8.10 Findings

F8.1. The applicant has met the requirements in 10 CFR 72.236(b). The applicant described the materials design criteria for SSCs important to safety in sufficient detail to support a safety finding.

F8.2. The applicant has met the requirements in 10 CFR 72.124(b). Neutron absorbing materials are demonstrated to effectively control criticality without significant degradation over the storage life.

F8.3. The applicant has met the requirements in 10 CFR 72.236(g). The properties of the materials in the storage system design have been demonstrated to support the safe storage of SNF.

F8.4. The applicant has met the requirements in 10 CFR 72.236(h). The materials of the SNF storage container are compatible with their operating environment such that there are no adverse degradation or significant chemical or other reactions.

F8.5. The applicant has met the requirements in 10 CFR 72.236(a) and 10 CFR 72.236(m). SNF specifications have been provided and adequate consideration has been given to compatibility with retrieval of stored fuel for ultimate disposal.

8.11 References

Alexander, D.J. “Effects of Irradiation on the Mechanical Properties of 6061-T651 Aluminum Base Metal and Weldments.” ASTM Special Technical Publication. Vol. 1325. pp. 1,027–1,044. 1999.

Caskey, G.R., R.S. Ondrejcin, P. Aldred, R.B. Davis, and S.A. Wilson. “Effects of Irradiation on Intergranular Stress Corrosion Cracking of Type 304 Stainless Steel.” Proceedings of 45th

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NRC, “Standard Review Plan for Spent Fuel Dry Storage Systems at a General License Facility,” Final Report NUREG-1536 Revision 1, Washington, DC: U.S. NRC, July 2010.

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Odette, G.R. and G.E. Lucas. “Embrittlement of Nuclear Reactor Pressure Vessels.” Journal of Metals. Vol. 53, Issue 7, pp. 18-22, 2001.

Ramsdell, Jr., J.V., C.E. Beyer, D.D. Lanning, U.P. Jenquin, R.A. Schwarz, D.L. Strenge, P.M. Daling, and R.T. Dahowski, “Environmental Effects of Extending Fuel Burnup Above 60 Gwd/MTU,” NUREG/CR-6703, Richland, WA: Pacific Northwest National Laboratory, January, 2001.

Sindelar, R.L., A.J. Duncan, M.E. Dupont, P.-S. Lam, M.R. Louthan, Jr., and T.E. Skidmore. NUREG/CR-7116, “Materials Aging Issues and Aging Management for Extended Storage and Transportation of Spent Nuclear Fuel.” Washington, DC: U.S. Nuclear Regulatory Commission. 2011.

9.0 OPERATING PROCEDURES EVALUATION

The applicant stated that changes associated with Amendment 15 do not result in any changes to the design of the major components of the Standardized NUHOMS® System and that changes are mostly related to the authorized contents. Each user of the standardized NUHOMS® System prepares written procedures for all normal operations (cask handling, loading movement and surveillance) and maintenance at the ISFSI prior to its operation. Operating procedures suggested generically in the UFSAR provide the basis for the user's written operating procedures. The applicant included revised operating procedures for the loading the 32PT DSC with damaged and failed fuel. For the loading of damaged fuel, the applicant’s revised procedures included the installation of top and bottom end caps into the basket locations where damaged fuel are also loaded. For failed fuel, the applicant’s procedures ensure that the failed fuel can lids are installed. The applicant’s procedures include controls to ensure that damaged and failed fuel are placed within known basket cell locations within the 32PT DSC. The applicant also included procedures for unloading the 32PT DSC with damaged and failed fuel.

The staff reviewed the revised operating procedures for the loading and unloading of damaged and failed fuel for the 32PT DSC. The staff determined that the procedures are complete and appropriately reference the Technical Specifications for loading of damaged and failed fuel. The staff determined that the revised operating procedures for loading were acceptable because the loading procedures include (1) the use of a failed fuel can and a failed fuel can lid for failed fuel, and (2) the use of top and bottom basket end caps to confine any fuel debris. In addition, the staff determined that the applicant’s unloading procedures are acceptable because the procedures are complete and ensure that the damaged and failed fuel, including fuel debris, can be unloaded from the 32PT DSC. The staff concludes that the proposed changes to Operations for Standardized NUHOMS® System continues to meet the requirements of 10 CFR Part 72.

10.0 ACCEPTANCE TESTS AND MAINTANANCE PROGRAM EVALUATION

The applicant's proposed changes to the acceptance tests and maintenance program for the Standardized NUHOMS® System include additional acceptance testing for the silver-indium-cadmium poison rod assemblies for use in W17x17 fuel assemblies in the 32PT DSC. These additional acceptance tests consist of linear density testing as described in SAR section M.9.1.7.11. The staff finds that the testing described is sufficient to demonstrate that the silver-indium-cadmium poison rod assemblies meet the minimum silver linear density requirement of Table 1-1h of the TS.

11.0 RADIATION PROTECTION EVALUATION

The staff reviewed proposed changes for the Standardized NUHOMS® System to ensure that it will continue to meet the regulatory dose requirements of 10 CFR Part 20 (Ref. 1), 10 CFR 72.104(a), 10 CFR 72.106(b), 10 CFR 72.212(b), and 10 CFR 72.236(d) (Ref. 2). The proposed amendment was also reviewed to determine whether the Standardized NUHOMS® System continues to fulfill the acceptance criteria listed in Section 11 of NUREG-1536, "Standard Review Plan for Dry Cask Storage Systems" (Ref. 3). The staff's review is based on information provided in proposed Amendment No. 15 to the Standardized NUHOMS® System SAR (Ref. 4) and responses to the staff's requests for additional information (Ref. 5).

Radiation Protection Design Criteria

Design Criteria

The applicable radiological protection design criteria are the limits and requirements of 10 CFR Part 20, 10 CFR 72.104, and 10 CFR 72.106. As required by 10 CFR Part 20 and 10 CFR 72.212, each general licensee is responsible for demonstrating site-specific compliance with these requirements. The Standardized NUHOMS® System Technical Specifications (Section 5.4 HSM or HSM-H Dose Rate Evaluation Program) also establishes dose rates limits for the TC and HSM that are based on calculated values used to ensure occupational and off-site radiological exposures from operating the system will meet regulatory limits.

The Standardized NUHOMS® System Technical Specifications also establish exterior contamination limits of 2,200 dpm/100 cm^2 for beta and gamma radiation, and 220 dpm/100 cm^2 for alpha radiation.

Design Features

There are no proposed changes to design features for the Standardized NUHOMS® System.

Occupational Exposures

For Amendment No. 15, the applicant proposed to add the loading of higher decay heat load assemblies and the loading of damaged and failed fuels into the 61BTH and the 24PTH, 32PTH1, 32PT, and 37PTH PWR storage and transfer systems. The shielding analyses performed for these systems is discussed in the shielding section of Appendices M, P, U, T and Z of the SAR.

The proposed 0.380 MTU/FA loadings have higher TC and HSM dose rates than the 0.490 MTU/FA currently authorized. The licensee developed new scaling factors for 0.380 MTU FAs instead of repeating the second sets for storage and transfer dose rates, occupational exposures, and site doses as shown in the SAR for 0.380 MTU /FA. Table M.10-1 Occupational Exposure Summary (32PT-S125/32PT-L125 DSC configuration) and Table M.10-2 Occupational Exposure Summary (32PT-S100/32PT-L100 DSC configuration) for the 32PT canisters show these scaling factors. The detail of these scaling factors are reported in the SAR Section U.5.4.12 for the 32PTH1 System and how these factors are applied to other systems are also reported in the SAR (FSAR Section P.5.4.11 for the 24PTH System, Section M.5.4.16 for the 32PT System, and Section Z.5.4.11 for the 37PTH).

As discussed in the Shielding section of the SAR, the applicant performed a calculation for the 32PT DSC inside the OS200 Transfer Cask using MCNP5 for normal and accident conditions. The MCNP model was rerun for decontamination and welding configurations operational steps for 0.380 MTU/FA. The results of dose rates for site, occupational exposure for 0.380 MTU/FA are compared with results of 0.490 MTU/FA dose rates, respectively. The scaling factors were determined by dividing the results of site and occupational exposure for 0.380 and .490 MTU/FA. Based on the results for the scaling factors shown in the Tables U.5-28 through U.5-33, the maximum scaling factors are selected for the 0.380 MTU/FA. For example for the HSM End (Side) Shield Wall Surface and HSM Back Shield Wall, scaling factor are 1.36 and 1.06. In this case the 1.36 is the derived maximum scaling factors that are used in the dose rates report. This is more conservative than results from MCNP5 calculation.

The results for the normal transfer of DSC, decontamination of DSC, and welding DSC configurations, MCNP models dose rates from Tables U-5.30, U-5.31 U-5.32 are reported in Table U-10.1 to determine the total exposure for the 0.380 MTU/FA and to determine the number of exposed workers during each steps of the operation, duration of each steps, and dose rate for each operational steps. The total cumulative operational exposure which is summation of all operational steps given at the bottom of the Table U-10.1 is 1934 person-mrem for 0.490 MTU/FA.

From Table U-5.33 also the total cumulative operational exposure for 0.380 MTU/FA is 2425 person-mrem. Therefore, the 0.380 MTU/FA loading operational exposure for all operational steps is 25 % more than loading 0.490MTU/FA or a factor of 1.25.

The results for 24PTH, 32PT, and 37PTH are shown in the Tables M-10.1, P-10.1, and Z-10.1 respectively.

Off-site Dose calculation:

The SAR presents the calculated direct radiation dose rates at distances from 6.1 meters (20 feet) to 600 meters from each face of two arrays of HSMs: a 2x10 back-to-back array of HSM-HS loaded with design basis fuel in Standardized NUHOMS® System DSCs, and two 1x10 front-to-front arrays of HSM-HS loaded with design-basis fuel in Standardized NUHOMS®

System DSCs. The HSM is modeled in MCNP as a box representing the HSM arrays. The DSC design basis heat load configuration for each of these evaluations is contained in the SAR for each of the DSCs. Section 10.2 of the Appendices M, P, U and Z of the SAR presents the calculated direct radiation dose rates at distances from 6.1 meters (20 feet) to 600 meters from each face of two arrays of HSMs: a 2x10 back-to-back array of HSM-Hs loaded with design basis fuel in Standardized NUHOMS® System DSCs, and two 1x10 front-to-front arrays of HSM-Hs loaded with design-basis fuel in Standardized NUHOMS® System DSCs. The total annual exposure for each ISFSI layout as a function of distance from each face is given in Section 10 of each appendix. The total annual exposure estimates assume 100% occupancy for 365 days.

The dose received by a person at 100 meters from the ISFSI for duration of 8 hours for an array of 2X10 of HSMs (2X10 back-to-back and front-to-front arrays) is 2*8 hours*8.75E-02 mrem per hour at 100 meter*1.25 scaling factor which is less than 5 mrem. At 500 meters, the dose is 2*8 hours*1.83E-04 mrem per hour*1.25 which is less than 0.01 mrem.

The SAR indicated that the general licensees may choose to modify the sequence of operations, and will also use site specific ALARA practices to mitigate occupational exposure.

Confirmatory Calculation

The staff evaluated the public dose estimates during normal and off-normal conditions. The primary dose pathway to individuals beyond the controlled area is from direct radiation (including skyshine). A discussion of the staff's evaluation and confirmatory analysis of the shielding calculations are presented in Section 5 of the SER.

The staff has reasonable assurance that compliance with 10 CFR 72.104(a) can be achieved using the Standardized NUHOMS® System with the HSM. The actual doses to individuals beyond the controlled area boundary depend on several site-specific conditions, such as fuel characteristics, cask-array configurations, topography, demographics, and use of engineered features (e.g., berms). In addition, the dose limits in 10 CFR 72.104(a) include doses from other fuel cycle activities, such as reactor operations. Consequently, final determination of compliance with 10 CFR 72.104(a) is the responsibility of each site license. Additionally, engineered features (e.g. earthen berms, shield walls) that are used to ensure compliance with 10 CFR 72.104(a) by each general licensee are to be considered important to safety and must be appropriately evaluated under 10 CFR 72.212(b).

The general licensee should establish a radiation protection program as required by 10 CFR Part 20, Subpart B, and will demonstrate compliance with dose limits to individual members of the public, as required in 10 CFR Part 20, Subpart D, by evaluations and measurements.

The staff evaluated the public dose estimates from direct radiation from accident conditions and natural phenomena events in Sections 5 of this SER and found them acceptable. A discussion of the staff's evaluation of the accident conditions and recovery actions are presented in Section 11 of the SER. The staff has reasonable assurance that the effects of direct radiation from bounding design basis accidents and natural phenomena will be below the regulatory limits in 10 CFR 72.106(b).

Evaluation Findings

F11.1 The SAR sufficiently describes the radiation protection design bases and design criteria for the SSCs important to safety.

F11.2 Radiation shielding and confinement features are sufficient to meet the radiation protection requirements of 10 CFR Part 20, 10 CFR 72.104, and 10 CFR 72.106.

F11.3 The SAR adequately evaluates the Standardized NUHOMS® System DSCs and their systems important to safety to demonstrate that they will reasonably maintain confinement of radioactive material under normal, off-normal, and accident conditions.

F11.4 The SAR sufficiently describes the means for controlling and limiting occupational exposures within the dose and ALARA requirements of 10 CFR Part 20.

F11.6 Operational restrictions necessary to meet dose and ALARA requirements in 10 CFR Part 20, 10 CFR 72.104, and 10 CFR 72.106 are the responsibility of the site licensee. The Standardized NUHOMS® System DSCs are designed to assist in meeting these requirements.

References

U.S. *Code of Federal Regulations*, Standards for Protection against Radiation, Title 10, Part 20.

U.S. *Code of Federal Regulations*, Licensing Requirements for the Independent Storage of Spent Nuclear Fuel, High-Level Radioactive Waste, and Reactor - Related Greater Than Class C Waste, Title 10, Part 72.

U.S. Nuclear Regulatory Commission, Standard Review Plan for Dry Cask Storage Systems at a General License Facility, NUREG-1536, July 2010.

TransNuclear, Safety Analysis Report of the Standardized NUHOMS® Modular Storage System for Irradiated Nuclear Fuel, 2117, Revision 15 (ML17363A276).

TN Americas LLC, Response to First Request for Additional Information on Application for Amendment No. 15 to Standardized NUHOMS Spent Fuel Storage Casks, Revision 2 (ML17363A276).

12.0 ACCIDENT ANALYSIS EVALUATION

The applicant did not request changes to the principal design criteria related to the SSCs important to safety. For this reason, the staff finds the applicant complied with the relevant general criteria established in 10 CFR Part 72, and does not require an accident analysis evaluation of the principal design criteria. Internal pressures changes were investigated as part of the thermal evaluation and found to be either bounding, or non safety significant for all cases, therefore no further confinement evaluation was necessary for accident conditions.

13.0 TECHNICAL SPECIFICATIONS

The staff reviewed the proposed amendment to determine that applicable changes made to the conditions in the certificate of compliance, and to the TSs for CoC No. 1004, Amendment No. 15 would be in accordance with the requirements of 10 CFR Part 72. The staff reviewed the proposed changes to the Technical Specifications to confirm the changes were properly evaluated and supported in the applicant's revised safety analysis report. The applicant's proposed changes to the Technical Specifications are as follows:

Table 13-1 - Conforming Changes to the Technical Specifications

TS page	TS Number	Description	Scope Item
Cover page	N/A	Amendment number changed to 15	none
Table of Contents, List of Tables, and List of Figures	N/A	Updated	none
1-1	1.1	Definition for intact, damaged and failed fuel broken into INTACT, then DAMAGED/FAILED	4
3-7	3.1.3	HLZC #6 added for 24PTH and HLZC #5 and #6 added for 32PTH1	6, 9
3-11	3.2.1	Added boron concentrations tables references for 32PT	4
4-39	4.3.3	Added flexibility to general licensees in verifying compliance regarding the storage pad location and the soil-structure interaction	11
5-7	5.2.4.e	Updated TC dose rates for 32PT, 24PTH, 32PTH, and 37PTH	1
5-14	5.4.2	Updated HSM does rates for 32PT, 24PTH, 61BTH, 32PTH1, 69BTH, and 37PTH	1
T-5	Table 1-1e	Remove the Table 1-1e row for "Fuel Cladding Material" in order to remove specificity to "Zircaloy" and therefore allow other cladding materials, consistent with the approach in other TS tables such as TS Table 1-1i for the 24PHB System, Table 1-1l for the 24PTH System, Table 1-1aa for the 32PTH1 System, and Table 1-1ll for the 37PTH System, etc.	5
T-5 and T-6	Table 1-1e	Changes made to the fuel specification table for the 32PT – including intact fuel	1, 4

		description and fuel damage definition clarification	
T-7	Table 1-1f	Changes made to the FA design characteristics for the 32PT	4
T-8	Table 1-1g	Changes made to the 32PT table for certain fuel assembly parameters (intact fuel) and to clarify the control component (CC) configurations; deleted “AIC” from Notes because there is no option for loading AIC PRAs in Table 1-1g	4
T-9 and T-10	Table 1-1g1	Changes made to the 32PT table for certain basket parameters (intact fuel) and to clarify the CC configurations	4
T-11	Table 1-1g2	New 32 PT table for certain basket parameters (damaged fuel) and to clarify the CC configurations	4
T-12	Table 1-1g3	New 32 PT table for certain basket parameters (damaged or failed fuel) and to clarify the CC configurations	4
T-13	Table 1-1h	Changes made to the B10 specification table for the 32PT, and update to the minimum silver content per AIC absorber rod	4
T-14	Table 1-1i	Changes made to the fuel specification table for the 24PHB, including clarification of the fuel damage definition	4
T-16	Table 1-1j	Changes made to the fuel specification table for the 61BT, including clarification of the fuel damage definition	4
T-19 and T-21	Table 1-1l	Changes made to the fuel specification table for the 24PTH, including clarification of the fuel damage definition and fuel class description	1, 4
T-30	Table 1-1r	Changes made to the B10 specification table for the 24PTH	6
T-32 and T-34	Table 1-1t	Changes made to the fuel specification table for the 61BTH, including clarification of the fuel damage definition and fuel class description	4, 8
T-35	Table 1-1u	Changes made to the FA design characteristics table for the 61BTH	8
T-36	Table 1-1v	Changes made to the 61BTH for certain basket parameters (intact fuel)	8
T-37	Table 1-1w	Changes made to 61BTH for certain basket parameters (damaged fuel)	8

T-38	Table 1-1w1	Changes made to 61BTH for certain basket parameters (failed and damaged fuel)	8
T-42 and T-43	Table 1-1aa	Changes made to the fuel specification table for the 32PTH1, including clarification of the fuel damage definition and fuel class description	1, 4
T-44	Table 1-1bb	Changes made to the FA design characteristics for the 32PTH1	1
T-53	Table 1-1gg	Changes made to the fuel specification for the 69BTH, including clarification of the fuel damage definition	4
T-59 and T-60	Table 1-1ll	Changes made to the fuel specification for the 37PTH, including clarification of the fuel damage definition	1
T-69	Table 1-2c page	Revision bar signifies deletion of Tables 1-2d through 1-2m	1
T-70	None	Tables 1-2d through 1-2m are deleted	1
T-75 to T-122	Tables 1-3a to 1-3p	These tables are the new, generic PWR FQTs	1
T-123 to T-125	None	These are the notes for the new, generic PWR FQTs	1
T-126 to T-137	Tables 1-4a to 1-4f	Changes made to 61BTH FQT	8
T-138	Table 1-4g	Changes made to 61BTH FQT	8
T-139 to T-142	Tables 1-4h and 1-4i	Changes made to 61BTH FQT	8
T-143 to T-144	None	Changes made to the notes for 61BTH FQT	8
T-145	None	Tables 1-5a through 1-5g are deleted	1
T-179	None	Revision bar signifies deletion of Tables 1-8a to 1-8f	1
F-2	Figure 1-2	Changes made to HLZC #1 for 32PT for damaged FAs	4
F-3	Figure 1-3	Changes made to HLZC #2 for 32PT for damaged and failed FAs	4
F-4	Figure 1-4	Changes made to HLZC #3 for 32PT for damaged FAs	4
F-5	Figure 1-4a	New HLZC #4 for the 32PT; deleted “Zone 2” from Note (2) because Note (2) is only associated with Zone 5	2
F-6	Figure 1-4b	New figure for location of damaged and failed fuel in the 32PT	4
F-14	Figure 1-11	Changes made to HLZC #1 for 24PTH	1
F-15	Figure 1-12	Changes made to HLZC #2 for 24PTH	1
F-16	Figure 1-13	Changes made to HLZC #3 for 24PTH	1
F-17	Figure 1-14	Changes made to HLZC #4 for 24PTH	1
F-18	Figure 1-15	Changes made to HLZC #5 for 24PTH	1

F-19	Figure 1-15a	New HLZC #6 for the 24PTH	6
F-31	Figure 1-25b	Changes made to HLZC #10 for 61BTH	8
F-36	Figure 1-28b	New HLZC #5 for 32PTH1	9
F-37	Figure 1-28c	New HLZC #6 for 32PTH1	9

The staff finds that the proposed changes to the Technical Specifications for the Standardized NUHOMS® System conform to the changes requested in the amendment application and do not affect the ability of the cask system to meet the requirements of 10 CFR Part 72. The proposed changes provide reasonable assurance that the Standardized NUHOMS® System will continue to allow safe storage of spent nuclear fuel.

14.0 QUALITY ASSURANCE EVALUATION

There were no changes to the applicant's quality assurance program requested in the amendment application.

15.0 CONCLUSIONS

The staff has performed a comprehensive review of the amendment application, during which the following requested changes to the Standardized NUHOMS® System were considered:

1. Unify and standardize the fuel qualification tables for four PWR systems (32PT, 24PTH, 32PTH1 and 37PTH) in order to simplify the TS. The standardized FQTs provide for minimum required cooling times, as low as two years, as a function of enrichment and BU for all the heat loads described in the various heat load zoning configurations HLZCs for these four PWR systems. Further, the FQTs are generated for three different MTU loadings per FA and allow for interpolation between MTU loadings and to establish cooling times for FAs that fall into the unanalyzed regions of the FQTs. For this purpose, the source term, dose rate, occupational exposure and site dose analyses have been revised for the four PWR systems described above. The TS and UFSAR Appendices M, P, U and Z have been revised accordingly.
2. For the 32PT System, add a new HLZC #4 to allow for the loading of FAs with decay heat up to 2.2 kW corresponding to a 2-year cooled fuel. The TS and UFSAR Appendix M have been revised to incorporate this new HLZC.
3. For the 32PT System, increase the maximum assembly average BU from 55 GWd/MTU to 62 GWd/MTU. The TS and UFSAR Appendix M have been revised to incorporate this change.
4. For the 32PT System, allow for the loading of damaged fuel assemblies confined within top and bottom end caps and failed fuel assemblies loaded within individual failed fuel canisters. Provide for a basket option to increase the number of poison plates from 24 to 32 resulting in an increase in the allowable enrichment of the authorized contents. Expand the definition of the PRAs to include RCCA materials, specifically silver neutron absorber. A clarification of the definition for damaged fuel for all DSCs was also made in the UFSAR sections and TS tables. Additionally, the TS now has a separate definition for intact fuel. The TS and UFSAR Appendix M have been revised to incorporate this change.

5. For the 32PT System, include other zirconium alloy cladding materials such as ZIRLO and M5. The TS and UFSAR Appendix M have been revised to incorporate this change.
6. For the 24PTH System, add a new HLZC #6 to allow for the loading of FAs with decay heat up to 2.5 kW corresponding to a 2-year cooled fuel, and a total heat load of 35 kW per basket. The TS and UFSAR Appendix P have been revised to incorporate this new HLZC and editorial changes are made to the TS for the descriptions of basket types.
7. For the 24PTH System, the OS197 is added as an authorized transfer cask (TC) for the transfer of the 24PTH-S-LC DSC in addition to the standardized TC. UFSAR Chapters P.1, P.2 and P.4 have been revised to incorporate this change.
8. For the 61BTH System, revise the existing HLZC #10 to allow loading FAs with decay heat up to 1.2 kW corresponding to a 2-year cooling time. GNF-2 and ATRIUM-11 FA designs are also added as authorized contents. Additionally, the FQTs with minimum cooling times of two years are generated for MTU loadings of 0.180 and 0.198 per fuel assembly at a decay heat of 1.2 kW and to establish cooling times for FAs that fall into the unanalyzed regions of all the FQTs. The TS and UFSAR Appendix T have been revised to incorporate these changes.
9. For the 32PTH1 System, add new HLZC #5 to allow for the loading of FAs with decay heat up to 1.1 kW for a total heat load of 35.2 kW per basket and HLZC #6 to allow for loading of FAs with decay heat up to 1.3 kW for a total heat load of 37.6 kW per basket. This is applicable for Type 1 DSCs using solid aluminum rails only. The TS and UFSAR Appendix U have been revised to incorporate these changes.
10. Provide a description in the UFSAR for the solar shield currently described in the TS for the TC during transfer operations. UFSAR Chapter 10 has been revised to incorporate this change.
11. Update Technical Specification 4.3.3 Item 11 to add flexibility to general licensees in verifying compliance regarding the storage pad location and the soil-structure interaction, which may affect the response of loaded HSMs.

Based on the statements and representations provided by the applicant in its amendment application, as supplemented, the staff concludes that the changes described above to the Standardized NUHOMS® System do not affect the ability of the cask system to meet the requirements of 10 CFR Part 72. Amendment No. 15 for the Standardized NUHOMS® System should be approved.

Issued with Certificate of Compliance No. 1004, Amendment No. 15
on December 14, 2018.