TRANSNUCLEAR, INC.

STANDARDIZED NUHOMS® HORIZONTAL MODULAR STORAGE SYSTEM

AMENDMENT NO. 7

PRELIMINARY SAFETY EVALUATION REPORT

SAFETY EVALUATION REPORT
AMENDMENT NO. 7 TO STANDARDIZED NUHOMS® SYSTEM

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SUMMARY

By application dated March 29, 2002, as supplemented on February 14, 2003 and July 10, 2003, Transnuclear, Inc. (TN) requested approval of an amendment, under the provisions of 10 CFR Part 72, Subpart K, to Certificate of Compliance No. 1004 for the Standardized NUHOMS® System.

TN requested changes to the Certificate of Compliance (CoC), including its attachments, and revision of the Final Safety Analysis Report (FSAR). The principle requested change was to add new fuel types and damaged fuel to the list of authorized contents for the Standardized NUHOMS®-61BT System. In addition, three minor changes to the Technical Specifications were requested to correct inconsistencies and remove irrelevant references.

The Nuclear Regulatory Commission (NRC) staff has reviewed the application using the guidance provided in NUREG-1536, "Standard Review Plan for Dry Cask Storage Systems," (SRP) January 1997. Only those SRP chapters with a corresponding applicant request for revision or change are addressed in the NRC staff's safety evaluation report. The staff did not review any changes performed, as authorized under the provisions of 10 CFR 72.48, associated with the Standardized NUHOMS®-61BT System. Based on the statements and representations in the application, as supplemented, the staff concludes that the TN NUHOMS®-61BT System, as amended, meets the requirements of 10 CFR Part 72. The proposed Amendment No. 7 changes to the CoC are indicated by double change bars in the margin. The pending changes for proposed Amendments No. 5 and No. 6 to the TN Standardized NUHOMS® System are indicated by single change bars in the margin.

1.0 GENERAL DESCRIPTION EVALUATION

This Safety Evaluation Report (SER) is issued in response to a request by Transnuclear Inc. (TN) to amend Certificate of Compliance (CoC) No. 1004 to add new Boiling Water Reactor (BWR) spent nuclear fuel types and damaged fuel to the Standardized NUHOMS® Horizontal Modular Storage System for Irradiated Nuclear Fuel 61BT Dry Shielded Canister (DSC) design.

TN performed a series of thermal, shielding, criticality, and source term analyses for the BWR fuel types to be added to the list of approved contents for the NUHOMS®-61BT DSC. Review of these analyses has shown that the design remains below all regulatory limits with the new fuel types and addition of damaged fuel to the list of approved contents. This SER specifically discusses the evaluation of analyses in support of the CoC amendment request and impacts on safety due to the proposed new fuel types and damaged fuel.

The revision to the list of approved contents, as provided in the CoC amendment request, has a minimal effect on the characteristics and operations of the licensed NUHOMS® system. Only characteristics and operations unique to the CoC amendment request are discussed in this SER. The CoC amendment request is a supplemental document. It must be considered along with information provided in the NUHOMS® Final Safety Analysis Report (FSAR). All references to the CoC amendment request in this SER assume a reference to the NUHOMS® FSAR. This review was performed in accordance with NUREG-1536, "Standard Review Plan for Dry Cask Storage Systems."

1.1 DCSS Description and Operational Features

The dry cask storage system (DCSS) components affected by this CoC amendment request primarily focus on the NUHOMS®-61BT DSC. The description is similar to that provided in the NUHOMS® FSAR for the DSC. Details of differences to accommodate the new fuel types and damaged fuel are described in Section K.1.2.1 of the amended FSAR.

1.2 Drawings

Drawings for structures, systems, and components (SSCs) important to safety are presented by the applicant in support of the CoC amendment request and contain the correct information with sufficient level of detail to illustrate the design, their analyses, and for the staff to perform a review and make findings and evaluate SSCs effectiveness.

1.3 DCSS Contents

The fuel specifications for the BWR fuel contents of the NUHOMS®-61BT DSC have been revised. Characteristics of the new fuel types and damaged fuel to be stored are presented in revision pages to the NUHOMS® FSAR.

1.4 Qualifications of the Applicant

The owner of the NUHOMS® design is TN. All design and specification activities, including quality assurance (QA) services, were performed by TN. The NUHOMS® FSAR states that fabrication of the DSC will be performed by a qualified steel fabrication shop. TN will retain full responsibility for control over design, analysis, and fabrication activities.

1.5 Quality Assurance

The applicant's QA program is identical to that approved for the NUHOMS® FSAR.

1.6 Evaluation Findings

The NRC staff makes the following findings regarding the general description of the Standardized NUHOMS® System design:

F1.1 A general description and discussion of the new fuel types and damaged fuel are presented in Attachment A of the CoC amendment request and, by reference, the NUHOMS FSAR, with special attention to design and operating characteristics, unusual or novel design features, and principle safety considerations.

F1.2 New drawings for SSCs important to safety are presented in the CoC amendment request. Specific SSCs are evaluated in Chapters 3 through 14 of this SER.

F1.3 Specifications for the spent fuel to be stored in the DCSS are provided in the CoC amendment request, Attachment B.

F1.4 The technical qualifications of the applicant to engage in the proposed activities are identical to those of the NUHOMS® FSAR previously found acceptable to the NRC staff and approved in Amendment No. 3 to the CoC.

F1.5 The QA program and implementing procedures are identical to those of the NUHOMS® FSAR previously found acceptable to the NRC staff and approved in Amendment No. 3 to the CoC.

F1.6 The staff concludes that the information presented in the CoC amendment request and, by reference, the NUHOMS® FSAR, satisfies the requirements for the general description under 10 CFR Part 72. This finding is based on a review that considered the regulation itself, Regulatory Guide 3.61, and accepted dry cask storage practices detailed in NUREG-1536.

1.7 References

U.S. Nuclear Regulatory Commission, Certificate of Compliance No. 1004, "Standardized NUHOMS® Horizontal Modular Storage System for Irradiated Nuclear Fuel," January 23, 1995, as amended February 12, 2002.

Transnuclear, Inc., "Application for Amendment No. 7 of NUHOMS® Certificate of Compliance No. 1004 for Dry Spent Fuel Storage Casks," Revision 0, March 29, 2002.

Transnuclear, Inc., “Response to Request for Additional Information (RAI) and Submittal of Revision 1 of Application for Amendment No. 7 of NUHOMS® Certificate of Compliance No. 1004 (TAC NO. L23436),” February 14, 2003.

Transnuclear West, Inc., “Final Safety Analysis Report for the Standardized NUHOMS® Horizontal Modular Storage System for Irradiated Nuclear Fuel,” NUH-003, Revision 6, October, 2001.

U. S. Nuclear Regulatory Commission, NUREG-1536, “Standard Review Plan for Dry Cask Storage Systems,” January 1997.

U.S. Code of Federal Regulations, Title 10, Part 72, “Licensing Requirements for the Independent Storage of Spent Nuclear Fuel and High-Level Radioactive Waste.”

U.S. Nuclear Regulatory Commission, Regulatory Guide 3.61, “Standard Format and Content for a Safety Analysis Report for a Spent Fuel Storage Cask,” February 1989.

2.0 PRINCIPLE DESIGN CRITERIA EVALUATION

The objectives of evaluating the principle design criteria are to ensure that they: (1) adequately define the limiting characteristics of the materials to be stored, (2) classify structures, systems and components (SSCs) according to their importance to safety, and (3) comply with the relevant general criteria established in 10 CFR Part 72.

The applicant has presented general details of the principle design criteria changes in Attachment C of the CoC amendment request and provided appropriate details in the associated sections of the CoC amendment request.

This review was performed in accordance with the NUREG-1536, "Standard Review Plan for Dry Cask Storage Systems."

2.1 Structures, Systems, and Components Important to Safety

With the exception of additional hardware needed to accommodate damaged fuel assemblies, the SSCs important to safety in the CoC amendment request are the same as those identified in the NUHOMS® FSAR. Those components that are unique to this application have been adequately described in the CoC amendment request and their relationship to performance of the safety functions and how they meet the general design criteria sufficiently characterized.

2.2 Design Bases for Structures, Systems, and Components Important to Safety

2.2.1 Spent Fuel Specifications

NRC regulations defining the design criteria requirements for materials to be stored are given in 10 CFR 72.2, 72.6, and 72.120. The application requests an amendment to CoC No. 1004 to revise the fuel specifications for the storage of BWR fuel types in the NUHOMS®-61BT DSC.

The staff concluded that the CoC amendment request provides adequate design criteria to characterize the spent fuel and satisfies the requirements of 10 CFR 72.2(a) and 72.120(b).

2.2.2 External Conditions

Normal, off-normal, accident, and natural phenomena design bases are the same as those evaluated for the NUHOMS FSAR.

2.3 Design Criteria for Safety Protection Systems

NRC regulations defining the design criteria requirements for SSCs important to safety are given in 10 CFR 72.24, 72.102, 72.104, 72.106, 72.120, 72.122, 72.124, 72.126, 72.128, 72.130, 72.144, 72.182, and 72.236. The principle design criteria for normal, off-normal, and accident design conditions include environmental conditions and natural phenomena. Principle design criteria are the same as those evaluated for the NUHOMS® FSAR.

2.4 Evaluation Findings

The NRC staff makes the following findings regarding the SSCs and design evaluation of the CoC amendment request:

F2.1 The CoC amendment request, docketed materials, and, by reference, the NUHOMS® FSAR, adequately identify and characterize the spent fuel to be stored in the NUHOMS®-61BT DSC and demonstrate that the stored materials are in conformance with the requirements given in 10 CFR Part 72. The staff concludes that the principal design criteria for the NUHOMS®-61BT DSC are acceptable for meeting the regulatory requirements of 10 CFR Part 72.

2.5 References

U.S. Code of Federal Regulations, Title 10, Part 72, "Licensing Requirements for the Independent Storage of Spent Nuclear Fuel and High-Level Radioactive Waste."

Transnuclear, Inc., "Application for Amendment No. 7 of NUHOMS® Certificate of Compliance No. 1004 for Dry Spent Fuel Storage Casks," Revision 0, March 29, 2002.

Transnuclear, Inc., "Response to Request for Additional Information (RAI) and Submittal of Revision 1 of Application for Amendment No. 7 of NUHOMS® Certificate of Compliance No. 1004 (TAC NO. L23436)," February 14, 2003.

U. S. Nuclear Regulatory Commission, NUREG-1536, "Standard Review Plan for Dry Cask Storage Systems," January 1997.

Transnuclear West, Inc., "Final Safety Analysis Report for the Standardized NUHOMS® Horizontal Modular Storage System for Irradiated Nuclear Fuel," NUH-003, Revision 6, October, 2001.

American Society of Mechanical Engineers, "Boiler and Pressure Vessel Code, Section III, Division 1, Subsection NB, NC, NF, NG and Appendices, 1998 with 199 Addenda."

3.0 STRUCTURAL EVALUATION

The staff reviewed the revised structural and materials analyses to ensure that the NUHOMS®-61BT cask system with additional contents continue to satisfy the structural and materials requirements of 10 CFR Part 72. The application proposes to add damaged BWR spent fuel assemblies and additional fuel types to the previously approved NUHOMS®-61BT DSC. The NUHOMS®-61BT DSC is designed to store 61 intact BWR fuel assemblies or a combination of up to 16 damaged fuel assemblies and the remainder intact fuel assemblies. The NUHOMS®-61BT DSC has three basket configurations (types A, B, and C) based on the boron content in the poison plates. Intact BWR fuel assemblies may be stored in any of the three NUHOMS®-61BT types, while damaged fuel assemblies may only be stored in Type C baskets with end caps installed on each of the sixteen fuel compartments where damaged fuel assemblies are stored. In support of the application, the applicant provided structural analyses for determining the effects of stresses on the fuel cladding associated with normal and off-normal storage conditions. The applicant also addressed the additional fuel types to be loaded into the NUHOMS®-61BT system.

Since this amendment only adds damaged spent fuel assemblies and additional fuel types, it is basically consistent with the NUHOMS®-61BT system which has been previously reviewed and approved. Therefore, the structural review was focused on the damaged fuel and additional intact fuel types. The review was conducted against the appropriate Part 72 regulations and industry standards and codes to ensure safe nuclear fuel storage. Review procedures are consistent with NUREG-1536, "Standard Review Plan for Dry Cask Storage Systems." Applicable structural calculations and analyses for damaged fuel were reviewed. Additional fuel types were compared against the previous fuel contents of the NUHOMS®-61BT system to ensure that the analyses reviewed and approved previously are also applicable to the BWR fuel types proposed for this amendment.

3.1 Structural Design Criteria

3.1.1 Cask Design Criteria

The NUHOMS®-61BT DSC (shell and closure including the damaged spent fuel assemblies and additional fuel types) is designed and fabricated based on the ASME Boiler and Pressure Vessel Code, Section III, Subsection NB, with exceptions as listed in FSAR Section K.3.1.2.3. Fabrication and testing conforms to the criteria in Subpart G of 10 CFR 72.

3.2 Structural Analysis

3.2.1 Additional fuel types

The additional fuel types to be added to the NUHOMS®-61BT DSC include the GE1(7x7), ENCIII-A-(7x7), ENC-III(7x7), ENC Va(8x8), and ENC Vb(8x8). Table 1-1d of the Technical Specifications provides BWR fuel assembly design characteristics of the proposed additional types to be stored in the standardized NUHOMS®-61BT DSC. Table 1-1c provides BWR fuel specification of all intact fuel to be stored in the NUHOMS®-61BT DSC. Both tables specify the fuel physical parameter requirements for the fuel assembly, i.e., the maximum assembly length is 176.2 inches; and the maximum assembly weight with channels is 705 lbs. Therefore, the

structural analyses for intact fuel types are applicable to the additional fuel types pertaining to normal, off normal, and accident conditions, the resulting stresses due to load combinations, and the safety margins for the fuel cladding under all load cases.

3.2.2 Damaged Fuel

The damaged fuel types are all 7x7 and 8x8 configurations which include the GE1, GE2, GE3, GE4, GE5, GE-pres, GE-Barrier, GE8 Type I, GE8 Type II, GE9, GE10, ENC III-A, ENC III, ENC Va and ENC Vb.(Table 1-1d)

Evaluations were performed to demonstrate structural integrity for the above damaged fuels under normal and off normal load conditions. A linear-elastic stress analysis and linear fracture mechanics methods were employed to evaluate fuel cladding integrity considering BWR fuel with 7x7 and 8x8 arrays. A cladding thickness reduction of 200 μm was assumed in the analysis. This is a conservative assumption for fuel with maximum burnup of 40 GWD/MTU to account for oxidation.

As shown in Table K.3.6-6, the controlling normal load is the handling/transfer load. The controlling off normal load is the jammed canister load. The respective safety margins are shown as the ratio of maximum cladding computed stresses to the yield stress of the zircaloy cladding material.

In addition, one foot end drop and one foot side drop analyses at 30g were also performed. The one foot side drop (for 8x8 fuel) yielded the maximum cladding stresses among all load conditions. Yet the ratio of the computed versus the yield stresses is <0.17, which signifies a large safety margin for normal and off normal loads.

3.2.3 Stability of the fuel tube cladding

The hypothetical one foot end drop shown in Table K.3.6-6 resulted in the maximum 7x7 fuel rod cladding compressive stresses of 10,973 psi, equivalent to a bounding compressive load of 441 lbs. The associated deceleration value for the one foot end drop is 30g. This is much less than the lower bound buckling capacity load using the buckling g load of 128g. Thus, stability of the fuel tube cladding is not a problem.

3.3. Materials Review

The staff reviewed materials issues considering addition of damaged fuel to the list of approved cask contents. Damaged fuel has previously been approved for storage in other NUHOMS® cask models. When damaged fuel is loaded into a NUHOMS®-61BT DSC, the damaged fuel bundle is confined with the addition of end caps. An end cap is a stainless steel cap installed at either end of the basket cell into which the damaged fuel bundle is loaded. The end caps close the ends of the cell and thereby isolate that cell from all the others of the fuel basket. This prevents any loose particulates from the damaged fuel from migrating into any other cell, thus alleviating criticality, thermal, or shielding concerns that could result from such an occurrence.

This method of isolating a damaged fuel assembly from the rest of the casks contents is acceptable for this cask design. However, it potentially has operational disadvantages when

compared to the method of enclosing a damaged fuel assembly in a separate damaged fuel can prior to inserting the loaded damaged fuel can into the fuel basket. A damaged fuel can allows retrieval by conventional fuel handling methods, whereas, the method employed for the NUHOMS®-61BT could require vacuum retrieval of any loose particles, should there be any, if the cask is ever unloaded. The NRC staff issued Interim Staff Guidance -1 (ISG-1), dated October 2, 1998, which defined damaged fuel as: “Spent nuclear fuel with known or suspected cladding defects greater than a hairline crack or a pinhole leak.” This definition applies to both storage and transportation.

The NRC definition of damaged fuel was extensively expanded and revised in ISG-1, Revision 1, dated October 25, 2002. ISG-1, Revision 1, added guidance for consideration of structural defects, whereas the previous definition was solely concerned with cladding confinement breaches. Another major change was the recognition of the differing design conditions for storage versus transportation. The conditions under which fuel is expected to remain intact under storage design conditions differ from those of transportation. Due to this difference in design requirements, the NRC staff provided an option of classifying fuel as damaged or intact for storage only situations versus dual purpose or transportation conditions.

This implies that spent fuel structural integrity may be evaluated against design storage conditions and classified as intact or damaged as appropriate without consideration of transportation conditions. This approach recognized the existence of several storage-only cask designs, which could not be expected to consider or comply with transportation design requirements and the associated loads thereby imposed upon the fuel.

For this amendment, which affects storage only, the applicant clarified that they would adhere to the earlier (October 2, 1998, ISG-1) definition of damaged fuel. The applicant further stated that a separate amendment would be submitted in the future for transportation. Adherence to the original version of ISG-1 is permissible. Consequently, the types of fuel defects that are recognized as damaged fuel under this amendment request are limited compared to the revised definition of damaged fuel. Such action carries a potential future burden of proof to demonstrate that any fuel loaded for storage is structurally suitable for transportation under the hypothetical accident conditions of transportation.

In summary, TN’s approach to the classification of damaged fuel is acceptable to the staff. However, the staff notes that it potentially places a future burden upon a cask owner to re-validate, in accordance with the latest version of ISG-1, the structural, thermal, dose, and criticality safety of any fuel that is to be transported under 10 CFR Part 71.

3.4 Conclusion

The staff concludes that the structural design of the damaged spent fuel types (see 3.2.2) and the additional intact fuel types (see 3.2.1) are in compliance with 10 CFR Part 72 and that the applicable design and acceptance criteria have been satisfied. The structural evaluation provides reasonable assurance that the proposed damaged fuel and additional fuel types are acceptable contents for loading in the NUHOMS®-61BT DSC. This finding is based on staff review that considered NUREG-1536, appropriate regulatory guides, applicable codes and standards, and satisfactory response to the staff RAI.

3.5 Evaluation Findings

The NRC staff makes the following findings regarding the structural evaluation of the CoC amendment request:

F3.1 The FSAR adequately describes all systems, structures, and components (SSC) that are important to safety, providing drawings and text in sufficient detail to allow evaluation of their structural effectiveness.

F3.2 The applicant has met the requirements of 10 CFR 72.24, "Contents of Application: Technical Information," with regard to information pertinent to structural evaluation.

F3.3 The applicant has met the requirements of 10 CFR 72.122(b) and (c) and 10 CFR 72.24(c)(3). The structures, systems, and components important to safety are designed to accommodate the combined loads of normal, off-normal, accident, and natural phenomena events with an adequate margin of safety. Stresses at various locations of the cask for various design loads are determined by analysis. Total stresses for the combined loads of normal, off-normal, accident, and natural phenomena events are acceptable and are found to be within limits of applicable codes, standards, and specifications.

3.6 References

Transnuclear, Inc., "Application for Amendment No. 7 of NUHOMS® Certificate of Compliance No. 1004 for Dry Spent Fuel Storage Casks," Revision 0, dated March 29, 2002.

Title 10 CFR Part 72, Subparts K and L.

U. S. Nuclear Regulatory Commission, NUREG-1536, "Standard Review Plan for Dry Cask Storage Systems," January 1997.

American Society of Mechanical Engineers, ASME Boiler and Pressure Vessel Code, Section III, Subsection NB, NC, NF, NG and Appendices, 1998 Edition with 1999 Addenda.

Transnuclear West, Inc., "Final Safety Analysis Report for the Standardized NUHOMS® Horizontal Modular Storage System for Irradiated Nuclear Fuel," NUH-003, Revision 6, October, 2001.

Transnuclear, Inc., "Response to Request for Additional Information (RAI) and Submittal of Revision 1 of Application for Amendment No. 7 of NUHOMS® Certificate of Compliance No. 1004 (TAC NO. L23436)," February 14, 2003.

4.0 THERMAL EVALUATION

The staff reviewed the revised thermal analyses and additional contents to ensure that the cask component and fuel material temperatures for the NUHOMS®-61BT cask system will continue to remain within the allowable values or criteria for normal, off-normal, and accident conditions. The staff has previously approved storage of intact BWR fuel assemblies in the NUHOMS®-61BT DSC for decay heat loads of up to 0.3 kW per assembly and up to a total of 18.3 kW per DSC. The applicant desires to add damaged fuel assemblies (DFAs) to the contents of the NUHOMS®-61BT DSC. The applicant proposes to store up to 16 DFAs in the four 2x2 corner fuel compartments of the NUHOMS®-61BT DSC. The applicant has also proposed adding additional intact fuel types including the GE1(7x7), ENCIII-A(7x7), ENC-III(7x7), ENC Va(8x8), and ENC Vb(8x8) to the approved contents of the NUHOMS®-61BT DSC. Physical descriptions of these additional fuel assembly types are provided in Table 1-1d of the TS.

The applicant reviewed two particular fuel assembly designs for storage as damaged fuel: the General Electric (GE) 2 (7x7-49/0) and 4 (8x8-63-1) fuel types, hereafter referred to as GE2 and GE4. These fuel types provided the lowest transverse thermal conductivity in the undamaged condition for temperatures up to 400°F (for the GE2) and over 400°F (for the GE4). The applicant calculated the effective thermal conductivity for two damaged fuel conditions per fuel assembly design. The first condition accounts for a reconfiguration of the DFA such that the fuel rods have moved in the axial and/or transverse (radial) directions, while the second condition accounts for reconfiguration of the fuel pellets into “rubble” at the bottom of the fuel compartment containing the DFA.

4.1 Calculation of Effective Thermal Conductivities

4.1.1 Reconfiguration of Fuel Rods

For fuel assembly conductivity in the axial direction (along the length of the assembly), movement of fuel rods in the axial direction or transverse direction has no effect, therefore the values for axial conductivity from an undamaged assembly were used.

For fuel assembly conductivity in the transverse direction (radially outward from the center of the basket), movement of fuel rods in the axial direction has no effect; however, movement of fuel rods in the transverse direction will effect the conductivity of the assembly.

The applicant modeled transverse movement of the fuel rods by changing the pitch size of the fuel rods in the GE2 and GE4 assemblies. The smallest pitch size analyzed represented a distance of 0.01 inches between fuel rods. The assembly with the lowest thermal conductivity was determined by the applicant to be the GE2 fuel assembly type with a pitch of 0.607 inches. This assembly yielded effective conductivity values listed in Section K.4.8.1.3 of the FSAR. These values are slightly less than the values of an intact assembly.

4.1.2 Fuel Rubble Configuration

The applicant examined the thermal effects of fuel rubble in the bottom of the DFA compartment. The applicant modeled the fuel rubble as a homogenized region with the same effective thermal conductivity in both the axial and transverse directions. The applicant

calculated the height of the fuel rubble region by dividing the combined volumes of the uranium dioxide (UO_2) fuel pellets, the Zircally fuel cladding, and the helium fill gas and dividing that value by the cross-sectional area of the compartment. The applicant determined that the GE-8 Type 2 fuel assembly provided the shortest height and thus was considered the bounding assembly for this configuration.

The applicant calculated the effective conductivity of the rubble region neglecting the conductivity of both the Zircaloy cladding and the helium fill gas. The conductivity used for UO_2 was 0.1926 BTU/hr-in-°F, and the conductivity calculated for the rubble region was 0.12 BTU/hr-in-°F. The applicant's calculation is detailed in Section K.4.8.1.2.

4.2 Calculation of Component Temperatures

The applicant used the existing DSC analysis models (described in Section K.4.4 of the NUHOMS®-61BT DSC FSAR), created in the ANSYS® finite element analysis code, to evaluate the basket and fuel temperatures for storage of DFAs. The values calculated for the effective conductivities of both reconfigured fuel and fuel rubble were applied to the model to determine component temperatures. The applicant provided results for a normal condition storage case, with the DSC in the (HSM) storage module and an ambient of 100°F. Temperature results were reported in Table K.4-7 in the FSAR. None of the components for which the applicant reported temperatures exceeded their material temperature limits.

4.3 Staff Review and Conclusions

The staff reviewed the application and all the calculations submitted by the applicant. Though ANSYS models were described in the application, the actual analysis files generated were not submitted for review. Given the nature of storing DFAs in the NUHOMS®-61BT DSC, and the existing analyses performed by the staff, additional confirmatory thermal analyses were not performed. The additional intact fuel assembly types requested by the applicant are within the maximum decay heat limits specified in Table 1-1c of the Technical Specifications and thus are bounded by the thermal analysis conducted for the original NUHOMS®-61BT DSC design. The staff concludes that there is reasonable assurance that the storage of additional fuel assembly types as well as DFAs in the NUHOMS®-61BT DSC design, as described in the applicant's submittal, is acceptable.

4.4 Evaluation Findings

The NRC staff makes the following findings regarding the thermal evaluation of the CoC amendment request:

F4.1 The NUHOMS® system, in part comprised of the NUHOMS®-61BT DSC, is designed to provide adequate heat removal capability when loaded with damaged fuel assemblies, as described in the FSAR, without active cooling systems.

F4.2 The staff concludes that the thermal design of the NUHOMS® system, in part comprised of the NUHOMS®-61BT DSC, is in compliance with 10 CFR Part 72, and that the applicable design and acceptance criteria have been satisfied. The evaluation of the thermal design provides reasonable assurance that the NUHOMS® system 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.

4.5 References

U.S. Code of Federal Regulations, Licensing Requirements for the Independent Storage of Spent Nuclear Fuel and High-Level Radioactive Waste, Title 10, Part 72.

U. S. Nuclear Regulatory Commission, NUREG-1536, “Standard Review Plan for Dry Cask Storage Systems,” January 1997.

5.0 SHIELDING EVALUATION

The staff evaluated the capability of the NUHOMS®-61BT DSC to provide adequate protection against direct radiation from the additional contents when used with the Standardized NUHOMS® System. The regulatory requirements for providing adequate radiation protection to licensee personnel and members of the public include 10 CFR Part 20 and 10 CFR 72.104, 72.106(b), 72.212, and 72.236(d).

The applicant requested additional fuel types and the inclusion of damaged fuel be added to the list of approved contents for the NUHOMS®-61BT DSC. The staff has previously approved the shielding evaluation, in Amendment No. 3 to the Certificate of Compliance (CoC), for the loading of damaged fuel. The applicant requested a new fuel qualification table for the currently approved fuel to allow additional combinations of maximum burnup, minimum cooling time, and minimum enrichment for the currently approved and new fuel. The applicant used a dose response function methodology (i.e., source strength to dose rate conversion factors) to determine alternate burnup, cooling time, and enrichment parameters for the spent nuclear fuel. The fuel parameters calculated with the dose response methodology result in dose rates and individual heat loads that are less than, or equal to, the maximum bounding dose and thermal limits derived from the calculations previously approved in Amendment No. 3 to the CoC. The applicant used the ANISN shielding code to calculate dose response functions and ORIGEN-2 to generate alternate source terms.

5.1 Shielding Design Features and Design Criteria

5.1.1 Shielding Design Features

The applicant did not request any changes to the shielding design features for the NUHOMS®-61BT DSC under this amendment request. The shielding design features for the NUHOMS®-61BT DSC have been approved in Amendment No. 3 to the CoC. The staff did not review any changes to the design features of the NUHOMS® dry cask storage system, the Horizontal Storage Module (HSM) or the Transfer Cask (TC), or any other changes made to the FSAR, performed under the change authority requirements of 10 CFR 72.48. Based on this information, the applicant’s shielding design provides reasonable assurance that the shielding design features of the NUHOMS®-61BT DSC can meet the radiological requirements of 10 CFR Part 20 and 10 CFR Part 72.

5.1.2 Shielding and Source Term Design Criteria

The overall radiological protection design criteria are the regulatory dose requirements in 10 CFR Part 20 and 10 CFR 72.104, 72.106(b), 72.212, and 72.236(d). The applicant analyzed the NUHOMS®-61BT with additional contents as described in Section K.5 of the amendment request. The amendment also specified methodological dose limit criteria used to determine the acceptable burnup, cooling time, and enrichment parameters. These include maximum dose limits of 497.2 mrem/hr on the radial surface of the TC and 29.60 mrem/hr on the roof surface of the HSM, calculated with the ANISN response function methodology. In addition, the heat load limit of 0.3 kW for each fuel assembly serve as additional source term design criteria, which in turn limits overall dose rates. Based on these design criteria, the applicant determined

alternate burnup, cooling time, and enrichment parameters that are equal to or less than the bounding dose rates approved in Amendment No. 3 to the CoC.

The staff reviewed the design criteria and found it acceptable. The shielding design criteria defined in the FSAR provides reasonable assurance that the NUHOMS®-61BT can meet the radiological requirements of 10 CFR Part 20 and 10 CFR Part 72. Dose rate limits based on the bounding shielding analysis are incorporated into technical specifications for the outside of the Standardized HSM door, the front vents, and end shield wall exterior of the HSM.

5.2 Source Specification

The applicant previously calculated design-basis source terms to perform bounding shielding calculations with DORT in Amendment No. 3 to the CoC. The GE 7x7 was chosen as the design basis fuel assembly because it had the highest initial heavy metal loading (0.198 MTU). The applicant also calculated multiple source terms for alternate burnup, cooling time, and enrichment combinations in Table K.2-11. The gamma and neutron source term calculations were performed with the ORIGEN-2 and are similar to the calculations previously approved for the design-basis source terms. The applicant determined the gamma and neutron source terms for each fuel combination in a similar manner previously approved in Amendment No. 3 to the CoC. The gamma source term included contribution from the fuel channel and fuel hardware to determine the fuel qualification tables. The applicant verified each combination was within the 0.3 kW heat limit. In addition, the applicant used these source terms with the ANISN dose response functions to verify that each combination resulted in dose rates bounded by design-basis dose rates.

The staff reviewed the updated source term analyses in Chapter K.5 of the FSAR. Based on information provided in the amendment request, the staff has reasonable assurance that the source terms for the additional fuel types requested in the amendment are bounded by the design-basis source terms. The staff has reasonable assurance that the gamma and neutron source terms used for the fuel qualification table are acceptable for the shielding analysis. The source term method used is similar to the method previously approved in Amendment No. 3 of the CoC.

5.3 Shielding Model Specifications

The shielding analysis for bounding dose rates was performed with DORT, a 2-D discrete ordinates code used to calculate the dose rates on and around the HSM and TC. To determine the total off-site dose, the MCNP computer code was used. The off-site dose models include 1) a 2x10 array of HSMs and, 2) two 1x10 arrays (facing front-to-front) loaded with design basis fuel in NUHOMS®-61BT DSCs. The applicant used ANISN, a one-dimensional discrete ordinates code, to determine dose response functions for the HSM and TC. The applicant indicated this shielding methodology was similar to the methodology previously approved for the NUHOMS®-24P and 52B canister. The applicant stated that the response function is used to account for the substantial shift in the gamma spectrum over the range of the burnup, enrichment and cooling time combinations.

The shielding models consist of one-dimensional representations of the HSM and TC, including the spent nuclear fuel source and NUHOMS®-61BT canister regions. These models are depicted in Figures K.5-16 and K.5-17 of the amendment request. The response functions were generated with the CASK-81, 22 neutron, 18 gamma couple cross-section library. The gamma (by energy group) and neutron dose response functions for both the HSM and TC are listed in Table K.5-20 of the amendment request. The applicant provided a sample ANISN input file in Section K.5.5.4 of the amendment request.

5.3.1 Shielding and Source Configuration

The shielding source is divided into 12 axial regions as summarized in Table K.5-14. The top and bottom 10% of the assembly is divided into two zones each and the remaining 80% is divided into eight equal zones. Volumetric sources are developed for all fuel regions. The source is divided into the following regions; active fuel, bottom end fitting, and top end fitting. The axial distribution of the gamma and neutron sources is assumed to follow the relative burnup profile depicted in Figure K.5-1. The fuel channel material and most of the basket materials are conservatively neglected in the shielding model which reduces the amount of actual shielding and results in a bounding dose rate. A number of other simplifications and bounding assumptions that reduce the amount of actual shielding are discussed in Section K.5.4.

The analysis includes streaming paths through the HSM air vents and the TC-DSC gap. The overall design eliminated other potential streaming paths. Evaluation of streaming from narrow and long holes is difficult for DORT. While DORT is subject to ray effects, this tends to over-predict radiation streaming.

5.3.2 Material Properties

The composition and densities of the materials used in the shielding analysis are presented in Tables K.5-15 through K.5-19 of the FSAR. The applicant stated the material densities used in the ANISN models are listed in Table K.5-19 of the FSAR. The homogenized fuel assembly region accounts for the uranium dioxide; zircaloy cladding and spacers; and steel present in the in-core region of the assembly, the basket inner fuel compartment, and the outer wrapper materials. The BWR fuel channels and all other components of the basket were ignored. Material properties were previously reviewed and accepted by the staff in Amendment No. 3 to the CoC. The staff did not review any changes to the material properties used in the NUHOMS® dry cask storage system, the HSM or the TC, or changes made to the FSAR, performed under the change authority requirements of 10 CFR 72.48.

5.3.3 Staff Evaluation

The staff evaluated the shielding models and found them acceptable. The shielding model and configuration has been previously approved in Amendment No. 3 to the CoC. The material compositions and densities used were appropriate and provide reasonable assurance that the NUHOMS®-61BT cask system was adequately modeled with the ANISN dose models. The staff notes that use of the ANISN 1-D model to represent the 3-D NUHOMS®-61BT shielding system results in additional uncertainties. However, the use of ANISN in the shielding analysis is essentially limited to evaluating the relative changes in dose rates versus relative changes in

source terms for alternate combinations of burnup, cooling time, and enrichment. The staff finds the use of ANISN acceptable for this specific design and contents for the following reasons: (1) higher energy gamma source terms dominate public dose rates and any ANISN-related uncertainties should be relatively systematic for each fuel combination; (2) the use of ANISN is consistent with the methodology previous used for the NUHOMS®-24P and 52B canisters; (3) the staff has incorporated specific dose rate limits in the Technical Specifications for the storage module; and (4) the general licensee will operate the NUHOMS®-61BT storage system with an established radiation protection program as required by 10 CFR Part 20, Subpart B.

5.4 Shielding Analyses

5.4.1 Computer Programs

The applicant's shielding analysis for bounding dose rates was previously performed with DORT and is presented in Section K.5.4 of the FSAR. The cross section data used are based on the CASK-81 22 neutron, 18 gamma energy group coupled cross section library. The dose response functions were calculated using ANISN.

5.4.2 Flux-to-Dose-Rate Conversion

The FSAR uses the ANSI/ANS Standard 6.1.1-1977 flux-to-dose rate conversion factors to calculate dose rates.

5.4.3 Normal Conditions

The applicant did not change the dose calculations for normal conditions because the design-basis source terms did not change as a result of the additional fuel types and alternate fuel parameters. The normal condition dose rates have been approved in Amendment No. 3 to the CoC. The Appendix K of the FSAR presents calculated dose rates for normal condition design-basis dose rates for the TC and the HSM in Tables K.5-2 and K.5-3.

5.4.4 Accident Conditions

The applicant did not change the dose calculations for accident conditions because the design-basis source terms did not change as a result of the additional fuel types and alternate fuel parameters. The accident condition dose rates have been approved in Amendment No. 3 to the CoC.   Table K.11-1 and K.11-4 of the FSAR show the maximum dose rates for accident conditions.

5.4.5 Occupational Exposures

The applicant did not change the dose calculations because the design-basis source terms did not change as a result of the additional fuel types and alternate fuel parameters.   Chapter K.10 of the FSAR presents the estimated occupational exposures that are based on dose rate calculations in Chapter K.5 to the FSAR.

5.4.6 Off-site Dose Calculations

The applicant did not change the dose calculations because the design-basis source terms did not change as a result of the additional fuel types and alternate fuel parameters. Section K.10 of the FSAR estimates the offsite dose rates from a 2x10 and two 1x10 arrays.

The staff has reasonable assurance that compliance with 10 CFR 72.104(a) can be achieved by general licensees. The general licensee must perform a site-specific evaluation, as required by 10 CFR 72.212(b), to demonstrate compliance . The actual doses to individuals beyond the controlled area boundary depend on several site specific conditions such as fuel characteristic, cask-array configurations, topography, demographics, and atmospheric conditions. In addition, 10 CFR 72.104(a) includes 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 the general licensee. A general licensee will also have an established 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, by evaluation and measurements.

5.5 Evaluation Findings

The NRC staff makes the following findings regarding the shielding evaluation of the CoC amendment request:

F5.1 Appendix K of the FSAR sufficiently describes the radiation protection design bases and design criteria for the structures, systems, and components important to safety.

F5.2 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.

F5.3 The NUHOMS®-61BT DSC is designed to provide redundant sealing of the confinement system.

F5.4 The staff concludes that the design of the radiation protection system of the NUHOMS®-61BT DSC, when used with the HSM, is in compliance with 10 CFR Part 72 and the applicable design and acceptance criteria have been satisfied. The evaluation of the radiation protection system design provides reasonable assurance that the NUHOMS®-61BT DSC will provide safe storage of intact and damaged BWR spent fuel. This finding is based on statements and information provided by the applicant, previous staff approval of Amendment No. 3 to the CoC, the regulation itself, the appropriate regulatory guides, and applicable codes and standards.

5.6 References

U.S. Code of Federal Regulations, Licensing Requirements for the Independent Storage of Spent Nuclear Fuel and High-Level Radioactive Waste, Title 10, Part 72.

U.S. Code of Federal Regulations, Standards for Protection Against Radiation, Title10, Part 20.

U. S. Nuclear Regulatory Commission, NUREG-1536, “Standard Review Plan for Dry Cask Storage Systems,” January 1997.

6.0 CRITICALITY EVALUATION

The staff reviewed the updated criticality analyses in the amendment request for the NUHOMS®-61BT cask system to ensure that the new contents remain subcritical under all credible normal, off-normal, and accident conditions, in accordance with 10 CFR Part 72.

6.1 Criticality Design and Features

The applicant did not request any changes to the criticality design and features for the NUHOMS®-61BT cask system. The criticality design and features have been approved in Amendment No. 3 to the CoC.

6.2 Fuel Specification

The applicant requested approval of the following new fuel versions: GE1 7x7, ENCIII-A (Exxon/ANF) 7x7, ENCIII 7x7, ENCIII-A 7x7, ENC Va 8x8, and ENC Vb 8x8 BWR fuel, as specified in Table K.6-3 of the amendment request. The ENC III, ENC Va, and ENC Vb fuel contain solid Zirc rods instead of water holes. The applicant requested approval of these new versions for both intact and damaged fuel configurations. The applicant also requested approval of all other 7x7 and 8x8 fuel versions listed in Table K.6-3 of the amendment request for the damaged fuel configuration.

The damaged fuel configuration for 7x7 and 8x8 fuel has been previously approved in Amendment No. 3 to the CoC, except for the new fuel versions discussed above. The staff reviewed the fuel types considered in the criticality analyses and verified that they bound the new fuel types specified in Section K.2 of the amendment request. The staff verified that all new fuel assembly parameters important to criticality safety have been adequately presented.

6.3 Model Specification

The applicant used the same criticality model specification previously approved in Amendment No. 3 to the CoC for intact fuel and damaged fuel. The applicant performed additional most-reactive-fuel analyses for the new fuel versions in both the intact and damaged fuel configurations. Based on information presented in Table K.6-6, the applicant determined that the GE12 10x10 BWR intact fuel assembly and the GE2 7x7 and GE9 8x8 damaged fuel assembly remain the most reactive fuel type, as previously approved in Amendment No. 3 to the CoC.

The staff reviewed the information on the additional analyses provided by the applicant and found it acceptable. The method used for the most-reactive-fuel analyses appears to be the same method previously approved in Amendment No. 3 of the CoC. Because there is no change to the most reactive fuel type when considering addition of the new fuel types to the list of approved contents, the design-basis intact and damaged fuel assemblies and criticality model remain the same.

6.4 Criticality Analyses

The applicant did not request a change to the criticality analyses because the design-basis intact and damaged fuel assemblies and criticality model did not change as a result of the new fuel contents. The criticality analyses has been approved in Amendment No. 3 to the CoC.

6.5 Evaluation Findings

The NRC staff makes the following findings regarding the criticality evaluation of the CoC amendment request:

F6.1 Systems, structures, and components important to criticality safety have not changed and have been previously described in sufficient detail to enable evaluation of their effectiveness for the new contents.

F6.2 The NUHOMS®-61BT cask system is designed to be subcritical under all credible conditions.

F6.3 The criticality design is based on favorable geometry and fixed neutron poisons.

F6.4 For the new fuel contents, the staff concludes that the criticality design features for the NUHOMS®-61BT cask system are in compliance with 10 CFR Part 72, and the applicable design and acceptance criteria have been satisfied. The evaluation of the criticality design and new contents provides reasonable assurance that the NUHOMS®-61BT cask system will allow safe storage of the requested spent nuclear fuel. This finding is based on statements and information provided by the applicant, previous staff approval of Amendment No. 3 to the CoC, the regulation itself, the appropriate regulatory guides, and applicable codes and standards.

6.6 References

U.S. Code of Federal Regulations, Licensing Requirements for the Independent Storage of Spent Nuclear Fuel and High-Level Radioactive Waste, Title 10, Part 72.

U. S. Nuclear Regulatory Commission, NUREG-1536, “Standard Review Plan for Dry Cask Storage Systems,” January 1997.

7.0 CONFINEMENT EVALUATION

The applicant did not request any changes to the confinement analysis in Chapter K.7 of the FSAR, resulting from addition of new fuel types and damaged fuel to the list of approved contents for the NUHOMS®-61BT DSC. The NUHOMS®-61BT DSC is tested leak-tight in accordance with ANSI N14.5-1997. The confinement analysis has been approved in Amendment No. 3 to the CoC and bounds the additional fuel contents. The staff has reasonable assurance that the confinement system and additional contents continue to satisfy the dose requirements of 10 CFR Part 72. This finding is based on statements and information provided by the applicant, previous staff approval of Amendment No. 3 to the CoC, the regulation itself, the appropriate regulatory guides and applicable codes and standards.

8.0 OPERATING PROCEDURES EVALUATION

The applicant did not request any changes to the operating procedures in Chapter K.8 of the FSAR, resulting from addition of new fuel types to the list of approved contents. The operating procedures have been approved in Amendment No. 3 to the CoC and address the new additional fuel contents, including damaged fuel. The staff has reasonable assurance that the operating procedures with additional contents continue to be in compliance with 10 CFR Part 72.

9.0 ACCEPTANCE TEST AND MAINTENANCE EVALUATION

The applicant did not request any changes to the acceptance test and maintenance program in Chapter K.9 of the FSAR, resulting from addition of new fuel types to the list of approved contents. The acceptance test and maintenance program has been approved in Amendment No. 3 to the CoC. The staff has reasonable assurance that the acceptance test and maintenance program, considering the additional contents, will continue to be in compliance with 10 CFR Part 72.

10.0 RADIATION PROTECTION EVALUATION

The applicant did not request any changes to the radiation protection in Chapter K.10 of the FSAR, resulting from addition of new fuel types to the list of approved contents. The radiation protection program has been approved in Amendment No. 3 to the CoC and bounds the additional fuel contents and resulting direct radiation dose rates. The staff has reasonable assurance that the radiation shielding and confinement features with the additional contents are sufficient to meet the radiation protection requirements of 10 CFR Part 20 and 10 CFR Part 72 for normal conditions and design-basis accidents. This finding is based on statements and information provided by the applicant, previous staff approval of Amendment No. 3 to the CoC, the regulation itself, the appropriate regulatory guides, and applicable codes and standards.

11.0 ACCIDENT EVALUATION

The applicant did not request any changes to the accident analyses in Chapter K.11 of the FSAR, resulting from addition of new fuel types to the list of approved contents. The accident analyses has been approved in Amendment No. 3 to the CoC and continues to bound the additional fuel contents. The staff has reasonable assurance that the NUHOMS®-61BT cask system with the additional fuel contents, has shielding and confinement features that are sufficient to meet the radiation protection requirements of 10 CFR Part 20 and 10 CFR Part 72 for normal and design-basis accident conditions. This finding is based on statements and information provided by the applicant, previous staff approval, the regulation itself, the appropriate regulatory guides, and applicable codes and standards.

12.0 CONDITIONS FOR CASK USE - TECHNICAL SPECIFICATIONS

The purpose of the review of the technical specifications for the cask is to determine whether the applicant has assigned specific controls to ensure that the design basis of the cask system is maintained during loading, storage, and unloading operations.

12.1 Conditions for Use

The conditions for use of the NUHOMS®-61BT DSC, in conjunction with the Standardized NUHOMS® Storage System, are clearly defined in the CoC and TS.

12.2 Technical Specifications

Based on the addition of new BWR fuel types and damaged fuel to the NUHOMS®-61BT DSC to the Standardized NUHOMS® Storage System, the TS have been revised to accommodate the new fuel types and damaged fuel types to be stored in the DSC. These changes have been identified in the TS attachment to the CoC. In addition, three minor changes to the Technical Specifications were requested to correct inconsistencies and remove irrelevant references. Specifically, Table 1-1b, for the row entitled "Burn-Up," the words "and per Figure 1.1" have been removed as this figure is only applicable to PWR. Secondly, the Limit/Specification of TS 1.2.4a for 61BT, 32PT, and the 24PHB DSC has been revised to say "$\leq 1.0 \times 10^{-7}$ reference cubic centimeters per second (cc/s)." This corrects incorrect characterization of ANSI N14.5-1997. Finally, reference to section 8.3 of the FSAR has been removed from TS 1.2.5 as this TS is applicable to all DSCs whereas FSAR section 8.3 is applicable to only the 24P and 52B DSC. The staff reviewed the proposed changes and finds them all acceptable.

Table 12-1 lists the TS for use of the NUHOMS®-61BT DSC system, in concert with the Standardized NUHOMS® Storage System.

12.3 Evaluation of Findings

F12.1 Table 12-1 of this SER lists the TS for the Standardized NUHOMS® Storage System. These TS are included as Appendix A to the CoC.

F12.2 The staff concludes that the conditions for use of the NUHOMS®-61BT DSC, in addition to three other minor clarifications, in conjunction with the Standardized NUHOMS® Storage system, identify necessary TS to satisfy 10 CFR Part 72 and that the applicant acceptance criteria have been satisfied. The TS provide reasonable assurance that the cask will provide for 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 practices.

Table 12-1

Standardized NUHOMS® Horizontal Modular Storage System Technical Specifications for use with the NUHOMS®-61BT DSC

1.1 General Requirements and Conditions
- 1.1.1 Regulatory Requirements for a General License
- 1.1.2 Operating Procedures
- 1.1.3 Quality Assurance
- 1.1.4 Heavy Loads Requirements
- 1.1.5 Training Module
- 1.1.6 Pre-Operational Testing and Training Exercise
- 1.1.7 Special Requirements for First System in Place
- 1.1.8 Surveillance Requirements Applicability
- 1.1.9 Supplemental Shielding

1.2 Technical Specifications, Functional and Operating Limits
- 1.2.1 Fuel Specifications
- 1.2.2 DSC Vacuum Pressure During Drying
- 1.2.3 24P and 52B DSC Helium Backfill Pressure
- 1.2.3a 61BT, 32PT, and 24PHB DSC Helium Backfill Pressure
- 1.2.4 24P and 52B DSC Helium Leak Rate of Inner Seal Weld
- 1.2.4a 61BT, 32PT, and 24PHB DSC Helium Leak Rate of Inner Seal Weld
- 1.2.5 DSC Dye Penetrant Test of Closure Welds
- 1.2.6 Deleted
- 1.2.7 HSM Dose Rates with a Loaded 24P, 52B or 61BT DSC
- 1.2.7a HSM Dose Rates with a Loaded 32PT DSC Only
- 1.2.7b HSM Dose Rates with a Loaded 24PHB DSC Only
- 1.2.8 HSM Maximum Air Exit Temperature
- 1.2.9 Transfer Cask Alignment with HSM
- 1.2.10 DSC Handling Height Outside the Spent Fuel Pool Building
- 1.2.11 Transfer Cask Dose Rates with a Loaded 24P, 52B, 61BT, or 32PT DSC
- 1.2.11a Transfer Cask Dose Rates with a Loaded 24PHB DSC
- 1.2.12 Maximum DSC Removable Surface Contamination
- 1.2.13 TC/DSC Lifting Heights as a Function of Low Temperature and Location
- 1.2.14 TC/DSC Transfer Operations at High Ambient Temperatures
- 1.2.15 Boron Concentration in the DSC Cavity Water for the 24P Design Only
- 1.2.15a Boron Concentration in the DSC Cavity Water for the 32PT Design Only
- 1.2.15b Boron Concentration in the DSC Cavity Water for the 24PHB Design Only
- 1.2.16 Provision of TC Seismic Restraint Inside the Spent Fuel Pool Building as a Function of Horizontal Acceleration and Loaded Cask Weight
- 1.2.17 61BT DSC Vacuum Drying Duration Limit
- 1.2.17a 32PT DSC Vacuum Drying Duration Limit
- 1.2.17b 24PHB DSC Vacuum Drying Duration Limit

1.3 Surveillance and Monitoring
- 1.3.1 Visual Inspection of HSM Air Inlets and Outlets (Front Wall and Roof Birdscreen)
- 1.3.2 HSM Thermal Performance

13.0 QUALITY ASSURANCE

The purpose of this review and evaluation is to determine whether TN has a quality assurance program that complies with the requirements of 10 CFR Part 20, Subpart G. The staff has previously reviewed and accepted the TN quality assurance program in the Standardized NUHOMS® Storage System FSAR.

14.0 DECOMMISSIONING

The decommissioning evaluation was previously reviewed and approved in the Standardized NUHOMS® Storage System FSAR. There were no changes proposed by the applicant with the addition of new fuel types and damaged fuel types to the NUHOMS®-61BT DSC to the list of authorized contents.

CONCLUSION

The NRC staff has performed a comprehensive review of the CoC amendment request and found that the addition of new fuel types and damaged fuel types to the NUHOMS®-61BT DSC to the list of authorized contents does not reduce the safety margin for the Standardized NUHOMS® System. The areas of review addressed in NUREG 1536, "Standard Review Plan for Dry Cask Storage Systems," January 1997, are consistent with the applicant's proposed changes. The Certificate of Compliance has been revised to include the TN requested changes. Based on the statements and representations contained in TN's application, as supplemented, the staff concludes that the addition of new fuel types and damaged fuel types to the Standardized NUHOMS® System NUHOMS®-61BT DSC list of authorized contents meets the requirements of 10 CFR Part 72.

Issued with Certificate of Compliance No. 1004, Amendment No. 7 on **DRAFT**.