TRANSNUCLEAR
AN AREVA COMPANY

July 22, 2011
E-31217

U. S. Nuclear Regulatory Commission
Attn: Document Control Desk
One White Flint North
11555 Rockville Pike
Rockville, MD 20852

Subject: Revision 1 to Transnuclear, Inc. (TN) Application for Amendment 13 to Standardized NUHOMS® System, Response to Request for Supplemental Information (Docket No. 72-1004; TAC No. L24519)

Reference: Letter from Jennifer Davis (NRC) to Kamran Tavassoli (TN), "Application for Amendment No. 13 to NUHOMS® Certificate of Compliance 72-1004 — Request for Supplemental Information (TAC No. L24519)," July 15, 2011

The letter referenced above advised TN that NRC staff had completed an acceptance review of our February 9, 2011 application for Amendment 13 to the Standardized NUHOMS® System Certificate of Compliance No. 1004 and that supplemental information is needed for the staff to continue their review. The information needed was enclosed in the letter as Requests for Supplemental Information (RSIs). The letter also included observations to allow TN to start earlier on items containing the potential to be asked at a later date. The letter indicated that responses to the observations are not required for the staff to begin a detailed technical review.

The purpose of this submittal is to respond to the RSIs and the observations.

- Enclosure 1 provides the proprietary version of the TN responses to the Request for Supplemental Information.
- Enclosure 2 provides the non-proprietary version of the TN responses to the Request for Supplemental Information.
- Enclosure 3 provides a list of changes in Certificate of Compliance (CoC), Technical Specifications (TS), and Safety Analysis Report (SAR) associated with the RSI responses.
- Enclosure 4 provides instructions for the CoC, TS and SAR page replacement.
- Enclosure 5 provides one proposed change to the CoC in response to observation 1-1. This change is indicated by a magenta colored cloud.
- Enclosure 6 includes the proposed changes to the TS. Changed areas are indicated by revision bars in the right margin and italics for inserted text. The new changes are shaded, to distinguish them from the changes proposed in revision 0 of this application.
- Enclosures 7 and 8 include the changed SAR pages for the proprietary and non-proprietary SAR versions, respectively. For Appendices Y and Z, changed areas are indicated by revision bars in the right margin and italics for inserted text. For appendices other than Y and Z, changed areas are indicated by revision bars in the right margin and

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gray shaded with italics for inserted text, to distinguish them from the changes proposed in revision 0 of this application. The changes that were proposed in Amendment 11 and are under NRC review are marked with a text box at the right side.

- Enclosure 9 provides a listing of the computer files contained on Enclosure 10.
- The computer files associated with the RSI responses are provided on DVDs as Enclosure 10. Since the DVDs in Enclosure 10 contain entirely proprietary information, no public version is provided.
- Enclosure 11 is the signed Affidavit Pursuant to 10 CFR 2.390.

Transnuclear looks forward to working with the NRC staff on this amendment application. TN is prepared to meet with the staff to resolve any questions you might have. Should the NRC staff require additional information to support review of this application, please do not hesitate to contact Mr. Kamran Tavassoli at 410-910-6944 or me at 410-910-6881.

Sincerely,

Jayant Bondre

Jayant Bondre, PhD
Vice President - Engineering

cc: Jennifer Davis (NRC SFST), as follows provided in a separate mailing:

- Four paper copies of this cover letter and Enclosures 1, 3 through 7, and 9 through 11
- One DVD containing this cover letter and all Enclosures
- One copy of Enclosure 10 (3 DVDs)

Enclosures:

1. Responses to Request for Supplemental Information, Proprietary Version
2. Responses to Request for Supplemental Information, Non-proprietary Version
3. List of changed pages of the CoC, TS, and SAR associated with RSI responses
4. CoC 1004 Amendment 13, Revision 1, Technical Specifications and SAR Page Replacement Instructions
5. CoC Marked up for Proposed Amendment 13, Revision 1 Changes
6. Changed Pages for the CoC 1004 Amendment 13, Revision 1, Technical Specifications,
7. Changed Pages for the CoC 1004 Application Safety Analysis Report, Revision 1, Proprietary Version
8. Changed Pages for the CoC 1004 Application Safety Analysis Report, Revision 1, Non-proprietary Version
9. Listing of Computer Files Contained in Enclosure 10
10. Computer Files Associated with CoC 1004 Amendment 13, Revision 1 and RSI responses on DVDs (Proprietary)
11. Affidavit Pursuant to 10 CFR 2.390

Enclosure 2 to TN E-31217

Responses to Request for Supplemental Information

(Non-proprietary Version)

RSI-1

RSI-1 Justify the performance characteristics of fuel with an average burnup of 70 GWd/MTU. The new 69BTH Dry Shielded Canister (DSC) is intended to store 69 Boiling Water Reactor (BWR) fuel assemblies with uranium dioxide (UO_2), or 69 intact fuel assemblies with U-Pu mixed-oxide (MOX) fuel pellets with a maximum assembly average burn up of 70 GWd/MTU, and a minimum cooling time of 3.0 years. As stated in the "Standard Review Plan for Spent Fuel Dry Storage Systems at a General License Facility," NUREG-1536 Revision 1, staff acceptance criteria for high-burnup fuel should be limited to burn up values up to that approved by the Office of Nuclear Reactor Regulation (NRR) for reactor operation; the current value is approximately 62.5 GWd/MTU for peak assembly rods. An appropriate justification should be submitted, including but not limited to the following:

1. Cladding oxide thickness,
2. Hydrogen content and hydride morphology,
3. Fission gas release, and
4. Technical bases and independent validation, including trend analysis for using SAS2H code for source term calculation for UO_2 and MOX fuel.
   Note that the SAS2H code of the SCALE 4.4 package used in the source term calculations has not been validated to 70 GWd/MTU. The statements in staff guidance in NUREG/CR-6701, "Review of Technical Issues Related to Predicting Isotopic Compositions and Source Terms for High-Burnup LWR Fuel," which the applicant quoted in the SAR, do not appear to be appropriate for use as a basis for code validation. Review of NUREG/CR-6701 indicates that the statements in Appendix A of NUREG/CR-6701 were for sensitivity studies rather than for code validation. NUREG/CR-6701 also shows that the fuel sample with maximum burn up of 73 GWd/MTU was taken from a reconstituted fuel assembly, and the value is for peak rod burn up. The corresponding fuel assembly burn up is about 60 GWd/MTU (peak burn up/pellet power peaking factor/rod peaking factor = 73/1.1/1.1 ≈ 60).

   In addition, the staff does not find NUREG/CR-7012 and NUREG/CR-7013 provide an adequate basis for SAS2H for use in shielding applications for high burn up fuel since these documents were written for use in burn up credit applications. The nuclides important for burn up credit analyses are not the same as the ones used for shielding analyses.

   This information is required to demonstrate compliance with 10 CFR 72.236(a).

Response to RSI-1

SAR, Appendix Y.3, Section Y.3.5.1 has been updated to address the performance characteristics of fuel with burnup up to 70 GWd/MTU in response to items 1 through 3.

For item 4, as discussed in response to RSI-2, the maximum allowable assembly average burnup for MOX fuel is limited to 62 GWd/MTHM, therefore, the performance characteristics of MOX fuel with burnup up to 70 GWd/MTHM are not addressed herein.

As discussed in the SAR, Appendix Y.5, Section Y.5.2, the SAS2H/ORIGEN-S modules from the SCALE 4.4 computer code package are employed to determine the thermal and radiological source terms for the BWR fuel assemblies. The validation of the computer code was not extended to 70 GWd/MTU due to lack of sufficient experimental data. Note that the reference to NUREG/CR-6701 (reference [5.7] in the SAR) was also made to highlight the lack of experimental data. The SAS2H models employed in the source term calculations and the source specification utilized in the shielding analysis calculations include additional conservatisms to ensure that the resulting dose rates are conservative.

For the purpose of calculating source terms, it is important that the code is able to track a large number of isotopes with sufficient cross-section data. For the ENDF-B/V cross section library employed herein, the uncertainties associated with nuclide concentrations (based on isotopic assay evaluations) are approximately 10% as described in the SAR, Appendix Y.5, Section Y.5.2, and are within the uncertainty

of the measurement. This estimate is valid for high burnup fuel up to 62 GWd/MTU. For higher burnups, the following conservatisms are included:

- Heat Load Zoning Configuration
    - The decay heat zone loading employed in the shielding analysis is discussed in Appendix Y.5, Page Y.5-2. The decay heat zones are modeled as shown below:
        - Zone 1, Zone 2 and Zone 3, 25 fuel assemblies, 0.49 kW/FA
        - Zone 4 and Zone 5, 44 fuel assemblies, 0.70 kW/FA
        - Total Decay Heat per DSC = 43.05 kW
    - The allowable decay heat loading configurations are included in the Technical Specifications (and in Appendix Y.2, Figure Y.2-1 through Y.2-6). These decay heat zones with the maximum allowable decay heat per assembly are as shown below:
        - Zone 1, 1 fuel assembly, 0.25 kW/FA
        - Zone 2, 8 fuel assemblies, 0.45 kW/FA
        - Zone 3, 16 fuel assemblies, 0.40 kW/FA
        - Zone 4, 20 fuel assemblies, 0.70 kW/FA
        - Zone 5, 24 fuel assemblies, 0.60 kW/FA
        - Total Decay Heat per DSC = 35 kW
    - Therefore, the source terms and consequently, the dose rates are calculated with a conservatism of approximately 20% in the total decay heat load.
- Specific Power
    - The specific power employed in the SAS2H calculations for high burnup fuel is 14 MW per fuel assembly which is significantly higher than that (5 MW per fuel assembly) for typical BWR fuel assemblies and results in a conservative prediction of source terms.
- Occupational Exposure
    - As described in the SAR, Appendix Y.10, Section Y.10.1, the occupational exposures for the 69BTH DSC are calculated assuming the more conservative dose rates from the 61BTH DSC documented in the UFSAR, Appendix T.10. The dose rates for the 61BTH DSC calculations are approximately 20% higher than those calculated for the 69BTH DSC.
- ISFSI Site Dose Evaluation
    - As described in the SAR, Appendix Y.10, Section Y.10.2, the ISFSI annual exposures at the site boundary for the 69BTH DSC are calculated assuming the more conservative dose rates from the 61BTH DSC documented in the UFSAR, Appendix T.10. The dose rates for the 61BTH DSC calculations are approximately 20% higher than those calculated for the 69BTH DSC.

- Accident Dose Rate Evaluation

  - As described in the SAR, Appendix Y.11, the calculated accident dose rates for the 69BTH DSC are well within the applicable Part 72 limits (by at least a factor of 200). Therefore the effect of uncertainties for accident exposure calculations is insignificant.

- MOX Fuel Assemblies

  - The maximum allowable burnup for MOX fuel assemblies is limited to 62 GWd/MTU. Therefore, a separate uncertainty evaluation for higher burnups for MOX assemblies is not required. Further, the conservatisms for UO_2 fuel assemblies described above for the 69BTH DSC are also applicable for MOX fuel assemblies.

In summary, the dose rates for the 69BTH DSC are evaluated with an additional conservatism of approximately 20% that covers for all applicable uncertainties associated with source term methodology. In addition, the occupational exposure and site dose calculations are performed using an additional conservatism of approximately 20%. Therefore, the use of the SAS2H/ORIGEN-S methodology for the purpose of calculating source terms is acceptable for higher burnup fuel without additional evaluations. SAR, Appendix Y, Section Y.5.2 is revised to include the above conclusion.

RSI-2

RSI-2 Justify the statement made in Section Y.4.9.4., "Effective Properties for MOX Fuel Assemblies, "[t]he characteristics of the MOX fuel assemblies are almost identical to the fuel assemblies they are replacing." Provide documentation supporting the statement, with a complete description of irradiated MOX fuel and the associated cladding properties, including, but not limited to:

1. Discussion of cladding oxide thickness,
2. Discussion of cladding hydrogen content,
3. Discussion of potential for fission gas release,
4. A plot of the long-term pressure inside fuel rods containing MOX, accounting for alpha decay of plutonium,
5. Discussion of the expected creep behavior of fuel rods containing MOX while in dry cask storage,
6. Engineering drawings providing the nominal dimensions and hardware present in the MOX assemblies,
7. Limits on the MOX fuel assembly growth as a result of reactor operation, and
8. Technical bases and independent validation, including trend analysis for using SAS2H code for source term calculation.

This information is required to demonstrate compliance with 10 CFR 72.236(a).

Response to RSI-2

The qualification of MOX fuel assemblies for the 69BTH DSC is discussed in the SAR, Appendix Y.2. The following are the restrictions imposed on the qualification of MOX fuel assemblies for storage.

- Maximum number of MOX fuel assemblies = 44

- Decay heat of MOX fuel assemblies ≤ 0.6 kW

- Maximum burnup of MOX fuel assemblies = 62 GWd/MTHM

- Damaged MOX fuel assemblies are not qualified

The qualification of MOX fuel assemblies for the 37PTH DSC is discussed in Appendix Z.2. The following are the restrictions imposed on the qualification of MOX fuel assemblies for storage.

- Maximum number of MOX fuel assemblies = 4
- Decay heat of MOX fuel assemblies ≤ 1.2 kW
- Maximum burnup of MOX fuel assemblies = 62 GWd/MTHM
- Damaged MOX fuel assemblies are not qualified
- Control Components are not authorized to be loaded in MOX fuel assemblies

Structural Properties and Mechanical Characteristics

The cladding materials, structural components, etc., used in MOX fuel designs are the same as those used in commercial UO_2 fuel designs. Although there are differences in the fuel pellets themselves, overall, the fuel must meet the same in-reactor design licensing criteria [1] regardless of these differences. Thus, although MOX fuel types might have higher fission gas release fractions, for example, operational limits (e.g., lower linear heat generation rate limits) or design changes would be imposed on these fuels to allow them to still meet the relevant design licensing criteria. Given that the fuel types must meet the same design licensing criteria and the cladding materials, structural components, etc., of the fuel assemblies are the same, only minor difference in end-of-life fuel performance is expected between these fuel types. In short, the end-of-life fuel performance measures (e.g., cladding oxide thicknesses, cladding hydrogen contents, fission gas pressures, creep behavior) are bounded by the design licensing criteria regardless of pellet fuel type.

International programs to evaluate performance of MOX fuel relative to that of UO_2 fuels have been carried out in the past 30 years. These programs are ongoing and are providing the data necessary to compare the performance of MOX and UO_2 fuels. A brief summary of the programs is presented in [2]. Furthermore, weapons-grade MOX fuel rods with rod-average burnups of 39.7 to 47.3 GWd/MTHM have been examined in a hot cell and were compared with those from previous work on both UO_2 and reactor-grade MOX fuel [3]. The results indicate that the weapons-grade MOX fuel had performed normally, similar to that of reactor-grade MOX fuel.

Long-Term Fuel Rod Internal Pressure

Proprietary information withheld pursuant to 10 CFR 2.390

Technical Bases for using SAS2H Code

The source term and shielding evaluation of MOX fuel assemblies for 69BTH DSC is discussed in the SAR, Appendix Y.5, Section Y.5.5. Since the qualification of MOX fuel assemblies is based on decay heat loading, the source terms are bounded by those for the UO_2 fuel assemblies. It is shown that the dose rates from the MOX fuel assemblies will be lower than those of the design basis UO_2 fuel assemblies by a factor of approximately 3.7. ORNL/TM-2003/2 (reference [5.14] of SAR, Section Y.5) and ORNL/TM-1999/326 (reference [5.16] of SAR, Section Y.5) discuss the applicability of SAS2H for the calculation source terms.

The source term and shielding evaluation of MOX fuel assemblies for 37PTH DSC is discussed in the SAR, Appendix Z.5, Section Z.5.5. Since the qualification of MOX fuel assemblies is based on decay heat loading, the source terms are bounded by those for the UO_2 fuel assemblies. It is shown that the dose rates from the MOX fuel assemblies will be lower than those of the design basis UO_2 fuel assemblies by a factor of approximately 3.7. ORNL/TM-2003/2 (reference [5.17] of SAR, Section Z.5) and ORNL/TM-1999/326 (reference [5.19] of SAR, Section Z.5) discuss the applicability of SAS2H for the calculation source terms.

For the purpose of calculating source terms, it is important that the code is able to track a large number of isotopes with sufficient cross-section data. For the ENDF-B/V cross section library employed herein, the uncertainties associated with nuclide concentrations (based on isotopic assay evaluations) are approximately 10% for UO_2 fuel assemblies. It is expected that the uncertainties associated with the MOX fuel assemblies are also within the same envelope although the MOX fuel uncertainties may be higher than the UO_2 fuel assemblies. Since it is shown that the dose rates with the MOX fuel assemblies are approximately 3.7 times lower than those for UO_2 fuel assemblies, the effect of uncertainties is not significant.

In summary, the dose rates for the 69BTH and 37PTH DSCs with MOX fuel assemblies are shown to be significantly lower than those with the UO_2 fuel assemblies. This conservatism sufficiently covers for all applicable uncertainties associated with source term methodology. In addition, the qualification of MOX fuel assemblies also imposes some restrictions that result in further reduction in dose rates.

References:

1. USNRC, NUREG-0800, Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition, Chapter 4, Section 4.2, "Fuel System Design", Revision 3.

2. Framatome ANP, "Fuel Qualification Plan," April 2001, NRC ADAMS: ML013390597

3. Kevin McCoy et all, "Hot Cell Examination of Weapons-Grade MOX Fuel," Proceedings of 2010 LWR Fuel Performance/TopFuel/WRFPM, September 26-29, 2010, pp. 82-88 (provided in Enclosure 8).

RSI-3

RSI- 3 Justify the statement made in Z.2.1.1, "Intact or Damaged UO2 Fuel," that fuel rods housing Blended Low-Enrichment Uranium (BLEU) fuel are similar to traditional UO2 containing fuel rods." TN should provide documentation supporting the statement, including a complete description of irradiated BLEU fuel and the associated cladding properties, and including, but not limited to:

1. Discussion of cladding oxide thickness,
2. Discussion of cladding hydrogen content,
3. Discussion of the potential for fission gas release,
4. Discussion of the expected creep behavior of fuel rods containing BLEU while in dry cask storage,
5. Engineering drawings providing the nominal dimensions and hardware present in the BLEU assemblies,
6. Limits on the BLEU fuel assembly growth as a result of reactor operation, and
7. Technical bases and independent validation, including trend analysis for using SAS2H code for source term calculation.

This information is required to demonstrate compliance with 10 CFR 72.236(a)

Response to RSI-3

Proprietary information withheld pursuant to 10 CFR 2.390

References:

1. Tennessee Valley Authority. "Additional Use of Blended Low Enriched Uranium (BLEU) in Reactors at TVA's Browns Ferry and Sequoyah Nuclear Plants", May 2011.
www.tva.gov/environment/reports/heu/ea.pdf

2. USNRC, NUREG-0800, Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition, Chapter 4, Section 4.2, "Fuel System Design", Revision 3.
3. USNRC ADAMS: ML090780181, Subject: Summary of January 28, 2009, Meeting with the Tennessee Valley Authority Regarding Proposed Fuel Transition Amendment (TAC No. ME0438).
4. USNRC ADAMS: ML090330363, "Loading BLEU Fuel in Browns Ferry Unit 1."

RSI-4

RSI-4 Revise the definition "Failed Fuel" in the Safety Analysis Report (SAR) to state that loose debris, without exception, must be placed in a failed fuel can. The definition "Damaged Fuel" in the SAR should prohibit the storage of assemblies which, although they can be handled by normal means, may have significantly larger defects in the cladding than pinhole and hairline cracks.

Interim Staff Guidance Document 1, Rev 2, (ISG-2) "Classifying the Condition of Spent Nuclear Fuel for Interim Storage and Transportation Based on Function," provides staff with technical guidance for appropriate definitions of damaged fuel.

This information is required to demonstrate compliance with 10 CFR 72.236(a).

Response to RSI-4

TN has revised the definition of failed fuel contained in the 24PTH and 61BTH DSCs to state that loose debris must be placed in a failed fuel can. Additional clarifications have been made to remove references to "damaged fuel" contained in the TN definition of "failed fuel". See revised TS Table 1-1l and SAR Table P.2-1 for the 24PTH and TS Table 1-1t and SAR Table T.2-1 for the 61BTH.

As discussed with the staff, the TN definition for damaged fuel contained in the SAR is the result of previous discussions with the staff during Amendment 10 licensing. Based on those discussions the following clarification was added in the Amendment 10 fuel class specifications for the 61BTH and 32PTH1 DSCs: "Damaged fuel assemblies beyond the definition contained below are not authorized for storage". Similar language has been added to the fuel class specifications for the 24PTH (TS Table 1-1l and SAR Table P.2-1); for the 24PHB (TS Table 1-1i and SAR Table N.2-1); and for the 61BT (TS Table 1-1j and SAR Table K.2-2), which were not in Amendment 10 scope.

In addition, as stated in the fuel specification tables for those DSCs that allow storage of damaged fuel, top and bottom end caps are required to be installed in those storage cells storing damaged fuel.

Based on the above no revision of the definition for damaged fuel is necessary.

RSI-5

RSI-5 Provide licensing drawings of the failed fuel cans.

There is no description of the failed fuel cans in the amendment application.

This information is required to demonstrate compliance with 10 CFR 72.236(a), and 72.236(d).and

Response to RSI-5

Failed fuel is allowed for storage only in the 24PTH and 61BTH DSCs. The following licensing drawings of the failed fuel cans are provided in the amendment application: NUH24PTH-72-1008 Revision 0A (see section P.1.5) for the 24PTH canister and drawing NUH61BTH-72-1105 Revision 0A (see section T.1.5) for the 61BTH canister.

The failed fuel cans in the 24PTH canister are described or noted in Appendix P, Sections P.1 (page P.1-1), P.1.1 (page P.1-6), P.1.2.2.1 (page P.1-7), P.1.2.3 (page P.1-8), and Table P.2-17 (page P.2-41).

The definition of the contents in a failed fuel can in the 24PTH canister is discussed in Appendix P, Section P.2.1 (page P.2-2a), and Table P.2-1 (page P.2-19).

The evaluations of the failed fuel cans in the 24PTH canister are discussed in the following sections.

- Structural evaluation: Appendix P, Section P.3.6.4
- Thermal evaluation: Appendix P, Section P.4.6.9
- Shielding evaluation: Appendix P, Section P.5
- Criticality evaluation: Appendix P, Section P.6.1 and Section P.6.4

Operational steps to handle failed fuel cans in the 24PTH canister are described in Appendix P, Section P.8 (page P.8-2a, page P.8-4, and page P.8-19).

The failed fuel cans in the 61BTH canister are described or noted in Appendix T, Sections T.1 (page T.1-1), T.1.1 (page T.1-3), T.1.2.2.1 (page T.1-5), and Table T.2-15 (page T.2-34).

The definition of the contents in a failed fuel can in the 61BTH canister is discussed in Appendix T, Section T.2.1 (page T.2-2), and Table T.2-1 (page T.2-15).

The evaluations of the failed fuel cans in the 61BTH canister are discussed in the following sections.

- Structural evaluation: Appendix T, Section T.3.6.3.4
- Thermal evaluation: Appendix T, Section T.4.6.9
- Shielding evaluation: Appendix T, Section T.5
- Criticality evaluation: Appendix T, Section T.6.2

Operational steps to handle failed fuel cans in the 61BTH canister are described in Appendix T, Section T.8 (page T.8-2a, page T.8-4, and page T.8-20).

All the above information was provided in the amendment application submittal on February 9, 2011.

RSI-6

RSI-6 Provide the density and specific heat of the MOX fuel pellets and Boral plates for the thermal evaluation. The applicant provided thermal conductivities of the MOX fuel pellets and the Boral plate for thermal evaluations in Chapters Y.4.2 (69BTH DSC) and Z.4.2 (37PTH DSC) without data for density and specific heat. The applicant should also provide the density and specific heat of both MOX fuel pellets, and Boral plate, to calculate the temperatures of both MOX and Boral plate and evaluate their thermal performance in 69BTH and 37PTH DSCs.

This information is required to demonstrate compliance with 10 CFR 72.236(a).

Response to RSI-6

The thermal analysis of the NUHOMS® -69BTH and NUHOMS® -37PTH DSCs for transient conditions (i.e., operations with time limits either in transfer cask or blocked vent accident in storage module) reported in the SAR, Chapter Y.4 and Chapter Z.4, respectively, are based on steady-state thermal evaluations of the DSC/basket models, in which the DSC shell temperatures were determined and retrieved from the transient analyses of the transfer cask or HSM-H models.

The thermal models for the transient analyses explicitly model either the transfer cask or the storage module components and the DSC shell. In these models, the baskets within the DSCs are modeled as a homogenized region with an effective thermal conductivity, effective density and effective specific heat. These transient analyses provide the DSC shell temperature profiles for steady-state thermal evaluations of the DSC/basket models to evaluate the maximum fuel cladding and DSC component temperatures.

Therefore, the density and specific heat of the various components are not explicitly used in determining the maximum temperatures. They are only used in determining the effective basket density and effective basket specific heat, which are an input for the transient analyses that include homogenized basket.

MOX Fuel Pellet Density and Specific Heat

In the current application, the effective density (ρ) and specific heat (C_p) for the MOX fuel assemblies are considered to be the same as that of UO_2 fuel assemblies as stated in Appendix Y, Chapter Y.4, Section Y.4.9.4 and Appendix Z, Chapter Z.4, Section Z.4.9.1.

Based on the information presented in Table 1, which is based on [1], using the UO_2 fuel assembly effective density and specific heat is conservative for a MOX fuel assembly. Table 1 presents the comparison of the heat capacity (ρxC_p) between UO_2 fuel assemblies and MOX fuel assemblies and shows that the heat capacity of the UO_2 fuel assemblies is lower than the MOX fuel assemblies.

Therefore, the effective density and specific heat determined for fuel assemblies in Appendix Y, Chapter Y.4, Section Y.4.9 for 69BTH DSC and Appendix Z, Chapter Z.4, Section Z.4.9 for 37PTH DSC are conservative in evaluating the thermal performance.

Boral Density and Specific Heat
In calculating the effective density and effective specific heat of the baskets as noted in Appendix Y, Chapter Y.4, Section Y.4.6.6.4 for 69BTH DSC and in Appendix Z, Chapter Z.4, Section Z.4.4.2 for 37PTH DSC, the density and specific heat of Boral are assumed to be equivalent to that of Aluminum 6061.

Boral is a clad composite of Boron Carbide and Al1100, with two distinct outer layers of Al1100 and a central core with a uniform aggregate of fine boron carbide particles held within an aluminum alloy matrix Based on the information in Section 18.2.4 and 18.2.5 of [2], the heat capacity (ρxC_p) at 500°F of the aluminum 1100 clad is 0.026 Btu/in^3-°F (0.268 Btu/lb$_m$-°F x 0.098 lb$_m$/in^3) and that of the central core is 0.029 Btu/in^3-°F (0.323 Btu/lb$_m$-°F x 0.09 lb$_m$/in^3).

Therefore, as shown above heat capacity (ρxC_p) of the aluminum 1100 clad is lower for Boral, which conservatively results in over-prediction of the maximum temperatures for transient analyses. However, the conservative heat capacity (ρxC_p) of aluminum 6061, which is lower than that of Al1100, is used in determining the effective basket properties.

Table 1: Comparison of Heat Capacity ($\rho \times C_p$) for UO_2 and MOX fuel assemblies (See Table 4.3 of [1])

Fuel T (K)	UO_2			MOX (5% PuO_2)		
	$\rho \times 10^{-4}$ (kg/m^3)	$C_p \times 10^{-2}$ (J/kg/K)	$0.95 \times \rho \times C_p \times 10^{-6}$ (J/m^3/K)	$\rho \times 10^{-4}$ (kg/m^3)	$C_p \times 10^{-2}$ (J/kg/K)	$0.95 \times \rho \times C_p \times 10^{-6}$ (J/m^3/K)
300	1.0961	2.3658	2.4635	1.0986	2.3698	2.4732
400	1.0929	2.6432	2.7443	1.0954	2.6521	2.7598
500	1.0897	2.8153	2.9144	1.0921	2.8253	2.9314
600	1.0865	2.9299	3.0241	1.0889	2.9403	3.0416
700	1.0832	3.0071	3.0945	1.0856	3.0181	3.1127
800	1.0800	3.0584	3.1378	1.0824	3.0702	3.1570
900	1.0766	3.0918	3.1624	1.0790	3.1050	3.1829
1000	1.0733	3.1140	3.1752	1.0757	3.1286	3.1972
1100	1.0699	3.1306	3.1819	1.0723	3.1466	3.2054
1200	1.0664	3.1465	3.1877	1.0688	3.1640	3.2125
1300	1.0628	3.1666	3.1971	1.0652	3.1850	3.2229
1400	1.0590	3.1950	3.2144	1.0614	3.2139	3.2407
1500	1.0551	3.2357	3.2434	1.0575	3.2545	3.2695
1600	1.0511	3.2925	3.2876	1.0534	3.3102	3.3127
1700	1.0468	3.3688	3.3502	1.0491	3.3845	3.3733
1800	1.0423	3.4679	3.4340	1.0447	3.4803	3.4540
1900	1.0376	3.5926	3.5415	1.0400	3.6005	3.5571
2000	1.0327	3.7457	3.6748	1.0350	3.7475	3.6848
2100	1.0275	3.9297	3.8359	1.0298	3.9239	3.8387
2200	1.0220	4.1468	4.0261	1.0243	4.1316	4.0204
2300	1.0162	4.3989	4.2468	1.0185	4.3726	4.2308

References:

1. Oak Ridge National Laboratory, "Thermophysical Properties of MOX and UO_2 Fuels including the Effect of Irradiation," by Popov, Carbajo, Ivanov, and Yoder, ORNL/TM-2000/351.
2. Standard Specification for Boral® Composite Panels, AAR Manufacturing.

RSI-7

RSI-7 Provide the bounding thermal analyses for 10% fuel rod rupture to ensure the DSC designs comply with the thermal requirements of Part 72 under off-normal conditions. The maximum DSC cavity internal design pressures for off- normal and accident conditions are different (e.g., 20 psig and 120 psig for the 69BTH DSC, and 20 psig and 140 psig for the 37PTH DSC, respectively, under off-normal and accident conditions). In addition to the analysis of 100% fuel rod rupture under accident conditions) the applicant should provide the bounding thermal analyses for 10%, fuel rod failure to ensure the internal pressure of each of the new DSC designs (69BTH, and 37PTH with their corresponding maximum heat loads) is below its design pressure limit under off-normal conditions.

This information is required to demonstrate compliance with 10 CFR 72.236(f), and 72.236(l},

Response to RSI-7

The 69BTH and 37PTH DSC thermal analyses provide average helium temperatures within DSC cavity, which are used for maximum DSC cavity pressure evaluation as described in the SAR, Section Y.4.7 and Section Z.4.7, respectively.

The fraction of the ruptured fuel rods is assumed to be 10% for off-normal cases (See Section Y.4.7.6 for 69BTH DSC and Section Z.4.7.6 for 37PTH DSC).

For 69BTH DSC cavity average helium temperatures are retrieved from analyses results presented in Section Y.4.6 and listed in Table Y.4-11 (with load cases description shown in Table Y.4-4). Based on Table Y.4-11, the normal transfer condition (load case T6) is the limiting condition with the highest average helium temperature for both normal and off-normal conditions for 69BTH DSC. Therefore the DSC cavity average temperature from normal transfer condition (load case T6 listed in Table Y.4-12) is selected to bound the DSC internal pressure for off-normal conditions for 69BTH DSC.

For 37PTH DSC, the DSC cavity average helium temperatures are retrieved from Section Z.4.6 and listed in Table Z.4-5 (with load cases description shown in Table Z.4-2). Based on Table Z.4-5, the highest average helium temperature for off-normal conditions is achieved under off-normal storage condition (load case S4). The 37PTH DSC cavity average helium temperatures are calculated based on bounding conditions and listed in Table Z.4-6. The bounding average helium temperature (load case S4) is used for off-normal conditions for DSC cavity pressure calculation.

The maximum DSC cavity pressures for off-normal condition are lower than the design pressures as shown in Table Y.4-13 for 69BTH DSC and in Table Z.4-7 for 37PTH DSC.

RSI-8

RSI-8 Provide the bounding thermal analyses for 50% air duct blockage (inet only) to ensure the DSC designs comply with the thermal requirements of Part 72 under off-normal conditions. The maximum DSC cavity internal design pressures for off-normal and accident conditions are different (e.g., 20 psig and 120 psig for the 69BTH DSC, and 20 psig and 140 psig for the 37PTH DSC, respectively, under off-normal and accident conditions). In addition to the analysis of 100% air duct blockage under accident conditions, the applicant should provide the bounding thermal analyses for 50% inlet air duct blockage to ensure the internal pressures of the new DSC designs (69BTH. and 37PTH, with their corresponding maximum heat loads) are below the design pressure limit under off-normal conditions.

This information is required to demonstrate compliance with 10 CFR 72.236(f), and 72.236(1).

Response to RSI-8

For 69BTH DSC

An off-normal thermal analysis is performed for 69BTH DSC in HSM-H with 35 kW heat load considering an ambient temperature of 117°F and 50% blockage of the HSM-H inlet vents. To determine the bounding effect of 50% blockage of the HSM-H inlet vents on the thermal performance of the 69BTH DSC in HSM-H, the blockage is considered to occur over the bottom half of the 30" high inlet vents reducing the area of the inlet by half.

The SAR, Appendix Y, Sections Y.4.4, Y.4.6 and Tables Y.4-1, Y.4-2, Y.4-4, Y.4-9, Y.4-10, and Y.4-11 are revised to include this analysis.

A comparison of the average temperatures (shown in Table Y.4-11) between the storage case of "off-normal hot with 50% inlet vent blockage" (load case S3A) and the bounding transfer case (load case T6), shows that the average cavity gas temperature for off-normal condition with 50% blockage of the HSM-H inlet vents is well bounded by the bounding transfer case (load case T6).

Therefore, the average temperatures from load case T6 associated with the off-normal internal DSC pressure of 16.7 psig, which is below the off-normal design pressure of 20 psig, as reported in Table Y.4-13 remains bounding for the off-normal storage conditions.

For 37PTH DSC

As noted in the SAR, Appendix Z, Section Z.4.1, the thermal evaluation of HSM-H with 32PTH1 DSC with a heat load of 31.2 kW as described in Appendix U, Section U.4.4 is used as the basis for the thermal evaluation of the HSM-H with 37PTH DSC. The justification for this approach is presented in Appendix Z, Section Z.4.4.2 of the UFSAR.

For off-normal storage condition with 50% blockage of the HSM-H inlet vents (load case S4A in Table Z.4-2), the thermal performance of the 37PTH DSC in HSM-H is determined using the same methodology described in Appendix U, Section U.4.4 with the blockage considered to occur over the bottom half of the 30" high inlet vents, reducing the area of the inlet by half.

Due to conservatism considered in the thermal evaluation of HSM-H loaded with 37PTH DSC for off-normal storage condition without vent blockage (load case S4 in Table Z.4-2), such as considering a heat load of 31.2 kW instead of allowable heat load of 30 kW and a 24-hour average ambient temperature of 117°F, the exit air temperature for off-normal storage conditions with 50% blockage of the HSM-H inlet vents (load case S4A) is lower than the one evaluated for off-normal storage condition (load case S4 in Table Z.4-2). Therefore, the thermal analysis performed for off-normal storage condition (load case S4) remains bounding for the off-normal storage condition with 50% blockage of the HSM-H inlet vent (load case S4A).

The off-normal internal DSC pressure of 17.1 psig reported in Table Z.4-7 remains bounding for the off-normal storage condition with 50% blockage of HSM-H inlet vents, which is below the off-normal design pressure of 20 psig.

The SAR, Appendix Z, Sections Z.4.4, Z.4.6, and Tables Z.4-2, Z.4-3, and Z.4-5 are revised to include this analysis.

Note that the same conclusions are also applicable to other DSC designs described in the UFSAR with 50% blockage of inlet vents described above. Technical Specification has been revised to reflect this condition.

RSI-9

RSI-9 Provide the bounding thermal analyses for extreme hot ambient temperatures to ensure the DSC designs comply with the thermal requirements of Part 72 under accident conditions. The maximum DSC cavity internal design pressures for off-normal and accident conditions are different (e.g., 20 psig and 120 psig for the 69BTH DSC, and 20 psig and 140 psig for the 37PTH DSC, respectively, under off~normal and accident conditions). In addition to the analysis for an ambient temperature of 105°F under off-normal conditions, the applicant should provide the bounding thermal analyses for extreme hot ambient temperatures to ensure the internal pressures of the new DSC designs (69BTH, and 37PTH), with their corresponding maximum heat loads) are below the design pressure limit under accident conditions.

This information is required to demonstrate compliance with 10 CFR 72.236(f), and 72.236(1).

Response to RSI-9

For 69BTH DSC

A thermal analysis is performed for 69BTH DSC in HSM-H with 35 kW heat load considering a 24-hour average ambient temperature of 117°F as an accident condition of extreme hot ambient temperature. The SAR, Appendix Y.4, Sections Y.4.4, Y.4.6, and Y.4.7 are revised to include this analysis.

A comparison of the average temperatures (shown in Table Y.4-11) between the storage accident case of "extreme hot ambient condition" (load case S6) and the transfer accident condition of "loss of neutron

shield" (load case T14), shows that the average cavity gas temperatures for accident conditions are well bounded by the transfer accident (load case T14).

The average helium temperatures from load case T14 were selected in the SAR (see Table Y.4-12) to determine the DSC internal pressure for accident conditions. The accident internal DSC pressure of 115.5 psig reported in Table Y.4-13 remains bounding for the storage accident case of extreme hot ambient temperature, which is below the accident design pressure limit of 120 psig.

37PTH DSC

As described in the SAR, Appendix Z.4, Section Z.4.4.2, the thermal analyses of the 37PTH DSC in the HSM-H with 30 kW heat load are based on the DSC shell temperature profiles resulting from the thermal analyses of the 32PTH1 DSC in the HSM-H with 31.2 kW heat load. The analyses of the 32PTH1 DSC in the HSM-H were discussed in Appendix U, Section U.4.4. The off-normal ambient temperature conservatively considered for these analyses is 117°F (24-hour average ambient temperature) as shown in Appendix U, Table U.4-1 and U.4-2. This average ambient temperature represents an accident condition of extreme hot ambient temperature. The SAR, Appendix Z.4, Sections Z.4.4 and Z.4.6, and Table Z.4-5 are revised to include the discussion of the accident condition of the extreme hot ambient temperature.

The average helium temperatures for off-normal condition (load case S4) shown in Table Z.4-5 are calculated based on a 24-hour average ambient temperature of 117°F and are therefore applicable for the extreme hot ambient accident condition (load case S6). These average helium temperatures are well bounded by the average helium temperatures for the transfer accident condition (load case T8).

The DSC internal pressure for accident conditions are calculated based on average temperatures from load case T8 and therefore, the accident internal DSC pressure of 117.3 psig reported in Table Z.4-7 remains bounding for the storage accident condition of extreme hot ambient temperature, which is well below the accident design pressure limit of 140 psig.

RSI-10

RSI-10 Provide a table to (1) list the bounding thermal analyses for cask storage for each type of DSC or for multiple types of DSCs and (2) summarize the bounding correlations among multiple types of DSCs under normal, off-normal and accident conditions.

The NUHOMS® Amendment 13 includes many changes and additions in types of DSCs, decay heat limits, poison materials, basket configuration, MOX fuel, etc. Under the premise that only some thermal analyses were selected and performed for cask storage, either for each type of DSC or for multiple types of DSCs, to bound many proposed combinations of DSC type, decay heat limit, content (intact, damaged, and MOX fuels), and poison material (borated aluminum alloy, MMC, or Boral), loading zone configuration and transfer cask (0S200 or OS200FC), the applicant should provide a table to list and summarize the thermal analyses selected as the bounding cases for each type of DSC and for multiple types of DSCs. Staff needs this table to verify that the selected thermal analyses are appropriate for each type of DSC or multiple types of DSCs under normal, off-normal and accident conditions.

This information is required to demonstrate compliance with 10 CFR 72.236(f) and 72.236(1).

Response to RSI-10

In the Standardized NUHOMS® System (CoC 1004), the HSM-H and HSM-HS storage modules are currently licensed for heat loads of up to 40.8 kW while the standardized HSM (including HSM Model 202) is currently licensed for heat loads up to 24 kW.

In the current amendment application, the use of HSM-HS storage module is expanded to include the previously approved DSCs (61BT, 32PT, 24PTH and 61BTH). The maximum allowable heat loads for previously approved DSCs are not changed in this application.

As described in Appendix U, Section U.1.2.1.2, HSM-HS storage module is an upgraded version of the HSM-H storage module design to meet the higher seismic criteria. The physical features (materials, geometry and dimensions) of the HSM-H and HSM-HS storage modules are identical. Therefore, the HSM-HS storage module is thermally identical to the HSM-H storage module. Similarly the HSM Model 202 is also similar to the HSM-H model as described in Appendix V, Section V.1.5 with the differences shown in Appendix V, Section V.1.5, Dwg # NUH-03-7002-SAR. As seen from the drawing, the differences do not affect the thermal performance of the storage module.

Therefore, the thermal evaluations performed for any HSM-H or HSM Model 202 are applied to the HSM-HS for the same DSC and same heat load.

Thermal evaluations are performed for new DSCs (69BTH and 37PTH) for storage in HSM-H/HSM-HS storage modules. The allowed maximum heat loads for 69BTH DSC and 37PTH DSC in HSM-H/HSM-HS storage modules are limited to 35.0 kW and 30.0 kW, respectively.

Further, in addition to the above changes, optional dose reduction hardware is considered for HSM-H and HSM-HS modules as described in Appendix P, Section P.1.2.1.2. However, as discussed in Appendix P, Section P.4.4.3.1, the effect of dose reduction hardware is shown to be negligible on the thermal performance of the HSM-H or HSM-HS.

The details of the thermal analyses performed to qualify the 69BTH DSC stored in the HSM-H is described in Appendix Y.4, Section Y.4.4 for the HSM-H model and Section Y.4.6 for the 69BTH DSC. These analyses are performed using ANSYS models explicitly developed for the HSM-H and 69BTH DSC.

The details of the thermal evaluations performed to qualify the 37PTH DSC stored in the HSM-H is described in Appendix Z.4, Section Z.4.4 for the HSM-H model and Section Z.4.6 for the 37PTH DSC. The evaluations performed for the HSM-H loaded with the 37PTH DSC are based on analyses performed for the HSM-H loaded with the 32PTH1 DSC. The justification for this approach is described in detail in Appendix Z, Section Z.4.4.2. The thermal evaluation of the 37PTH DSC is performed using ANSYS models explicitly developed for the 37PTH DSC. Detail of the 37PTH DSC model is described in Appendix Z, Section Z.4.6.

The MOX fuel assemblies are considered only in the 69BTH and 37PTH DSCs. The effects of the MOX fuel assemblies on thermal performance are discussed in response to RSI-6.

Failed fuel assemblies are added only to the allowable content of 61BTH and 24PTH DSCs. The thermal evaluation of the failed fuel assemblies in the 24PTH DSC is described in the SAR, Appendix P, Section P.4.6.9. The thermal evaluation of the failed fuel assemblies in the 61BTH DSC is described in the SAR, Appendix T, Section T.4.6.9.

The effects of the damaged fuel assemblies on the thermal performance are discussed in Appendix Y, Sections Y.4.6.2 and Y.4.6.7 for the 69BTH DSC and in Appendix Z, Sections Z.4.6.2 and Z.4.6.8 for the 37PTH DSC. The location and the number of damaged fuel assemblies in the other DSCs are not subject of the Amendment 13 application.

The following table provides a list of bounding thermal evaluation of storage for the various DSCs in the Standardized NUHOMS® system (CoC 1004) that are currently licensed and the new DSCs that are under review.

DSC/Storage Module Type	Bounding Heat Load	Bounding Thermal Analysis For Storage	Justification/Reference
NUHOMS®-61BT DSC in HSM-HS	18.3 kW	The thermal performances of NUHOMS®-61BT DSC in HSM Model 202 is evaluated in Appendix V, Section V.4.4 (Table V.4-1).	The design and structures of air channels and heat transfer features of HSM-HS are identical to the HSM Model 202 described in Appendix V. Thermal evaluation performed for the HSM Model 202 are applicable for the HSM-HS loaded with the subject DSCs in Amendment 13.
NUHOMS®-32PT DSC in HSM-HS	24kW	The thermal performances of NUHOMS®-32PT DSC in HSM Model 202 is evaluated in Appendix V, Section V.4.4 (Table V.4-1).	
NUHOMS®-24PTH-S-LC DSC in HSM-HS	24kW	The thermal performances of NUHOMS®-24PTH-S-LC DSC in HSM Model 202 is evaluated in Appendix V, Section V.4.4 (Table V.4-1).	
NUHOMS®-24PTH-S or 24PTH-L DSC in HSM-HS	40.8 kW	The thermal performances of NUHOMS®-24PTH DSC in HSM-H is evaluated in Appendix P, Section P.4.4 and P.4.6.	The HSM-HS is thermally identical to the HSM-H module as described in Appendix U, Section U.1.2.1.2. Thermal evaluation performed for the HSM Model 202 are applicable for the HSM-HS loaded with the subject DSCs in Amendment 13.
NUHOMS®-61BTH Type 1 DSC in HSM-HS	22 kW	The thermal performances of NUHOMS®-61BTH DSC in HSM-H is evaluated in Appendix T, Section T.4.4 and T.4.6.	
NUHOMS®-61BTH Type 2 DSC in HSM-HS	31.2 kW		
NUHOMS®-69BTH DSC in HSM-HS	35 kW	The thermal evaluation of 69BTH DSC in HSM-H is provided in Appendix Y, Section Y.4.4 and Y.4.6 in Amendment 13.	The HSM-HS is thermally identical to the HSM-H module as described in Appendix U, Section U.1.2.1.2. Thermal evaluation performed for the HSM Model 202 are applicable for the HSM-HS loaded with the subject DSCs in Amendment 13.
NUHOMS®-37PTH DSC in HSM-HS	30 kW	The thermal evaluation of 37PTH DSC in HSM-H is provided in Appendix Z, Section Z.4.4 and Z.4.6 in Amendment 13. DSC shell temperatures based on thermal results of 32PTH1 DSC in the HSM-H with 31.2 kW heat load in Appendix U, Section U.4.4 are used as boundary conditions for the 37PTH DSC model to determine the maximum fuel cladding temperatures for storage conditions in the 37PTH DSC.	

RSI-11

RSI-11 Provide a table to (1) list the bounding thermal analyses of cask transfer for each type of DSC or for multiple types of DSCs and (2) summarize the bounding correlations among multiple types of DSCs under normal, off-normal and accident conditions.

The NUHOMS® Amendment 13 includes many changes or additions in types of DSCs, decay heat limits, poison materials, basket configuration, MOX fuel, etc. Under the premise that only some thermal analyses were selected and performed for cask transfer (OS200/OS200FC), either for each type of DSC or for multiple types of DSCs, to bound many proposed combinations of DSC type, decay heat limit, content (intact, damaged, and MOX fuels), and poison material (borated aluminum alloy, MMC. or Boral), loading zone configuration and transfer cask (OS200 or OS200FC), the applicant should provide a table to list and summarize the thermal analyses selected as the bounding cases for each type of DSC or for multiple types of DSCs. Staff needs this table to verify that the selected thermal analyses are appropriate for each type of DSC or multiple types of DSCs under normal, off-normal and accident conditions.

This information is required to demonstrate compliance with 10 CFR 72.236(f) and 72.236(1).

Response to RSI-11

In the Standardized NUHOMS® system (CoC 1004), the OS200/OS200FC transfer cask (TC) is currently licensed to transfer only the 32PTH1 DSC with heat loads of up to 40.8 kW.

In the current amendment application, the use of OS200/OS200FC TC is expanded to include previously approved DSCs (61BT, 32PT, 24PTH and 61BTH) with an aluminum internal sleeve within OS200/OS200FC TC and new DSCs (69BTH and 37PTH) within OS200/OS200FC TC without an aluminum internal sleeve.

For DSCs approved prior to Amendment 13, i.e., 61BT, 32PT, 24PTH and 61BTH, sensitivity analyses are performed based on the bounding normal transfer operations outside the fuel building to determine the effect on the maximum fuel cladding temperature. The sensitivity analyses show that the maximum fuel cladding temperatures during transfer in OS200/OS200FC transfer cask are bounded by the maximum fuel cladding temperatures previously determined for transfer operations. The details of the sensitivity analyses are presented in the following tables. In qualifying the OS200/OS200FC transfer cask for previously approved DSCs no new conditions/time limits are imposed for transfer operations in this amendment application.

For the new DSCs included in Amendment 13, i.e., 69BTH and 37PTH DSCs, the OS200/OS200FC transfer cask thermal analyses presented in Appendix U, Section U.4.5 of the UFSAR for 32PTH1 DSC are used to obtain the DSC shell profiles for normal /off-normal and accident conditions. The DSC shell profiles are then used with the detailed models of the 69BTH and 37PTH DSCs to determine the maximum fuel cladding temperatures for normal /off-normal and accident conditions. The details of the thermal analyses are presented in the corresponding tables.

The justifications for using the DSC shell profiles from the thermal analyses of OS200/OS200FC TC with 32PTH1 DSC (presented in Appendix U, Section U.4.5 of the UFSAR Rev.11) for 69BTH and 37PTH DSCs are presented in the UFSAR Appendix Y, Section Y.4.5 and Appendix Z, Section Z.4.5, respectively.

Further, no new TCs are considered for the 24PHB, 24PTH-S-LC and 32PTH1 DSCs for transfer operations.

The MOX fuel assemblies are considered only in the 69BTH and 37PTH DSCs. The effects of the MOX fuel assemblies on thermal performance are discussed in response to RSI-6.

Failed fuel assemblies are added only to the allowable content of 61BTH and 24PTH DSCs. The thermal evaluation of the failed fuel assemblies in the 24PTH DSC is described in the SAR, Appendix P, Section P.4.6.9. The thermal evaluation of the failed fuel assemblies in the 61BTH DSC is described in the SAR, Appendix T, Section T.4.6.9.

The effects of the damaged fuel assemblies on the thermal performance are discussed in Appendix Y, Sections Y.4.6.2 and Y.4.6.7 for the 69BTH DSC and in Appendix Z, Sections Z.4.6.2 and Z.4.6.8 for the 37PTH DSC. The location and the number of damaged fuel assemblies in the other DSCs are not subject of the Amendment 13 application.

The following tables present the list of thermal evaluations performed for this amendment application to allow the use of OS200/OS200FC transfer casks with the previously approved DSCs (61BT, 32PT, 24PTH and 61BTH) and new DSCs (69BTH and 37PTH).

Summary of the system configuration in the SAR, Appendix P, Section P.4.1 is revised to clarify that the application of the OS200TC is not extended to include the 24PTH-S-LC DSC.

NUHOMS® 61BT in OS200 Transfer Cask

DSC/TC Type	Heat Load	Bounding Analysis	SAR Section	Conclusion
61BT in OS200 with internal sleeve	18.3 kW	DSC in TC Normal Hot Condition with 100°F ambient for horizontal transfer operations as listed in Table K.4-2 of UFSAR Rev.11.	Section K.4.9	The maximum temperatures of the 61BT DSC in OS200 TC as shown in Section K.4.9 of the SAR are bounded by the DSC temperatures during transfer in OS197.

NUHOMS® 32PT in OS200 Transfer Cask

DSC/TC Type	Heat Load	Bounding Analysis	SAR Section	Conclusion
32PT in OS200 with internal sleeve	24 kW	DSC in TC Normal Hot Condition with 100°F ambient for horizontal transfer operations as listed in Table M.4-2 and Table M.4-3 of UFSAR Rev.11.	Section M.4.4.1.6.2	The maximum temperatures of the 32PT DSC in OS200 TC as shown in Section M.4.4.1.6.2 of the SAR are bounded by the DSC temperatures during transfer in OS197.

NUHOMS® 24PTH in OS200/OS200FC Transfer Cask

DSC/TC Type	Heat Load	Bounding Sensitivity Analysis	SAR Section	Conclusion
24PTH-S/-L Type 1 in OS200FC with internal sleeve	40.8 kW	DSC in TC Normal Hot Condition with 100°F ambient for horizontal transfer operations as listed in Table P.4-14 and P.4-15 of UFSAR Rev.11.	Section P.4.6.8	The maximum temperatures of the 24PTH DSC in OS200FC TC as shown in Section P.4.6.8 of the SAR are bounded by the DSC temperatures during transfer in OS197
24PTH-S/-L Type 1 in OS200FC with internal sleeve	31.2 kW	DSC in TC Normal Hot Condition with 100°F ambient for horizontal transfer operations as listed in Table P.4-14 and P.4-16 of UFSAR Rev.11.	Section P.4.6.8	The maximum temperatures of the 24PTH DSC in OS200FC TC as shown in Section P.4.6.8 of the SAR are bounded by the DSC temperatures during transfer in OS197

An explicit thermal sensitivity analysis is not performed for the 24PTH-S Type 2 DSCs for transfer in OS200FC TC with a maximum heat load of 31.2 kW. Except for the rail inserts, the basket designs of the 24PTH-S/-L Type 1 DSC is identical to the 24PTH-S/-L Type 2 DSC. Therefore, a similar thermal behavior is expected for 24PTH Type 1 and Type 2 DCSs. Since analysis of the 24PTH-S Type 1 DSC in the OS200FC TC (SAR Section P.4.6.8) demonstrates large margins to fuel cladding temperature limit, which are also bounded by the evaluation for transfer in the OS197FC TC, the maximum temperatures and time limits determined for transfer in OS197FC TC are considered bounding for transfer in OS200FC TC for all allowable types of the 24PTH DSC.

NUHOMS® 61BTH in OS200/OS200FC Transfer Cask

DSC/TC Type	Heat Load	Bounding Sensitivity Analysis	SAR Section	Conclusion
61BTH Type 1 in OS200/OS200FC with internal sleeve	22 kW	DSC in TC Normal Hot Condition with 100°F ambient for horizontal transfer operations as listed in Table T.4-12 and Table T.4-13 of UFSAR Rev.11.	Section T.4.5.5	There is negligible impact on the maximum temperatures of the 61BTH Type 1 DSC in OS200/OS200FC TC as shown in Section T.4.5.5.2 of the SAR.
61BTH Type 2 in OS200FC with internal sleeve	31.2 kW	DSC in TC Normal Hot Condition with 100°F ambient for horizontal transfer operations as listed in Table T.4-12 and Table T.4-14 of UFSAR Rev.11.	Section T.4.5.5	There is negligible impact on the maximum temperatures of the 61BTH Type 1 DSC in OS200FC TC as shown in Section T.4.5.5.2 of the SAR.

NUHOMS® 69BTH in OS200/OS200FC Transfer Cask

DSC/TC Type	Heat Load	Bounding Analysis	SAR Section	Conclusion
69BTH in OS200FC	35.0 kW	For all operating conditions, DSC shell profiles for the 69BTH DSC with 35 kW are from 32PTH1 DSC thermal analysis with 40.8 kW as noted in Section Y.4.5.3 of the SAR. <u>Normal/Off-Normal Transfer</u> DSC in TC Normal Hot Condition with 100°F ambient (Load Case# T6 in Table Y.4-4) is the bounding case as discussed in Section Y.4.6.5 of the SAR. <u>Accident Transfer</u> DSC in TC Loss of air circulation, loss of neutron shield, and no sunshade (Load Case # T14 in Table Y.4-4) is the bounding case as discussed in Section Y.4.5.3 of the SAR.	Section Y.4.5 and Y.4.6	As shown in Table Y.4-9 of the SAR, the maximum temperature of the fuel cladding remains below the allowable limits.
69BTH in OS200/OS200FC	24.0 kW	For all operating conditions, DSC shell profiles for the 69BTH DSC with 24 kW are from 32PTH1 DSC thermal analysis with 24 kW as noted in Section Y.4.5.3 of the SAR. <u>Normal/Off-Normal Transfer</u> DSC in TC Normal Hot Condition with 100°F (Load Case# T1 in Table Y.4-4) is the bounding case as discussed in Section Y.4.6.5 of the SAR. <u>Accident Transfer</u> Bounded by the 35 kW load case, Load Case # T14 in Table Y.4-4 of the SAR.	Section Y.4.5 and Y.4.6	As shown in Table Y.4-9 of the SAR, the maximum temperature of the fuel cladding remains below the allowable limits.

NUHOMS® 37PTH in OS200/OS200FC Transfer Cask

DSC Type	Heat Load	Bounding Analysis	SAR Section	Conclusion
37PTH in OS200FC	30.0 kW	For all operating conditions, DSC shell profiles for the 37PTH DSC with 30 kW are from 32PTH1 DSC thermal analysis with 31.2 kW as noted in Section Z.4.5.2 of the SAR. <u>Normal/Off-Normal Transfer</u> DSC in TC, Vertical Operations with 140°F ambient (Load Case# T5) is bounding case as shown in Table Z.4-3 of the SAR. <u>Accident Transfer</u> DSC in TC Loss of air circulation, loss of neutron shield, and no sunshade (Load Case # T8 in Table Z.4-3) is the bounding case as discussed in Section Z.4.5.3 of the SAR.	Section Z.4.5 and Z.4.6	As shown in Table Z.4-3 of the SAR, the maximum temperature of the fuel cladding remains below the allowable limits.
37PTH in OS200/OS200FC	24.0 kW	<u>Normal/Off-Normal Transfer</u> DSC in TC, Vertical Operations with 140°F ambient (Load Case# T12) is bounding case as shown in Table Z.4-3 of the SAR. <u>Accident Transfer</u> Bounded by the 30 kW load case, Load Case # T8 in Table Z.4-3 of the SAR.	Section Z.4.5 and Z.4.6	As shown in Table Z.4-3 of the SAR, the maximum temperature of the fuel cladding remains below the allowable limits.

RSI-12

RSI-12 Provide a table to (1) list the transfer time limit for each type of DSC, under all proposed conditions, and (2) summarize the bounding correlations among multiple types of DSCs under normal conditions.

The NUHOMS® Amendment 13 includes many changes or additions in types of DSCs, decay heat limits, poison materials, basket configuration, MOX fuel, etc. Under the premise that only some thermal analyses were selected and performed for cask transfer (OS200/OS200FC), either for each type of DSC or for multiple types of DSCs, to bound many proposed combinations of DSC type, decay heat limit, content (intact, damaged, and MOX fuels), and poison material (borated aluminum alloy, MMC, or Boral), loading zone configuration and transfer cask (OS200 or OS200FC), the applicant should provide a table to list the transfer time limit for each combination and explain why the transfer time limit of one combination can bound or be applied to another combination. Staff needs this table to verify that the selected thermal analyses are appropriate and most conservative for each type of DSC or for multiple types of DSCs under normal conditions.

This information is required to demonstrate compliance with 10 CFR 72.236(f).

Response to RSI-12

As described in response to RSI-11, in the Standardized NUHOMS® system (CoC 1004), the OS200/OS200FC transfer cask (TC) is currently licensed to transfer only the 32PTH1 DSCs with heat loads of up to 40.8 kW.

In the current amendment application, the use of OS200/OS200FC transfer cask is expanded to include previously approved DSCs (61BT, 32PT, 24PTH and 61BTH) with an aluminum internal sleeve in OS200/OS200FC TC and new DSCs (69BTH and 37PTH) without an aluminum internal sleeve in OS200/OS200FC TC.

The details of the thermal analyses performed to qualify new DSCs 69BTH and 37PTH are presented in response for RSI# 11.

The details of the thermal analyses performed to qualify the previously approved DSCs (61BT, 32PT, 24PTH and 61BTH) for transfer in OS200/OS200FC are presented as response to RSI-11. In qualifying the OS200/OS200FC transfer cask for previously approved DSCs no new conditions/time limits are imposed for transfer operations in this amendment application.

For example, in the Standardized NUHOMS® system (CoC 1004), 61BT DSC is licensed for transfer in OS197 TC without any time limits. The same transfer condition without any time limits is considered for transfer of 61BT DSC in OS200 transfer cask based on the analysis presented in Section K.4.9 of the SAR.

Similarly in the Standardized NUHOMS® system (CoC 1004), 61BTH Type 2 DSC is licensed for transfer in OS197FC-B TC with a transfer time limit of 26 hours for HLZC # 5,6, 8 and a transfer time limit of 13 hours for HLZC # 7. The same transfer time limits are considered for transfer of 61BTH Type 2 DSC in OS200FC transfer cask as shown in Section T.4.5.5 of the SAR.

The MOX fuel assemblies are considered only in the 69BTH and 37PTH DSCs. The effects of the MOX fuel assemblies on thermal performance are discussed in response to RSI-6.

Failed fuel assemblies are added only to the allowable content of 61BTH and 24PTH DSCs. The thermal evaluation of the failed fuel assemblies in the 24PTH DSC is described in the SAR, Appendix P, Section P.4.6.9. The thermal evaluation of the failed fuel assemblies in the 61BTH DSC is described in the SAR, Appendix T, Section T.4.6.9.

The effects of the damaged fuel assemblies on the thermal performance are discussed in Appendix Y, Sections Y.4.6.2 and Y.4.6.7 for the 69BTH DSC and in Appendix Z, Sections Z.4.6.2 and Z.4.6.8 for the 37PTH DSC. The location and the number of damaged fuel assemblies in the other DSCs are not subject of the Amendment 13 application.

The following table provides the time limit for transfer operations for various DSCs in OS200/OS200FC transfer cask with appropriate references from SAR/UFSAR.

DSC	Maximum Heat load	Time limit for transfer in OS200/OS200FC TC	Reference for Justification/Notes
61BT in OS200	18.3 kW	No time limit	SAR Section K.4.9 (includes justification why previous transfer results remain bounding for OS200 TC)
32PT in OS200	24 kW	No time limit	SAR Section M.4.4.1.6.2 (includes justification why previous transfer results remain bounding for OS200 TC)
24PTH-S or 24PTH-L with aluminum inserts in rails and transfer in OS200FC	40.8 kW	11.5 hours	Table P.4-14 and Table P.4-20 of the UFSAR rev.11. SAR Section P.4.6.8 (includes justification why previous transfer results remain bounding for OS200FC TC)
24PTH-S or 24PTH-L with aluminum insert in rails and transfer in OS200	31.2 kW	No time limit	Table P.4-14 and Table P.4-20 of the UFSAR rev.11. SAR Section P.4.6.8 (includes justification why previous transfer results remain bounding for OS200 TC)
24PTH-S or 24PTH-L without aluminum insert in rails and transfer in OS200FC	31.2 kW	27.3 hours	Table P.4-14 and Table P.4-20 of the UFSAR rev.11. See justification in RSI-11 response.
61BTH Type 1 in OS200	22 kW	No time limit	Section T.4.5.4 of the UFSAR rev.11 and SAR Section T.4.5.5.2. (includes justification why previous transfer results are applicable for OS200 TC)
61BTH Type 2 in OS200FC	31.2 kW	26 hours 13 hours for HLZC#7	Section T.4.5.4 of the UFSAR rev.11 and SAR Section T.4.5.5.2. (includes justification why previous transfer results remain bounding for OS200FC TC)
69BTH in OS200FC	35 kW	13 hours	SAR Section Y.4.5.4 Time limit applies for heat loads greater than 24 kW up to 35 kW.
69BTH in OS200/OS200FC	24 kW	No time limit	SAR Section Y.4.5.4
37PTH in OS200FC	30 kW	14 hours	SAR Section Z.4.5.4 Time limit applies for heat loads greater than 24 kW up to 30 kW.
37PTH in OS200/OS200FC	24 kW	No time limit	SAR Section Z.4.5.4

RSI-13

RSI-13 Describe the geometrical configuration and characteristics of the fuel debris, and justify its impacts on the adjacent intact fuel assemblies or the cask.

For damaged fuel assemblies in the 69BTH, 37PTH and other DSCs, fuel debris such as broken rods, loose pellets and/or of cladding may exist in the fuel compartment. The fuel debris can be in a type of rubble fuel assembly which may be concentrated in a smaller area, create hot spots in the cask, and increase the cladding temperatures of the adjacent intact assemblies. The applicant should provide more information about the fuel debris, and should provide a thermal analysis if the fuel debris exists as a type of rubble fuel assembly.

This information is required to demonstrate compliance with 10 CFR 72.236(a) and 72.236(l).

Response to RSI-13

The definition of the damaged BWR/PWR fuel assemblies in the SAR, Appendices Y.2 (Table Y.2-1) and Z.2 (Table Z.2-1) limits the extent of damage in the fuel assembly such that a fuel assembly is able to be handled by normal means. This restriction does not allow the presence of loose pellets, loose cladding or loose rods in a damaged fuel assembly. Therefore, no fuel rubble exists in the damaged fuel compartments during normal and off-normal conditions.

For the postulated drop accident condition, the cladding of damaged fuel assemblies can experience further damage, which in the worst case the damaged fuel assemblies become rubble. The effects of fuel rubble due to the accident conditions are evaluated in Appendix Y.4, Section Y.4.6.7 for 69BTH DSC and in Appendix Z.4, Section Z.4.6.8 for 37PTH DSC.

Damaged fuel assemblies for other DSC were approved under previous amendments.

Failed fuel assemblies are added only to the allowable content of 61BTH and 24PTH DSCs. The thermal evaluation of the failed fuel assemblies in the 24PTH DSC is described in the SAR, Appendix P, Section P.4.6.9. The thermal evaluation of the failed fuel assemblies in the 61BTH DSC is described in the SAR, Appendix T, Section T.4.6.9.

RSI-14

RSI-14 Provide more information regarding the optional features for the OS200FC Transfer Cask (TC) recovery system, and identify it as either an active cooling or a passive cooling system. Section Z.4.5 of the SAR states that the design of the OS200FC TC is identical to the design of the OS200 TC with the exception of the optional features, such as a slotted cask lid and cask bottom spacer to allow air circulation through the TC/DSC annulus as a recovery system when the time limit for transfer operation is exceeded. The applicant should provide more details regarding the optional features to help identify the recovery system as either an active cooling or a passive cooling system.

This information is required to demonstrate compliance with 10 CFR 72.236(f).

Response to RSI-14

The OS200/OS200FC transfer cask, described in SAR Section Z.4.5 of this amendment application, is identical to the transfer cask that is currently licensed in Standardized NUHOMS® system (CoC 1004) and is approved to transfer 32PTH1 DSC with heat load of up to 40.8 kW. The OS200/OS200FC transfer cask is described in Appendix U, Section U.1.2.1.3 of the UFSAR Rev. 11.

As described in Appendix U, Section U.1.2.1.3 of the UFSAR Rev. 11, the OS200 TC is provided with an optional top lid with design features which enable an exit path for air circulation through the TC/DSC annulus. The TC when used with this optional top lid is designated as OS200FC TC.

The air circulation feature of the OS200FC TC is an active cooling system and is only relied upon as one of the recovery options during transfer under normal/off-normal conditions if the heat loads are greater than 24 kW and the specific time limits cannot be met to maintain the fuel cladding temperature below 752°F. The other mitigating options are described in Technical Specification 3.1.3.

The air circulation recovery system employed in the OS200FC TC is similar to the recovery system currently licensed and employed in OS197FC/OS197HFC with 24PTH DSC described in Appendix P, Chapter P.1 of and OS197FC-B with 61BTH DSC described in Appendix T, Chapter T.1 of UFSAR Rev. 11.

The air circulation feature of the OS200FC TC is not relied upon for heat removal during postulated accident conditions for transfer operations to maintain the fuel cladding temperature below 1058°F.

RSI-15

RSI-15 Describe/summarize the existence of gaps in thermal evaluations for the 37PTH and 69BTH DSCs under off-normal and accident conditions.

The applicant assumed the existence of gaps (1) between the basket outer surface and canister inner surface, (2) between shield plugs and canister shell inner surface, (3) between basket rails and components, (4) between any two adjacent plates or components within the cross section of fuel compartments, (5) in the axial direction between rail pieces, (6) between any two adjacent plates between the shield plugs and canister cover plates, and (7) in the axial gap between the canister inner bottom plate and the bottom basket assembly. The applicant should provide a table to summarize whether these gaps exist or not for thermal evaluations during off-normal and accident conditions (for each type of DSC). The information provided will help staff accelerate the thermal review.

This information is required to demonstrate compliance with 10 CFR 72.236(f).

Response to RSI-15

The gaps used in thermal evaluations for 37PTH and 69BTH DSCs under off-normal conditions are described in Appendix Y, Section Y.4.6.2 and Appendix Z, Section Z.4.6.2 of the SAR, respectively. These gaps are conservatively used in steady-state DSC thermal analyses for normal, off-normal and accident conditions.

The analyses of the accident conditions for OS200/OS200FC TC presented in Appendix U, Section U.4.5.4.2 of the UFSAR rev.11, demonstrated that the "loss of neutron shield" accident scenario, which also includes the loss of air circulation, results in the highest maximum component temperatures of OS200FC TC. As described in the UFSAR, Appendix U, Section U.4.5.4.2, the "loss of neutron shield" accident is a steady state analysis and bounds the maximum temperatures for the fire accident, which is a transient analysis. Since the bounding accident case is resulting from a steady state condition, the gaps considered for normal/off-normal conditions are applicable to evaluate the bounding maximum temperatures for accident conditions.

RSI-16

RSI-16 Justify the assumptions made in modeling and application of the seismic loads to the finite element model. The modeling approach may not be consistent with similar analyses approved for models included in Amendment 10 to the Standardized NUHOMS® Systems (CoC 72-1004). Explain the differences between the Amendment 10 analyses and the one performed for Amendment 13.

For the 69BTH and 37PTH HSM model the standard seismic loads were enveloped by a combined loading of 0.36g axial + 0.41g transverse + 1.25g vertical for Level "C" and for the HSM-HS model they were enveloped by a combined loading of 1.6g axial + 2.0g transverse + 3.464g vertical for Level "D". These seismic "g" values are quite different than those that were used for high seismic HSM-HS models in the past under Amendment 10.

This information is required to demonstrate compliance with 10 CFR 72.236(l).

Response to RSI-16

The assumptions made in the modeling and application of the seismic loads to the finite element models are generally the same between Amendment 10 and Amendment 13. The attached tables summarize the seismic inputs, analysis models, and assumptions used for the seismic analysis of the 61BTH and 32PTH1 DSCs when stored in the HSM-H and HSM-HS, respectively in Amendment 10, compared with those used in Amendment 13 for the 37PTH and 69BTH DSCs in the HSM-H/HSM-HS.

Seismic inputs, analysis models, and assumptions used for the seismic analysis (Part 1 of 3)

	Amendment 10					Amendment 13				
	Component	**HSM-H (Level C)**		**HSM-HS (Level D)**		**Component**	**HSM-H (Level C)**		**HSM-HS (Level D)**	
Seismic Design Criteria	32PTH1 [U.2.2.3]	0.30g H	RG 1.60	1.0g H	Enhanced RG 1.60 (Note 1)	37PTH	0.30g H	RG 1.60	1.0g H	Enhanced RG 1.60 (Note 1)
		0.25g V	RG 1.60	1.0g V		[Z.2.2.3]	0.25g V	RG 1.60	1.0g V	
	61BTH [T.2.2.3]	0.30g H	RG 1.60	Not in Amendment 10 (Note 2)		69BTH	0.30g H	RG 1.60	1.0g H	Enhanced RG 1.60 (Note 1)
		0.20g V	RG 1.60			[Y.2.2.3]	0.25g V	RG 1.60	1.0g V	
Amplified Accelerations Used (Note 3)	**Component**	**HSM-H**		**HSM-HS**		**Component**	**HSM-H**		**HSM-HS**	
	32PTH1 DSC [U.3.7.2.1]	0.41g H	Transverse	2.0g H	Transverse	37PTH [Z.3.7.2.1] [Z.3.7.2.2]	0.41g H	Transverse	2.0g H	Transverse
		0.36g H	Axial	1.6g H	Axial		0.36g H	Axial	1.6g H	Axial
		0.25g V	Vertical	1.0g V	Vertical		0.25g V	Vertical	1.0g	Vertical
	61BTH DSC [T.2.2.3]	0.41g H	Transverse	Not in Amendment 10		69BTH [Y.3.7.2.1]	0.41g H	Transverse	2.0g H	Transverse
		0.36g H	Axial				0.36g H	Axial	1.6g H	Axial
		0.20g	Vertical				0.25g V	Vertical	1.0g	Vertical
	HSM-H (Loaded with 32PTH1 DSC) [U.3.7.2.3.1]	0.37g H	Transverse	N / A		HSM-H (Loaded with 37PTH DSC) [Z.3.7.2.3.] [U.3.7.2.3.1] Note 6	0.37g H	Transverse	N / A	
		0.33g H	Axial				0.33g H	Axial		
		0.25g V	Vertical				0.25g V	Vertical		
	HSM-H (Loaded with 61BTH DSC) [T.2.2.3]	0.37g H	Transverse	N / A		HSM-H (Loaded with 69BTH DSC) [Y.3.7.2.3] [U.3.7.2.4] Note 6	0.37g H	Transverse	N / A	
		0.33g H	Axial				0.33g H	Axial		
		0.20g V	Vertical				0.25g V	Vertical		
	HSM-HS (Loaded with 32PTH1 DSC) [U.3.7.2.4.1]	N / A		1.61g H	Transverse	HSM-HS (Loaded with 37PTH DSC) [Z.3.7.2.4] [U.3.7.2.4.1] Note 6	N / A		1.61g H	Transverse
				1.38g H	Axial				1.38g H	Axial
				1.0g V	Vertical (Note 4)				1.0g V	Vertical (Note 4)

See notes at the end of the table.

Seismic inputs, analysis models, and assumptions used for the seismic analysis (Part 2 of 3)

	Amendment 10			Amendment 13		
Models Used and Analysis Description	32PTH1 Shell Assembly	[U.3.6.1.2]	3D Top End and Bottom End ANSYS Half Symmetric Models were used. Equivalent Static Analysis: Level C (Axial 0.36g, Transverse 0.41g, Vertical 0.25g) and Level D (Axial 1.6g, Transverse 2.0g, Vertical 1.0g)	37PTH Shell Assembly (Note 7)	[Z.3.6.1.2] [Z.3.7.2.1]	3D Top End and Bottom End ANSYS Half Symmetric Models were used. Equivalent Static Analysis: Level C (Axial 0.36g, Transverse 0.41g, Vertical 0.25g) and Level D (Axial 1.6g, Transverse 2.0g, Vertical 1.0g)
	32PTH1 Basket Assembly (Note 4)	[U.3.6.1.3]	Basket 3D 15 inch long section was used. Canister shell, fuel compartments tubes, support plates, fusion welds and aluminum rails were considered. Equivalent Static Analysis: Level C (Axial 0.36g, Transverse 0.41g, Vertical 0.25g) and Level D (Axial 1.6g, Transverse 2.0g, Vertical 1.0g)	37PTH Basket Assembly (Notes 4, 7)	[Z.3.6.1.3] [Z.3.7.2.2] Note 4	Basket 3D 1 inch long section was used. Canister shell, fuel compartments grid assembly, welds and aluminum rails were considered. Equivalent Static Analysis: Level C (Axial 0.36g, Transverse 0.41g, Vertical 0.25g) and Level D (Axial 1.6g, Transverse 2.0g, Vertical 1.0g)
	61BTH Shell Assembly	[T.3.6.1.2]	3D Top End and Bottom End ANSYS Half Symmetric Models were used. Equivalent Static Analysis: Level C (Axial 0.36g, Transverse 0.41g, Vertical 0.25g)	69BTH Shell Assembly (Note 7)	[Y.3.6.1.2] [Y.3.7.2.1]	3D Top End and Bottom End ANSYS Half Symmetric Models were used. Equivalent Static Analysis: Level C (Axial 0.36g, Transverse 0.41g, Vertical 0.25g) and Level D (Axial 1.6g, Transverse 2.0g, Vertical 1.0g)
	61BTH Basket Assembly	[T.3.6.1.3] Note 5	Basket 3D 3 inch long section was used. Canister shell, fuel compartments tubes, compartment wraps, welds and aluminum or stainless steel rails were considered. Equivalent Static Analysis: Level C (Axial 0.36g, Transverse 0.41g, Vertical 0.25g)	69BTH Basket Assembly (Notes 4, 7)	[Y.3.6.1.3] Note 5	Basket 3D 1 inch long section was used. Canister shell, fuel compartments tubes, compartment wraps, welds and aluminum rails were considered. Equivalent Static Analysis: Level C (Axial 0.36g, Transverse 0.41g, Vertical 0.25g) and Level D (Axial 1.6g, Transverse 2.0g, Vertical 1.0g)
	HSM-H (Loaded with 32PTH1 DSC) [U.3.7.2.3.1]	[U.3.7.11.6]	Same HSM-H ANSYS model as in P.3.7.11.6 Modified seismic criteria based on 0.30gH and 0.25g V Equivalent static analysis w/ 0.37g/0.37g/0.25g and SRSS	HSM-H (Loaded with 37PTH1 DSC) [Z.3.7.2.3]		Evaluation in [U.3.7.2.3] is applicable
	HSM-H (Loaded with 61BTH DSC) [T.2.2.3]	[T.3.7.2]	Evaluation bounded by P.3.7.2. Uses ANSYS model in P.3.7.11.6	HSM-H (Loaded with 69BTH DSC) [Y.3.7.2.3] [U.3.7.2.4] Note 6		Evaluation bounded by U.3.7.2.4. Uses same ANSYS model as in P.3.7.11.6
	HSM-HS (Loaded with 32PTH1 DSC) [U.3.7.2.4.1]	[U.3.7.11.6] [U.3.7.2.4]	Same HSM-H ANSYS model as in P.3.7.11.6 Modified seismic criteria based on 1.0g H and 1.0g V Equivalent static analysis w/ 1.61g/1.61g/1.0g and SRSS	HSM-HS (Loaded with 37PTH DSC) [Z.3.7.2.4]		Evaluation in [U.3.7.2.4] is applicable

See notes at the end of the table.

Seismic inputs, analysis models, and assumptions used for the seismic analysis (Part 3 of 3)

Notes

(1) Enhanced RG 160 means amplified region of the RG 1.60 spectra is extended from 9 Hz to 16 Hz for the horizontal and to 20 Hz for the vertical directions. See Figure U.2-4.

(2) The 61BTH in HSM-HS is not part of Amendment 10.

(3) These amplifications are based on the frequency analysis of HSM-H/HSM-HS loaded with a respective DSC using the RG 1.60 or the "Enhanced" RG 1.60 amplifications, as applicable, at 3% damping for the DSC and 7% damping for the HSM/HSM-HS.

(4) In the basket assembly analysis of the 32PTH1, 37PTH, and 69BTH in HSM-HS the vertical acceleration including gravity is 3.464g (instead of 2.0g) in order for the model to remain stable due to the lateral 2g acceleration in an equivalent static analysis.

(5) Enveloping load of 2g axial+2g Transverse+ 2g Vertical is used in the seismic analysis (SAR Appendix T, Section T.3.6.1.3.2).

(6) The evaluations of the HSM-H and HSM-HS with 32PTH1 in Appendix U are applicable.

(7) 37PTH and 69BTH ANSYS Models are based on the models developed for the MP197HB application submittal (Docket No. 71-9302, TAC No. L24336).

RSI-17

RSI-17 Provide details in a tabular form for the planned heat loads and limits of high burn up, and the seismic "g" loads in all three orthogonal directions for the six HSM-HS uses (namely the 61BT, 32PT, 24PTH, 61BTH, 69BTH, and 37PTH DSCs).

It is not clear what heat load, and which limits of high burn up, and seismic "g" loads are planned for the high seismic capability (HSM-HS) for the previously approved canisters under NUHOMS® Amendment 10.

This information is needed to demonstrate compliance with 10 CFR 72.236(l).

Response to RSI-17

The following table provides a summary of the maximum heat load, maximum weight, maximum burnup, and the seismic loading used in the Amendment 13 storage evaluation of all the DSCs in the HSM-HS.

NUHOMS® DSCs Heat Load, Burnup Limits, and Seismic Loads for On-Site Storage in the High Seismic HSM-HS

<table>
<tr><th>DSC Type</th><th>Max Storage Heat Load (kW)</th><th>DSC Maximum Burnup (GWd/MTU)</th><th>On-Site Storage Basic Seismic Criteria(1)</th><th>On-Site Storage Amplified Accelerations Used for DSC Evaluation(2)</th></tr>
<tr><td>61BT</td><td>18.3</td><td>40</td><td rowspan="13">HSM-HS:
· 1.0g H
· 1.0g V</td><td rowspan="13">HSM-HS:
· 2.0g Transverse
· 1.6g Axial
· 1.0g Vertical</td></tr>
<tr><td>32PT</td><td>24.0</td><td>55</td></tr>
<tr><td rowspan="2">24PTH</td><td>24.0 (24PTH-S-LC)</td><td rowspan="2">62</td></tr>
<tr><td>40.8</td></tr>
<tr><td rowspan="2">61BTH</td><td>22.0 (Type 1)</td><td rowspan="2">62</td></tr>
<tr><td>31.2 (Type 2)</td></tr>
<tr><td rowspan="2">32PTH1</td><td>31.2 (Type 2)</td><td rowspan="2">62</td></tr>
<tr><td>40.8 (Type 1)</td></tr>
<tr><td rowspan="2">69BTH</td><td>24.0</td><td rowspan="2">70</td></tr>
<tr><td>35.0</td></tr>
<tr><td rowspan="2">37PTH</td><td>24.0</td><td rowspan="2">62</td></tr>
<tr><td>30.0</td></tr>
</table>

Notes:

(1) The basic seismic criteria are based on an enhanced RG 1.61 Response Spectra. Enhanced RG 1.60 means the amplified region of the RG 1.60 spectra is extended from 9 Hz to 16 Hz in the horizontal direction and from 9 HZ to 20 Hz in the vertical direction, as shown in Appendix U, Figure U.2-4.

(2) These accelerations result from a frequency analysis of the loaded HSM-HS loaded with the heaviest DSC which returns the largest spectral acceleration based on the enhanced RG 1.60 response spectra at 3% damping. Thus, these spectral accelerations are bounding for all evaluated DSCs.

RSI-18

RSI-18 Provide in a tabular form comparison and significant changes/modifications for all the structural analysis models included in NUHOMS® Amendment 13, with the previously approved NUHOMS® Amendment 10, and the currently in-progress NUHOMS® Amendment 11. Also provide a concise table with all the models showing actual design stresses and the allowable stresses for the governing load conditions.

It is not clear with multiple models and multiple transfer casks, adding high heat load, adding high burn up fuel, adding MOX fuel, and other miscellaneous "Materials", which loading governs for which model.

This information is needed to demonstrate compliance with 10 CFR 72.236(l).

Response to RSI-18

See response to RSI-16 for a comparison of significant changes/modifications for the structural analysis models in Amendment 13 with previously approved Amendment 10. See response to RSI-17 for a summary of the main design parameters (maximum heat load, maximum burnup, maximum seismic accelerations) for all the DSCs in Amendment 13 to be allowed for storage in the HSM-HS.

The following table provides a summary of the controlling load combinations and maximum stress intensities for the DSC components in Amendment 10 and Amendment 13 (and pertinent clarifications of the Amendment 11 results). Applicable SAR pages are revised to provide corrections.

Amendment	DSC Type	DSC Component	Load Condition/Service Level	Controlling Load Combination	Maximum Stress Intensity (ksi)	Allowable Stress Intensity (ksi)	SAR Reference
Amendment 10	32PTH1	Shell Assembly	Normal/Off Normal Levels A/B	UL-6	26.43	27.2	Table U.3.7-18
			Accident Level C	HSM-8	21.83	22.4	Table U.3.7-19
			Accident Level D	TR-10	42.74	44.4	Table U.3.7-20
		Basket Assembly Type 1	Accident Level D	High Seismic	26.6	44.4	Table U.3.7-2
			Accident Level D	75g Side Drop	39.38	44.4	Table U.3.7.5
		Basket Assembly Type 2	Accident Level D	Seismic	26.2	44.4	Table U.3.7-3
			Accident Level D	75g Side Drop	51.45	57.06	Table U.3.7-8
	61BTH Type 1	Shell Assembly	Normal/Off Normal Levels A/B	UL-5, UL-6	25.44	29.1	Table T.3.7-12
			Accident Level C	HSM-8 Seismic	33.01	34.8	Table T.3.7-14
			Accident Level D	TR-10 Side Drop	58.98	59.6	Table T.3.7-16
		Basket Assembly	Accident Level D	TR-10 Side Drop	44.37	56.97	Table T.3.7-16
	61BTH Type 2	Shell Assembly	Normal/Off Normal Levels A/B	HSM-1	59.44	60.0	Table T.3.7-13
			Accident Level C	UL-7	33.06	34.0	Table T.3.7-15
			Accident Level D	TR-10 Side Drop	55.86	57.6	Table T.3.7-17
		Basket Assembly	Accident Level D	TR-10 Side Drop	45.95	56.97	Table T.3.7-17
Amendment 11	61BT and 32PT	Amendment 11 incorporates the OS197L transfer cask (TC) for use with the 61BT and 32PT DSCs, with maximum decay heat of 13 kW. The stress evaluations for these DSCs is based on maximum decay heat load of 18.3 kW and 24 kW as presented in Appendices K and M, respectively. Thus, there were no revisions to the stress evaluations in those appendices as part of Amendment 11. The stress evaluation of the OS197L TC is contained in Appendix W (W.3 Structural Evaluation).					
Amendment 13 Note 1	61BT	Shell Assembly	Accident Level D	High Seismic	40.04	45.5	Table K.3.7-17
		Basket Assembly	Accident Level D	High Seismic	23.7	57.1	Table K.3.7-16
	32PT (Note 2)	Shell Assembly	Accident Level D	High Seismic	48.9	58.6	Table M.3.7-14
		Basket Assembly-- 1 piece R90	Accident Level D	High Seismic	60.6	77.0	Table M.3.7-12
		Basket Assembly-- 3 piece R90	Accident Level D	High Seismic	69.3	77.0	Table M.3.7-12
	24PTH	Shell Assembly	Accident Level D	High Seismic	48.9	58.6	Table P.3.7-28
		Basket Assembly	Accident Level D	High Seismic	20.8	44.4	Table P.3.7-26
	61BTH Type 1	Shell Assembly	Accident Level D	High Seismic	40.04	45.5	Table T.3.7-20
		Basket Assembly	Accident Level D	High Seismic	23.7	57.1	Table T.3.7-19
	61BTH Type 2	Shell Assembly	Accident Level D	High Seismic	48.9	58.6	Table T.3.7-23
		Basket Assembly	Accident Level D	High Seismic	23.7	57.1	Table T.3.7-19
	69BTH	Shell Assembly	Normal/Off Normal Levels A/B	TR-8	43.2	52.5	Table Y.3.7-9
			Accident Level C	UL-8	41.3	63.0	Table Y.3.7-10
			Accident Level D	UL-8	55.2	63.0	Table Y.3.7-11
		Basket Assembly	Accident Level C	Seismic	12.3	34.2	Y.3.6.1.3.4 D
			Accident Level D	High Seismic	32.5	56.5	
			Accident Level D	Side Drop	41.73	56.5	Table Y.3.7-4
	37PTH	Shell Assembly	Normal/Off Normal Levels A/B	TR-8	43.2	52.5	Table Z.3.7-7
			Accident Level C	UL-8	41.3	63.0	Table Z.3.7-8
			Accident Level D	UL-8	55.2	63.0	Table Z.3.7-9
		Basket Assembly	Accident Level C	Seismic	14.7	34.2	Table Z.3.7-2
			Accident Level D	High Seismic	36.4	56.5	Table Z.3.7-2
			Accident Level D	Side Drop	50.92	57.1	Table Z.3.7-4

(1) In Amendment 13, the already licensed 61BT, 32PT, 24PTH and 61BTH are evaluated for high seismic loads (storage in the HSM-HS); hence only the revised stress results due to high seismic load combination are presented here. For the new DSCs 69BTH and 37PTH, the controlling load combinations are presented here.

(2) The 32PT is also evaluated for increased accident pressure of 125 psig. The maximum stress intensity in the shell assembly for the Level D load combination with accident pressure of 125 psig is 45.07 psi. The allowable stress is 58.3 psi (stress ratio=0.77). The stress ratio for the Level D accident side drop load combination is 0.95 (based on maximum stress intensity of 55.86 and allowable stress of 58.6 psi, as shown in Appendix M, Table M.3-7). Hence, the accident pressure of 125 psig load combination remains bounded by the side drop load combination.

(3) The stress evaluations of the HSM-HS and OS200 TC for use with the 61BT, 32PT, 24PTH, 61BTH are bounded by the stress evaluations of these two components when loaded with the 32PTH1 DSC, as described in Amendment 10 Appendix U, Sections U.3.6 and U.3.7. See discussion in Appendix U, pages U.3.1-1 and U.3.1-1a of Amendment 13.

RSI-19

RSI-19 Provide all the created *DATABASE_OPTION output components for ASCII files related to the LS-DYNA structural files listed in Enclosure 9 to TN E-29954.

Staff noted via the LS-DYNA input files, that *DATABASE_OPTION cards were created to view element data, material energies, global data, nodal point data, etc. However, these ASCII files were not included in the NUHOMS® Amendment 13 application. These output files are necessary to view results information.

This information is required to demonstrate compliance with 10 CFR 72.236(l).

Response to RSI-19

The required output data for each corresponding analysis are listed in Enclosure 9 and provided in Enclosure 10 to this submittal.

RSI-20

RSI-20 (A) Provide the fabrication helium leakage rate test for the entire confinement boundary of the new and modified DSCs

Consistent with the guidance in ANSI 14.5-1997, leak testing of the confinement boundary should encompass welds, joints, and surfaces of the confinement boundary including the base material. The staff does not have sufficient data to generically grant an exception of helium leak testing of base material that may be procured, fabricated, and operated under various conditions for multiple types of canisters, although the likelihood of helium leakage through thick, forged base material for any given canister confinement boundary may be low. In addition, there is not sufficient evidence to correlate the minimum flaw sizes that are detectable during other fabrication examinations (e.g. UT) with the minimum flaw sizes in any orientation that may cumulatively result in leak rates greater than 1.0 x 10-7 ref cm3/sec. The applicant should list the operating procedures for helium leak testing of the confinement boundary in the SAR and TS, for helium leakage rate test to the entire confinement boundary of the Standardized NUHOMS® DSCs.

Alternatively, the applicant should provide a basis for demonstrating that the materials, forging, fabrication, and testing of the entire confinement boundary construction provides reasonable assurance that leakage through the canister during its entire service life is not credible, without confirmation by helium leak test. The basis should describe the physical properties of the confinement boundary after fabrication, potential types of flaws (e.g., stringers), and other mechanisms that could potentially result in leakage. In addition, the industry leak test data for canister bodies, or the applicable data for similar types of nuclear components, should be provided to validate the assumed integrity of the base metal and fabrication welds.

RSI-20 (B) Provide the tables to list/summarize the confinement boundary components for all types of DSC designs and display the confinement boundary in the geometry drawing (e.g., dashed line to show the confinement boundary for each DSC design).

This information is needed to determine compliance with 10 CFR 72.236(j) and 72.236(l).

Response to RSI-20 (A)

The Tech Spec has been revised with the addition of Section 4.5, Leakage Testing of the DSC Confinement Boundary. This section specifies leakage testing of the DSC confinement boundary.

Sections Y9.1.3 and Z.9.1.3 of the SAR have been modified accordingly with the following:

The DSC canister confinement boundary is tested using two procedures as described below.

Procedure 1 is accomplished during fabrication:

Upon completion of all canister shell welding and attachment of the inner bottom cover plate to the shell, a temporary seal plate is placed over the open end of the DSC. A bag or other enclosure is placed around the outside of the entire DSC and it is filled with helium. The DSC cavity is evacuated and a helium leakage test is performed using a port in the seal plate. This test is used to show that the entire DSC confinement boundary tested is leak tight (1×10^{-7} ref cm^3/s).

Procedure 2 of the testing occurs after the DSC has been loaded with fuel assemblies:

The DSC cavity has been dried, back filled with helium and the inner top cover plate and the vent and drain port cover plates have been welded in place. After these welds are completed, a temporary test cover is installed or the outer top cover plate is welded in place with at least the root pass of the weld. The cavity between inner cover plate and temporary test cover plate or outer top cover plate is evacuated and a helium leakage test is performed using the test port in the temporary cover plate or outer top cover plate. The leakage test thus includes the weld attaching the inner top cover plate to the canister shell, the vent and drain port cover plate welds and the base metal of the inner top cover plate and port cover plates. The ports are filled with helium prior to welding the vent and drain port cover plates. This test verifies that the tested welds and covers are leak tight (1×10^{-7} ref cm^3/s).

Response to RSI-20 (B):

The confinement boundary of each DSC is shown in Appendix Y, Figure Y.3.1-1 for the 69BTH DSC and in Appendix Z, Figure Z.3.1-1 for the 37PTH DSC. In these figures, the confinement boundaries are shown with solid lines and the redundant sealings are shown with dashed lines.

The DSC components that make up the confinement boundary are listed in the table below with their part numbers as shown in the SAR drawings. The SAR drawings are provided in Y.1.5 and Z.1.5 for the 69BTH and 37PTH DSCs, respectively.

DSC Confinement Boundary Components					
NUHOMS® 69BTH DSC (Y.1.2.1.1)			NUHOMS® 37PTH DSC (Z.1.2.1.1)		
Component	DWG No.	Item No.	Component	DWG No.	Item No.
DSC Shell	NUH69BTH-72-1003	1	DSC Shell	NUH37PTH-72-1003	1
Inner Top Cover Plate	NUH69BTH-72-1001	3	Inner Top Cover Plate	NUH37PTH-72-1001	3
Inner Bottom Cover Plate	NUH69BTH-72-1003	2, or 5 and 8, or 4 and 5, or 9	Inner Bottom Cover Plate	NUH37PTH-72-1003	2, or 5 and 8, or 4 and 5, or 9
Siphon and Vent Block	NUH69BTH-72-1002	3	Siphon and Vent Block	NUH37PTH-72-1002	3
Siphon/Vent Port Cover Plates	NUH69BTH-72-1001	6	Siphon/Vent Port Cover Plates	NUH37PTH-72-1001	6
Associated Welds	All above drawings	---	Associated Welds	All above drawings	---

OBSERVATIONS

1.0 GENERAL DESCRIPTION/CERTIFICATE OF COMPLIANCE

OB 1-1 Explain the apparent inconsistency in allowed decay heat in the HSM as compared to the design decay heat in several of the DSCs.

The proposed CoC describes the HSM as:

> The HSM is a reinforced concrete unit with penetrations located at the top and bottom of the walls for air flow, and is designed to store DSCs with up to 24.0 kW decay heat.

However, there are DSCs in the proposed Amendment 13 with higher design decay heat: 40.8 kW for the 24PTH-S (Table 1-1l), 31.2kW for the 61BTH (Table 1-1t), and 40.8 kW for the 32PTH1-S, 32PTH1-M and the 32PTH1-L (Table 1-1aa).

This information is required to demonstrate compliance with 10 CFR 72.236(a).

Response to OB 1-1

For clarification of allowed heat loads in HSM-H, the following text was proposed for CoC in Amendment 11 after the subject sentence which is under review by NRC.

> "The HSM-H is an enhanced version of the HSM and is designed to store DSCs with up to 40.8 kW decay heat."

This change is marked up with a magenta colored cloud on the proposed CoC change in Enclosure 4 of this submittal.

4.0 THERMAL EVALUATION

OB 4-1 Correct the polynomial functions for the thermal conductivity of helium and air used in thermal calculations. The applicant mistyped the polynomial functions for the thermal conductivity of helium (page Z.4-5) and air (page Z.4-6) for the 37PTH and other DSCs. The polynomial function $k = \sum (Ci\ Ti)$ should be modified to $k = \sum (C_i\ T^i)$ in Chapter 4 for all types of DSCs.

This information is required to demonstrate compliance with 10 CFR 72.236(f)

Response to OB 4-1

The subject polynomial functions are corrected in Appendix Y.4 and Appendix Z.4 for the 69BTH DSC and 37PTH DSC, respectively as requested.

6.0 SHIELDING EVALUATION

6-1 References to the term "in core" should be replaced by "fuel" to refer to the source term from the spent fuel, in order to maintain coherence in the Standardized NUHOMS® system SAR.

This information is required to demonstrate compliance with 10 CFR 72.236(a)

Response to OB 6-1

The term "in-core" is replaced with "active fuel" in Appendix Y.5 for 69BTH DSC and Appendix Z.5 for 37PTH DSC. The Standardized NUHOMS® system UFSAR will be updated in the future for consistency.

8.0 MATERIALS EVALUATION

OB 8-1 Clarify the definition of "Control Components." Provide an example of material that is "positioned or operated within the envelope of the fuel assembly during reactor operation" so that it is also considered a Control Component, but not specifically listed in the proposed contents of the package. It is unclear what might be considered a Control Component other than what has been listed in the proposed contents of the package.

This information is required to demonstrate compliance with 10 CFR 72.236(a).

Response to OB 8-1

The definition of Control Components (CCs) in the NUHOMS® Technical Specifications was intended to include non-fuel hardware such as instrument tube tie rods, guide tube anchors, etc., which are inserted into the fuel assembly after the fuel assembly is discharged from the core. The Technical Specifications currently also include the design basis thermal and radiological characteristics for the CCs.

These components do not contribute to thermal and radiological source terms since they are not irradiated in the reactor core. The thermal and radiological analyses for the un-irradiated components are bounded by the existing safety analyses of fuel assemblies with irradiated CCs. The total weight of the fuel assemblies with these components is bounded by the weight limit included in the Technical Specifications and remains bounded by that employed in the structural analyses. The criticality analysis as documented in the UFSAR (for example, 37PTH DSC in Appendix Z.6, Section Z.6.4.3) conservatively uses BPRAs to model the CCs. The BPRAs are modeled as solid $^{11}B_4C$ in the guide tubes and instrument tubes which bounds all other CCs including non-fuel hardware.

Therefore, these components do not adversely affect the behavior of the fuel assembly and the existing safety analyses of the fuel assembly with CCs remain applicable.

For further clarification, the definition of CCs in the Technical Specifications is revised accordingly. The definition of the CCs in Appendix M, Appendix N, Appendix P, Appendix U and Appendix Z of the UFSAR is also revised for consistency.

OB 8-2 Explain how the top and bottom end caps will be attached and removed from the cells used to contain damaged fuel. Specify an overview of the loading and unloading procedures used to attach and remove the end caps. It is unclear how the end caps are attached and/or removed and if retrievability of the fuel is ensured.

This information is required to demonstrate compliance with 10 CFR 72.236(a).

Response to OB 8-2

For all the baskets with damaged fuel end cap designs, the damaged fuel assemblies are located between the top and bottom end caps. The top and bottom end caps closely fit inside the basket fuel compartment so that their movement is limited to axial sliding within the fuel compartment. After the top shield plug is installed, these end caps are trapped between the DSC top shield plug and the DSC inner bottom plate and cannot slide out of the basket damaged fuel compartment. As an example, relative lengths of the basket compartment, DSC cavity, and the end caps are calculated as follows for the 69BTH DSC.

As shown on drawing NUH69BTH-72-1001 (sheet 2 of 4), the length of the DSC internal cavity is 179.50 in.

As shown on drawing NUH69BTH-72-1011 (sheet 2 of 5), the length of the basket compartment is 164.00 in. For the basket containing damaged fuel, the damaged fuel basket compartment extends another 12 in. (as indicated on drawing NUH69BTH-71-1014, note 2A); therefore the total length of the basket compartment is 176.0 in.

The loading sequence is to first install the bottom end cap (drawing NUH69BTH-72-1015, bottom end cap, side view, 2.0 in. fits inside the bottom of the fuel compartment), then load the fuel, then install the 6.00 in. top end cap (drawing NUH69BTH-72-1015, top end cap, Section C-C; the bottom 3.50 in. fits inside the top of the fuel compartment and the remaining 2.50 in. aligns and stays on top of the basket fuel compartment). Therefore, the total length of the damaged fuel compartment is 178.50 in. (including the protruding top end cap length of 2.50 in.). The loading sequence is provided in Section Y.8.1.2 of Appendix Y.

Based on this, neither top nor bottom damaged fuel end caps need to be attached to the fuel compartment used to contain damaged fuel, since their respective lengths (2.00 in. and 3.50 in.) are greater than 179.50 – 178.50 = 1.00 in, which does not allow them to slide out of the damaged fuel compartments.

The unloading sequence is to first remove the shield plug after opening the DSC. The top end cap can then be removed through the use of a tool that is threaded into the two top end cap tool sockets. The top of the fuel assembly is then accessible and can be retrieved by normal means. The unloading sequence is provided in Section Y.8.2.2 of Appendix Y.

A similar description applies for the 37PTH DSC.

OB 8-3 Provide the setting time to achieve the minimum 5,000 psi compressive strength requirement for the proposed Type III-based concrete and describe how acceptance tests are performed.

Portland Type III cement is an early-setting, high strength cement. Concretes made from Portland Type III cement may have lower strengths over longer time frames than Type II-based-concretes.

This information is required to demonstrate compliance with 10 CFR72.140.

Response to OB 8-3

The required design concrete compressive strength for HSMs is 5000 psi. Regardless of the cement type used (Type II or Type III as classified per ASTM C150) and rate of strength gain, achievement of the minimum 5000 psi compressive strength will be verified. Concrete samples are prepared and tested for strength in accordance with the applicable code of construction (ACI-318) and TN construction specifications. Concrete samples are tested for compressive strength after 28 days of curing and the associated official break test result records are maintained. Therefore, while Type III cement may yield high early strength, achieving the minimum design strength will be ensured and verified prior to certification of the HSM for use at an ISFSI site.

OB 8-4 Define "SST" on the licensing drawings and impose limits on the chemical composition of the stainless steels using a widely accepted standard, for example the Unified Numbering System (UNS) designation.

The staff presumes that the term "SST" is an abbreviation for stainless steel, but the term is not defined on the licensing drawings. Although the components made of "SST" are not safety related, for purposes of corrosion resistance and galvanic reactions, composition limits of "SST" or appropriate reference to ASTM standards for "SST" are requested.

This information is required to demonstrate compliance with 10 CFR 72.140 and 72.146(a).

Response to OB 8-4

Notes have been added to the NUHOHMS® 69BTH and 37PTH licensing drawings (Appendices Y.1.5 and Z.1.5, respectively) to specify a wide range of austenitic stainless steels that are all well defined by the ISO standards or commonly used industrial designations. Components fabricated from these austenitic stainless steels have good corrosion resistance and will not experience significant chemical, galvanic, or other reactions in air, helium, or water environments.

OB 8-5 Justify that galling will not occur between the stainless steel canister and the proposed aluminum basket rails. Clarify if stainless steel basket rails are being proposed. Sliding contact between oxide passivated metals can cause galling. Although galling between aluminum and stainless steel are not as well-documented as stainless/stainless or aluminum/aluminum galling, the potential for galling between dissimilar metals exists.

This information is required to demonstrate compliance with 10 CFR 72.236(d) and 72.236(g).

Response to OB 8-5

The 37PTH and 69BTH DSCs have only aluminum rails. Sliding contact between stainless steel and aluminum surfaces can produce galling only under certain conditions. This often occurs when stainless steel fasteners are threaded into a hole tapped into aluminum. This is because two parameters that promote galling are heat and relative velocity between the two contacting surfaces. The heat is usually due to friction and that is greatly increased by increasing pressure between the two contacting surfaces.

For the 37PTH and 69BTH the aluminum/stainless steel surfaces do not slide relative to one another during normal usage. The only time sliding contact occurs is during insertion of the basket (with rails attached) into the canister during fabrication. This process is done in a controlled manner such that contact is minimized as is the relative velocity between the two surfaces. Contact is minimized because the basket has a smaller diameter than the canister and it is centered as much as possible inside the canister during the insertion. Therefore, the conditions necessary to initiate galling are minimized.

Severe galling would prevent the basket from being fully inserted and thus the canister could not be used without retracting and resolving the galling. TN's experience with canister fabrication reinforces the explanation given above.

In a hypothetical event that galling would occur during the basket insertion, the consequences of galling during the basket insertion process are minimal. Since aluminum is the softer metal, any damage would be limited to the rails. The canister shell (containment boundary) would not be damaged. Surface damage to the rails would not impact their ability to support the basket in a horizontal orientation nor would the heat transfer characteristics of the rails be affected.

OB 8-6 Specify controls and testing to ensure that soluble carbide phases, such as aluminum carbide do not form from the interaction of boron carbide and molten aluminum during processing. Molten aluminum has the potential for reacting with boron carbide to form aluminum carbide, which reacts with water to form flammable acetylene gas. The generation of acetylene could be a significant safety concern during cask loading operations.

This information is required to demonstrate compliance with 10 CFR 72.236(h).

References for OB 8-6:

1) H. Nayeb-Hashemi and D. Shan, "Evaluation of heat damage on B4C particulate reinforced aluminum alloy matrix composite using acoustic emission techniques," v266, p8-17, (1999).

2) J.C. Viala, et al., "Chemical reactivity of aluminum with boron carbide," Journal of Materials Science, v32, p4559-4573, (1997).

3) A.J. Pyzik and D.R. Beaman, "Al-B-C phase development and effects on mechanical properties of B4C/Al-derived composites," Journal of the American Ceramic Society, v78, p305-312, (1995).

4) M. Kouzeli, C.S. Marchi, A. Mortensen, "Effect of reaction on the tensile behavior of infiltrated boron carbide-aluminum composites," Materials Science and Engineering A, v337, p264-273, (2002).

Response to OB 8-6

As part of qualification testing, Transnuclear evaluates the MMCs by immersion in water, and compares the formation of gas bubbles from the MMC to the gas bubbles formed on commercially pure aluminum. MMCs made from a molten aluminum process have not shown any significant amount of gas generation greater than the aluminum.

Transnuclear uses two manufacturing methods that involve molten aluminum, rather than a powder metallurgy process. In both cases, the practical requirements of the process limit the temperature and dwell time for contact between molten aluminum and boron carbide

1) Direct chill or open mold casting: The melt must be stirred to distribute the boron carbide uniformly. Experience has shown that if the melt is held too long before casting, reaction products make the melt too viscous to cast properly, and result in rejected product. The supplier typically adds a small amount of titanium to the aluminum, and limits the time between adding the boron carbide and casting to limit the formation of reaction products.
2) In infiltration casting, a green boron carbide porous billet is set into a molten aluminum bath under inert atmosphere, and the aluminum is drawn into the billet by capillary action. The method requires proprietary treatment of the boron carbide to assure that the molten aluminum will wet it. Any formation of reaction products would block the pores and increase the viscosity of the aluminum, and result in rejected product.

The aluminum is generally held at about 800°C. The formation of Al_4C_3 in significant amounts could only occur from aluminum and B4C at a much higher temperature of about 1200°C. Therefore, the formation of Al_4C_3 in significant amounts does not occur in manufacturing of MMC plates specified by Transnuclear and these practical considerations are adequate to prevent the suggested problem, without imposition of formal controls.

Enclosure 3 to TN E-31217

List of CoC, TS, and UFSAR Page Changes Associated with Amendment 13, Revision 1

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Page number	**Reason for change**
CoC page 2	Observation 1-1

Technical Specification	
Page number	**Reason for change**
4-41 (new page)	RSI-20 (A)
5-6	RSI-20 (A)
5-9	RSI-8
T-5	Observation 8-1
T-10	RSI-4 and Observation 8-1
T-12	RSI-4
T-15	RSI-4
T-16	Observation 8-1
T-26	RSI-4
T-33	Observation 8-1
T-49	Observation 8-1

Safety Analysis Report	
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K.3.7-42b	RSI-18
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M.2-13	Observation 8-1
M.3.7-28d	RSI-18
N.2-2a	Observation 8-1
N.2-9	RSI-4 and Observation 8-1
N.5-1a	Observation 8-1
P.2-2	Observation 8-1
P.2-19	RSI-4
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T.3.7-37	RSI-18
T.3.7-37A	RSI-18
U.2-2	Observation 8-1
U.2-2a	Observation 8-1
U.2-14	Observation 8-1
U.3.1-12a	RSI-18
Y.3-39	RSI-1
Y.3-39a	RSI-1
Y.3-120	RSI-18
Y.3-178	RSI-1
Y.4-6	Observation 4-1
Y.4-14	RSI-9
Y.4-14a	RSI-8 and RSI-9
Y.4-15	RSI-8
Y.4-30	RSI-8 and RSI-9
Y.4-30a	RSI-8 and RSI-9
Y.4-60	RSI-8 and RSI-9
Y.4-61	RSI-8 and RSI-9

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Y.4-63	RSI-8 and RSI-9
Y.4-68	RSI-8 and RSI-9
Y.4-69	RSI-8 and RSI-9
Y.4-70	RSI-8 and RSI-9
Y.5-4	Observation 6-1
Y.5-6	RSI-1
Y.5-7	Observation 6-1
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Y.9-2	RSI-20 (A)
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Z.4-6	Observation 4-1
Z.4-9	RSI-8 and RSI-9
Z.4-9a	RSI-8 and RSI-9
Z.4-10	RSI-8 and RSI-9
Z.4-10a	RSI-8 and RSI-9
Z.4-20	RSI-8 and RSI-9
Z.4-40	RSI-8 and RSI-9
Z.4-41	RSI-9
Z.4-43	RSI-9
Z.5-1	Observation 8-1
Z.5-6	Observation 6-1
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Z.5-14	Observation 6-1
Z.5-61	Observation 6-1
Z.5-63	Observation 6-1
Z.5-64	Observation 6-1
Z.5-65	Observation 6-1
Z.5-66	Observation 6-1
Z.5-67	Observation 6-1
Z.5-68	Observation 6-1
Z.5-69	Observation 6-1
Z.9-2	RSI-20 (A)
Z.9-2a	RSI-20 (A)
Z.11-12	RSI-8

List of CoC, TS, and UFSAR Page Changes Associated with Amendment 13, Revision 1

Safety Analysis Report - Drawings	
Drawing number	**Reason for change**
NUH69BTH-72-1001, Rev. 0B	Observation 8-4
NUH69BTH-72-1002, Rev. 0B	Observation 8-4
NUH69BTH-72-1003, Rev. 0B	Observation 8-4
NUH69BTH-72-1004, Rev. 0B	Observation 8-4
NUH69BTH-72-1011, Rev. 0B	Observation 8-4
NUH69BTH-72-1015, Rev. 0B	Observation 8-4
NUH37PTH-72-1002, Rev. 0B	Observation 8-4
NUH37PTH-72-1004, Rev. 0B	Observation 8-4
NUH37PTH-72-1011, Rev. 0B	Observation 8-4
NUH37PTH-72-1015, Rev. 0B	Observation 8-4

CoC 1004 Amendment 13, Revision 1 Technical Specifications and SAR Page Replacement Instructions

Certificate of Compliance	
Old page	**Revision 1 Replacement Page**
CoC page 2	CoC page 2

Technical Specifications	
Old page	**Revision 1 Replacement Page**
New page	4-41
5-6	5-6
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T-16	T-16
T-26	T-26
T-33	T-33
T-49	T-49

SAR Proprietary Version	
Old page	**Revision 1 Replacement Page**
Cover Page	Cover Page
K.2-13	K.2-13
K.3.7-42b	K.3.7-42b
M.2-2	M.2-2
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T.3.7-37	T.3.7-37
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Enclosure 5 to TN E-31217

CoC Markup for Proposed Amendment 13, Revision 1 Changes

and steel or aluminum

which provide the transition to the DSC shell,

or grid

The principal component subassemblies of the DSC are the shell with integral bottom cover plate, bottom shield plug or shield plug assemblies, ram/grapple ring, top shield plug or shield plug assemblies, top cover plate, and basket assembly. The shell length is fuel-specific. The internal basket assembly for the 24P, 24PHB, and 52B DSCs is composed of guide sleeves, support rods, and spacer disks. This assembly is designed to hold 24 PWR fuel assemblies or 52 BWR assemblies.

An alternate basket assembly configuration, consisting of assemblies of stainless steel fuel compartments ~~held in place by~~ basket rails and a holdown ring, is designed to hold 61 BWR assemblies. The 32PT, and 32PTH1 DSC basket assembly configurations are similar, consisting of welded stainless steel plates or tubes that make up a grid of fuel compartments supported by aluminum basket rails, and are designed to accommodate 32 PWR assemblies. The 24 PTH DSC basket assembly configuration consists of stainless steel tubes supported by basket rails and is designed to accommodate 24 PWR assemblies.

Insert C (attached)

The basket assembly aids in the insertion of the fuel assemblies, enhances subcriticality during loading operations, and provides structural support during a hypothetical drop accident. The DSC is designed to slide from the transfer cask into the HSM and back without undue galling, scratching, gouging, or other damage to the sliding surfaces.

multi-walled

comprised of lead and/or steel gamma shield and neutron shield layers,

The HSM is a reinforced concrete unit with penetrations located at the top and bottom of the walls for air flow, and is designed to store DSCs with up to 24.0 kW decay heat. The penetrations are protected from debris intrusions by wire mesh screens during storage operation. The DSC Support Structure, a structural steel frame with rails, is installed within the HSM. An alternate version of the HSM-H design has been ~~provided to allow the use of the NUHOMS® system in locations where higher~~ seismic levels exist.

The HSM-H is an enhanced version of the HSM and is designed to store DSCs with up to 40.8 kW decay heat.

The TC is designed and fabricated as a lifting device to meet NUREG-0612 and ANSI N14.6 requirements. It is used for transfer operations within the Spent Fuel Pool Building and for transfer operations to/from the HSM. The TC is a cylindrical vessel with a bottom end closure assembly and a bolted top cover plate. Two upper lifting trunnions are located near the top of the cask for downending/uprighting and lifting of the cask in the Spent Fuel Pool Building. The lower trunnions, located near the base of the cask, serve as the axis of rotation during downending/uprighting operations and as supports during transport to/from the Independent Spent Fuel Storage Installation (ISFSI). The 32PT DSC is transferred in a TC with a radial liquid neutron shield.

Insert A (attached)

With the exception of the TC, fuel transfer and auxiliary equipment necessary for ISFSI operations are not included as part of the Standardized NUHOMS® System referenced in this Certificate of Compliance (CoC). Such site-specific equipment may include, but is not limited to, special lifting devices, the transfer trailer, and the skid positioning system

The maximum loaded weight is 110 tons for the Standardized OS197 and OS197L TCs. The maximum loaded weight is 130 tons for the OS197H and OS200 TCs.

c. Drawings

,

The drawings for the Standardized NUHOMS® System are contained in Appendices E, K, M, N, P, T ~~and~~ U of the FSAR.

~~and W~~ , W, Y, and Z

d. Basic Components

,

The basic components of the Standardized NUHOMS® System that are important to safety are the DSC, HSM, and TC. These components are described in Section 4.2, Table K.2-8 (Appendix K), Table M.2-18 (Appendix M), Table P.2-17 (Appendix P), Section T.2.3 (Appendix T) ~~and~~ Section U.2.3 (Appendix U) of the FSAR.

~~and~~ Section W.2.3 (Appendix W)

(including the trailer shielding for the OS197L TC)

, Section Y.2.3 (Appendix Y, and Section Z.2.3 (Appendix Z)

Enclosure 6 to TN E-31217

Changed Pages for the CoC 1004, Amendment 13 Technical Specifications, Revision 1

4.5 LEAKAGE TESTING of the CONFINEMENT BOUNDARY

The DSC shell (including the inner bottom cover plate) base metal and associated confinement boundary welds are tested during fabrication to $1x10^{-7}$ ref cm^3/s. The inner seal welds, inner top cover and port covers are tested upon closure of the loaded DSC as specified in Section 5.2.4 c) of the Technical Specifications.

Remote operations and appropriate ALARA practices shall be used due to very high dose rates during movement of the loaded OS197L TC from fuel pool to the decontamination area and from the decontamination area to the transfer trailer. When remote operations are used, approved written procedures shall be in place to govern these operations. When remote operations are used redundancy of equipment and their quality standards shall be considered and appropriate quality standards for the remote handling equipment shall be assigned.

b) All DSC closure welds except those subjected to full volumetric inspection shall be dye penetrant tested in accordance with the requirements of the ASME Boiler and Pressure Vessel Code Section III, Division 1, Article NB-5000. The liquid penetrant test acceptance standards shall be those described in Subsection NB-5350 of the Code.

This criteria is applicable to all DSCs. The welds include inner and outer top and bottom covers, and vent and siphon port covers.

If the liquid penetrant test indicates that the weld is unacceptable:

1. The weld shall be repaired in accordance with approved ASME procedures, and
2. The new weld shall be re-examined in accordance with this specification.

c) Following completion of the seal weld of the DSC inner top cover plate/top shield plug assembly, (including vent and siphon port cover) this weld shall be leak tested with a helium leak detection device. The leak testing is performed to the criteria as listed below:

DSC Model	**Leak Test Criterion**
24P, 52B	$\leq 1\times10^{-4}$atm.cm^3/sec
61BT, 32PT, 24PHB, 24PTH, 61BTH, 32PTH1, *69BTH, or 37PTH*	$\leq 1\times10^{-7}$*Ref.*cm^3/sec

If the leakage rate of the inner seal weld exceeds the specified criterion, check and repair (a) the inner seal welds (b) the inner top cover and port covers for any surface indications resulting in leakage.

d) Following placement of each loaded TC/DSC into the cask decontamination area but prior to seal weld of the DSC inner top cover plate/top shield plug assembly to DSC shell, the DSC smearable surface contamination levels on the outer top 1 foot surface of the DSC shall be less than 2,200 dpm/100 cm^2 from beta and gamma sources, and less than 220 dpm/100 cm^2 from alpha sources.

5.2.5 HSM or HSM-H Thermal Monitoring Program

This program provides guidance for temperature measurements that are used to monitor the thermal performance of each HSM.

Note: Only one of the two alternate surveillance activities listed below (5.2.5a or 5.2.5b) shall be performed for monitoring the HSM or HSM-H thermal performance.

a) Daily Visual Inspection of the HSM or HSM-H Air Inlets and Outlets (Front Wall and Roof Bird Screens)

A daily visual surveillance shall be conducted of the exterior of the air inlets and outlets to ensure that HSM air vents are not blocked for periods longer than assumed in the safety analysis.

In addition, a visual inspection shall be performed to ensure that no materials accumulate between the modules (only applicable for HSM designs with gap between adjacent modules) that could block the air flow.

If the surveillance shows blockage of air vents *(any blockage of the outlet vents or more than 50% of the inlet vents)*, they shall be cleared. If the bird screen is damaged, it shall be replaced.

b) Daily HSM or HSM-H Temperature Measurement

Verify the thermal performance of each HSM or HSM-H via a direct temperature measurement on a daily basis. The temperature measurement could be any parameter such as (1) a direct measurement of the HSM or HSM-H temperatures, (2) a direct measurement of the DSC temperatures, (3) a comparison of the inlet and outlet temperature difference to predicted temperature differences for each individual HSM or HSM-H, or (4) other means that would identify and allow for the correction of off-normal thermal conditions that could lead to exceeding the concrete and fuel clad temperature criteria. If air temperatures are measured, they must be measured in such a manner as to obtain representative values of inlet and outlet air temperatures. Also, due to the proximity of adjacent HSM or HSM-H modules, care must be exercised to ensure that measured air temperatures reflect only the thermal performance of an individual module, and not the combined performance of adjacent modules.

If the temperature measurement shows a significant unexplained difference, so as to indicate the approach to the concrete material or fuel clad temperature criteria, take appropriate action to determine the cause and return the canister to normal operation. If the measurement or other evidence suggests that the concrete accident temperature criteria (350°F for HSM or the elevated temperature used in Section 5.5 to perform concrete testing for HSM-H) has been exceeded for more than 24 hours, the licensee can provide analysis results and/or test results in accordance with ACI-349, appendix A.4.3, demonstrating that the

Table 1-1e
PWR Fuel Specifications for Fuel to be Stored in the NUHOMS®-32PT DSC

PHYSICAL PARAMETERS:	
Fuel Class	Only intact (including reconstituted) B&W 15x15, WE 17x17, CE 15x15, WE 15x15, CE 14x14 and WE 14x14 class PWR assemblies or equivalent reload fuel manufactured by other vendors that are enveloped by the fuel assembly design characteristics listed in Table 1-1f.
Reconstituted Fuel Assemblies	≤ 32 assemblies per DSC with up to 56 stainless steel rods per assembly or unlimited number of lower enrichment UO2 rods per assembly.
Fuel Cladding Material	Zircaloy
Fuel Damage	Cladding damage in excess of pinhole leaks or hairline cracks is not authorized to be stored as "Intact PWR Fuel."
Control Components (CCs)	• Up to 32 CCs are authorized for storage *in the 32PT DSC.* • Authorized CCs include Burnable Poison Rod Assemblies (BPRAs), Thimble Plug Assemblies (TPAs), Control Rod Assemblies (CRAs), Rod Cluster Control Assemblies (RCCAs), Axial Power Shaping Rod Assembly (APSRAs), Orifice Rod Assemblies (ORAs), Vibration Suppression Inserts (VSIs), Neutron Source Assemblies (NSAs) and Neutron Sources. *Non-fuel hardware that are positioned within the fuel assembly after the fuel assembly is discharged from the core such as Guide Tube or Instrument Tube Tie Rods or Anchors, Guide Tube Inserts, BPRA Spacer Plates or devices that are positioned and operated within the fuel assembly during reactor operation such as those listed above are also considered as CCs.* • Design basis thermal and radiological characteristics for the CCs are listed in Table 1-1ee.
Maximum Assembly plus CC Weight	-1365 lbs for 32PT-S100 & 32PT-L100 System -1682 lbs for 32PT-S125 & 32PT-L125 System
CC Damage	CCs with cladding failures are acceptable for loading.
THERMAL/RADIOLOGICAL PARAMETERS:	
Fuel Burnup and Cooling Time *with or* without CCs[1]	Per Table 1-2d, Table 1-2e, Table 1-2f, Table 1-2g, Table 1-2h, and Figure 1-2 or Figure 1-3 or Figure 1-4, except that for a 32PT DSC contained in an OS197L TC, see Tables 1-6c and 1-6d, and Figure 1-30.
Initial Enrichment	Per Table 1-1g, *1-1g1,* and Figure 1-5 or Figure 1-6 or Figure 1-7, *as applicable.*

[1] BPRAs are considered as being representative of all CCs.

Table 1-1i
PWR Fuel Specifications for Fuel to be Stored in the Standardized NUHOMS®-24PHB DSC

Title or Parameter	***Specifications***
Fuel *Class*	Intact *or damaged,* unconsolidated B&W 15x15 (with or without *CCs), intact* WE 17x17, *intact* WE 15x15, *intact* CE 14x14 *and intact* WE 14x14 Class PWR fuel assemblies (all without *CCs)* or equivalent reload fuel manufactured by other vendor, with the following requirements: *Damaged fuel assemblies beyond the definition contained below are not authorized for storage.*
Maximum No. of Reconstituted Assemblies per DSC with Stainless Steel Rods	4
Maximum No. of Stainless Steel Rods per Reconstituted Assembly	10
Maximum No. of Reconstituted Assemblies per DSC with Low Enriched Uranium Oxide Rods	24
Fuel Damage	*Damaged PWR fuel assemblies are assemblies containing missing or partial fuel rods or fuel rods with known or suspected cladding defects greater than hairline cracks or pinhole leaks. The extent of damage in the fuel assembly is to be limited such that a fuel assembly is being able to be handled by normal means. Missing fuel rods are allowed. Damaged fuel assemblies shall also contain top and bottom end fittings or nozzles or tie plates depending on the fuel type.*
Control Components	• *Up to 24 CCs are authorized for storage in 24PHBL DSCs only.* • *Authorized CCs include Burnable Poison Rod Assemblies (BPRAs), Thimble Plug Assemblies (TPAs), Control Rod Assemblies (CRAs), Rod Cluster Control Assemblies (RCCAs), Axial Power Shaping Rod Assemblies (APSRAs), Orifice Rod Assemblies (ORAs), Vibration Suppression Inserts (VSIs), Neutron Source Assemblies (NSAs), and Neutron Sources. Non-fuel hardware that are positioned within the fuel assembly after the fuel assembly is discharged from the core such as Guide Tube or Instrument Tube Tie Rods or Anchors, Guide Tube Inserts, BPRA Spacer Plates or devices that are positioned and operated within the fuel assembly during reactor operation such as those listed above are also considered as CCs.* • *Design basis thermal and radiological characteristics for the CCs are listed in Table 1-1n(1).*
Physical Parameters (without *CCs)*	
Maximum Assembly Length (unirradiated, *intact assembly with Maximum Burnup ≤ 55 GWd/MTU)*	*165.785 in* (*Standard Cavity)* *171.23 in* (Long Cavity)
Maximum Assembly Length (unirradiated, damaged assembly with Maximum Burnup ≤ 45 GWd/MTU)	*165.785 in (Standard Cavity)* *171.23 in (Long Cavity)*

Table 1-1j
BWR Fuel Specification of Damaged Fuel to be Stored in the Standardized NUHOMS®-61BT DSC

PHYSICAL PARAMETERS:	
Fuel Design:	7x7, 8x8 BWR damaged fuel assemblies manufactured by General Electric or Exxon/ANF or equivalent reload fuel that are enveloped by the Fuel assembly design characteristics listed in Table 1-1d for the 7x7 and 8x8 designs only. *Damaged fuel assemblies beyond the definition contained below are not authorized for storage.*
Cladding Material:	Zircaloy
Fuel Damage:	Damaged BWR fuel assemblies are fuel assemblies containing fuel rods with known or suspected cladding defects greater than hairline cracks or pinhole leaks. Missing cladding and/or crack size in the fuel pins is to be limited such that a fuel pellet is not able to pass through the gap created by the cladding opening during handling and retrievability is assured following Normal/Off-Normal conditions. Damaged fuel shall be stored with Top and Bottom Caps for Failed Fuel. Damaged fuel may only be stored in the 2x2 compartments of the "Type C" NUHOMS®-61BT Canister.
Channels:	Fuel may be stored with or without fuel channels.
Maximum Assembly Length (unirradiated)	176.2 in
Nominal Assembly Width (excluding channels)	5.44 in
Maximum Assembly Weight	705 lbs
RADIOLOGICAL PARAMETERS:[2]	No interpolation of Radiological Parameters is permitted between groups.
Group 1: Maximum Burnup:	27,000 MWd/MTU
Minimum Cooling Time:	5-years[1]
Maximum Initial Lattice Average Enrichment:	4.0 wt. % U-235
Maximum Pellet Enrichment:	4.4 wt. % U-235
Minimum Initial Bundle Average Enrichment:	2.0 wt. % U-235
Maximum Initial Uranium Content:	198 kg/assembly
Maximum Decay Heat:	300 W/assembly
Group 2: Maximum Burnup:	35,000 MWd/MTU
Minimum Cooling Time:	8-years[1]
Maximum Initial Lattice Average Enrichment:	4.0 wt. % U-235
Maximum Pellet Enrichment:	4.4 wt. % U-235
Minimum Initial Bundle Average Enrichment:	2.65 wt. % U-235
Maximum Initial Uranium Content:	198 kg/assembly
Maximum Decay Heat:	300 W/assembly
Group 3: Maximum Burnup:	37,200 MWd/MTU
Minimum Cooling Time:	6.5-years[1]
Maximum Initial Lattice Average Enrichment:	4.0 wt. % U-235
Maximum Pellet Enrichment:	4.4 wt. % U-235
Minimum Initial Bundle Average Enrichment:	3.38 wt. % U-235
Maximum Initial Uranium Content:	198 kg/assembly
Maximum Decay Heat:	300 W/assembly

Table 1-1l
PWR Fuel Specification for the Fuel to be Stored in the NUHOMS®-24PTH DSC

(Part 1 of 3)

PHYSICAL PARAMETERS:	
Fuel Class	Intact or damaged *or failed* unconsolidated B&W 15x15, WE 17x17, CE 15x15, WE 15x15, CE 14x14 and WE 14x14 class PWR assemblies (with or without control components) that are enveloped by the fuel assembly design characteristics listed in Table 1-1m. Equivalent reload fuel manufactured by other vendors but enveloped by the design characteristics listed in Table 1-1m is also acceptable. *Damaged and/or failed fuel assemblies beyond the definition contained below are not authorized for storage.*
Fuel Damage	Damaged PWR fuel assemblies are assemblies containing missing or partial fuel rods or fuel rods with known or suspected cladding defects greater than hairline cracks or pinhole leaks. The extent of cladding damage in the fuel rods is to be limited such that a fuel *assembly needs to be handled by normal means.*
Failed Fuel	*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 and it is located at a position of at least 10" above the top of the bottom shield plug of the DSC.* *Fuel debris may be associated with any type of UO_2 fuel provided that the maximum uranium content and initial enrichment limits are met. The total weight of each failed fuel can plus all its contents shall be less than 1682 lb.*
Partial Length Shield Assemblies (PLSAs)	WE 15x15 class PLSAs which have only ever been irradiated in peripheral core locations with following characteristics are authorized: • Maximum burnup, 40 GWd/MTU • Minimum cooling time, 6.5 years • Maximum decay heat, 900 watts

Table 1-1l
PWR Fuel Specification for the Fuel to be Stored in the NUHOMS®-24PTH DSC
(Part 2 of 3)

Reconstituted Fuel Assemblies: • Maximum No. of Reconstituted Assemblies per DSC with Irradiated Stainless Steel Rods • Maximum No. of Irradiated Stainless Steel Rods per Reconstituted Fuel Assembly • Maximum No. of Reconstituted Assemblies per DSC with unlimited number of low enriched UO_2 rods and/or Unirradiated Stainless Steel Rods and/or Zr Rods or Zr Pellets	4 10 24
Control Components (CCs)	• Up to 24 CCs are authorized for storage in 24PTH-L, 24PTH-S, and 24PTH-S-LC DSCs only. • Authorized CCs include Burnable Poison Rod Assemblies (BPRAs), Thimble Plug Assemblies (TPAs), Control Rod Assemblies (CRAs), Rod Cluster Control Assemblies (RCCAs), Axial Power Shaping Assembly Rods (APSRAs), Orifice Rod Assemblies (ORAs), Vibration Suppression Inserts (VSIs), Neutron Source Assemblies (NSAs), and Neutron Sources. *Non-fuel hardware that are positioned within the fuel assembly after the fuel assembly is discharged from the core such as Guide Tube or Instrument Tube Tie Rods or Anchors, Guide Tube Inserts, BPRA Spacer Plates or devices that are positioned and operated within the fuel assembly during reactor operation such as those listed above are also considered as CCs.* • Design basis thermal and radiological characteristics for the CCs are listed in Table 1-1n.
Nominal Assembly Width *for Intact and Damaged Fuel Only*	8.536 inches
No. of Intact Assemblies	≤24
No. and Location of Damaged Assemblies	Maximum of 12 damaged fuel assemblies. Balance may be intact fuel assemblies, empty slots, or dummy assemblies depending on the specific heat load zoning configuration. Damaged fuel assemblies are to be placed in Location A and/or B as shown in Figure 1-16. The DSC basket cells which store damaged fuel assemblies are provided with top and bottom end caps.
No. and Location of Failed Assemblies	*Up to 8 failed fuel assemblies. Balance may be intact and/or damaged fuel assemblies, empty slots, or dummy assemblies depending on the specific heat load zoning configuration.* *Failed fuel assemblies are to be placed in Location A as shown in Figure 1-16. Failed fuel assembly/fuel debris is to be encapsulated in an individual failed fuel can (FFC) provided with a welded bottom closure and a removable top closure.*
Maximum Assembly plus CC Weight	1682 lbs

Table 1-1t
BWR Fuel Specification for the Fuel to be Stored in the NUHOMS®-61BTH DSC

(Part 1 of 3)

PHYSICAL PARAMETERS:	
Fuel Class	Intact or damaged *or failed* 7x7, 8x8, 9x9 or 10x10 BWR assemblies manufactured by General Electric or Exxon/ANF or FANP or reload fuel manufactured by other vendors that are enveloped by the fuel assembly design characteristics listed in Table 1-1u. Damaged *and/or failed* fuel assemblies beyond the definition contained below are not authorized for storage.
Fuel Damage	Damaged BWR fuel assemblies are assemblies containing fuel rods with known or suspected cladding defects greater than hairline cracks or pinhole leaks. The extent of *cladding* damage in the fuel assembly is to be limited such that *a* fuel assembly *needs* to be handled by normal means. Missing fuel rods are allowed.
Failed Fuel	*Failed fuel is defined as ruptured fuel rods, severed fuel rods, loose fuel pellets, or fuel assemblies that cannot be handled by normal means. Failed 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 and it is located at a position of at least 10" above the top of the bottom shield plug of the DSC.* *Fuel debris may be associated with any type of UO_2 fuel provided that the maximum uranium content and initial enrichment limits are met. The total weight of each failed fuel can plus all its content shall be less than 705 lb.*
RECONSTITUTED FUEL ASSEMBLIES: • Maximum No. of Reconstituted Assemblies per DSC with Irradiated Stainless Steel Rods • Maximum No. of Irradiated Stainless Steel Rods per Reconstituted Fuel Assembly • Maximum No. of Reconstituted Assemblies per DSC with unlimited number of low enriched UO_2 rods or Zr Rods or Zr Pellets or Unirradiated Stainless Steel Rods	 4 10 61
No. of Intact Assemblies	≤ 61

Table 1-1aa
PWR Fuel Specification for the Fuel to be Stored in the NUHOMS®-32PTH1 DSC

PHYSICAL PARAMETERS:	
Fuel Class	Intact or damaged unconsolidated B&W 15x15, WE 17x17, CE 15x15, WE 15x15, CE 14x14, WE 14x14 and CE 16x16 class PWR assemblies (with or without control components) that are enveloped by the fuel assembly design characteristics listed in Table 1-1bb. Reload fuel manufactured by other vendors but enveloped by the design characteristics listed in Table 1-1bb is also acceptable. Damaged fuel assemblies beyond the definition contained below are not authorized for storage.
Fuel Damage	Damaged PWR fuel assemblies are assemblies containing missing or partial fuel rods or fuel rods with known or suspected cladding defects greater than hairline cracks or pinhole leaks. The extent of damage in the fuel assembly is to be limited such that the fuel assembly will still be able to be handled by normal means. *Missing fuel rods are allowed. Damaged fuel assemblies shall also contain top and bottom end fittings or nozzles or tie plates depending on the fuel type.*
Reconstituted Fuel Assemblies:	
• Maximum No. of Reconstituted Assemblies per DSC With Irradiated Stainless Steel Rods	4
• Maximum No. of Irradiated Stainless Steel Rods per Reconstituted Fuel Assembly	10
• Maximum No. of Reconstituted Assemblies per DSC with unlimited number of low enriched UO_2 rods, or Zr Rods or Zr Pellets or Unirradiated Stainless Steel Rods	32
Control Components (CCs)	• Up to 32 CCs are authorized for storage in 32PTH1-S, 32PTH1-M and 32PTH1-L DSCs . • Authorized CCs include Burnable Poison Rod Assemblies (BPRAs), Thimble Plug Assemblies (TPAs), Control Rod Assemblies ((CRAs), Rod Cluster Control Assemblies (RCCAs), Axial Power Shaping Rod Assemblies (APSRAs), Orifice Rod Assemblies (ORAs), Vibration Suppression Inserts (VSIs), Neutron Source Assemblies (NSAs) and Neutron Sources. *Non-fuel hardware that are positioned within the fuel assembly after the fuel assembly is discharged from the core such as Guide Tube or Instrument Tube Tie Rods or Anchors, Guide Tube Inserts, BPRA Spacer Plates or devices that are positioned and operated within the fuel assembly during reactor operation such as those listed above are also considered as CCs.* • Design basis thermal and radiological characteristics for the CCs are listed in Table 1-1ee.
No. of Intact Assemblies	≤ 32
No. and Location of Damaged Assemblies	Up to 16 damaged fuel assemblies with balance intact fuel assemblies, or dummy assemblies are authorized for storage in 32PTH1 DSC. Damaged fuel assemblies are to be placed in the center 16 locations as shown in Figures 1-26 through 1-28. The DSC basket cells which store damaged fuel assemblies are provided with top and bottom end caps.
Maximum Assembly plus CC Weight	1715 lbs

Table 1-1ll
PWR Fuel Specification for the Fuel to be Stored in the NUHOMS®-37PTH DSC

(Part 1 of 2)

PHYSICAL PARAMETERS: *Fuel Class*	*Intact or damaged unconsolidated WE 17x17, CE 16X16, CE 15x15, WE 15x15, CE 14x14, and WE 14x14 class PWR assemblies (with or without control components) that are enveloped by the fuel assembly design characteristics listed in Table 1-1nn. Reload fuel manufactured by other vendors but enveloped by the design characteristics listed in Table 1-1nn is also acceptable. Damaged fuel assemblies beyond the definition contained below are not authorized for storage.*
Fuel Damage	*Damaged PWR fuel assemblies are assemblies containing missing or partial fuel rods or fuel rods with known or suspected cladding defects greater than hairline cracks or pinhole leaks. The extent of damage in the fuel assembly is to be limited such that a fuel assembly is being able to be handled by normal means. Missing fuel rods are allowed.* *Damaged fuel assemblies shall also contain top and bottom end fittings or nozzles or tie plates depending on the fuel type.*
Reconstituted Fuel Assemblies: • *Maximum No. of Reconstituted Assemblies per DSC with Irradiated Stainless Steel Rods* • *Maximum No. of Irradiated Stainless Steel Rods per Reconstituted Fuel Assembly* • *Maximum No. of Reconstituted Assemblies per DSC with Unlimited Number of Low Enriched UO_2 Rods, or Zr Rods or Zr Pellets or Unirradiated Stainless Steel Rods*	 *4* *10* *37*
Control Components (CCs)	• *Up to 37 CCs are authorized for storage in 37PTH-S, and 37PTH-M DSCs.* • *Authorized CCs include Burnable Poison Rod Assemblies (BPRAs), Thimble Plug Assemblies (TPAs), Control Rod Assemblies (CRAs), Rod Cluster Control Assemblies (RCCAs), Axial Power Shaping Rod Assemblies (APSRAs), Orifice Rod Assemblies (ORAs), Neutron Source Assemblies (NSAs), Vibration Suppression Inserts (VSIs) and Neutron Sources. Non-fuel hardware that are positioned within the fuel assembly after the fuel assembly is discharged from the core such as Guide Tube or Instrument Tube Tie Rods or Anchors, Guide Tube Inserts, BPRA Spacer Plates or devices that are positioned and operated within the fuel assembly during reactor operation such as those listed above are also considered as CCs.* • *Design basis thermal and radiological characteristics for the CCs are listed in Table 1-1qq.*

Enclosure 8 to TN E-31217

Changed Pages for the CoC 1004, Amendment 13 Safety Analysis Report, Revision 1, Non-proprietary Version

NON-PROPRIETARY

AREVA

TRANSNUCLEAR, INC.

CoC 1004

Amendment 13

Standardized NUHOMS® System

SAFETY ANALYSIS REPORT

Revision 1

July 2011

7135 Minstrel Way, Suite 300 • Columbia, MD 21045

Table K.2-2
Damaged BWR Fuel Assemblies Characteristics

PHYSICAL PARAMETERS: Fuel Design:	7x7, 8x8 BWR damaged fuel assemblies manufactured by General Electric or Exxon/ANF or equivalent reload fuel that are enveloped by the Fuel assembly design characteristics listed in Table K.2-3 for the 7x7 and 8x8 designs only. *Damaged fuel assemblies beyond the definition contained below are not authorized for storage.*
Cladding Material:	Zircaloy
Fuel Damage:	Damaged BWR fuel assemblies are fuel assemblies containing fuel rods with known or suspected cladding defects greater than hairline cracks or pinhole leaks. Missing cladding and/or crack size in the fuel pins is to be limited such that a fuel pellet is not able to pass through the gap created by the cladding opening during handling and retrievability is assured following Normal/Off-Normal conditions. Damaged fuel shall be stored with Top and Bottom Caps for Failed Fuel. Damaged fuel may only be stored in the 2x2 compartments of the "Type C" NUHOMS®-61BT Canister.
Channels:	Fuel may be stored with or without fuel channels.
Maximum Assembly Length (unirradiated)	176.2 in
Nominal Assembly Width (excluding channels)	5.44 in
Maximum Assembly Weight	705 lbs
RADIOLOGICAL PARAMETERS(1):	
Group 1: Maximum Burnup:	27,000 MWd/MTU
Minimum Cooling Time:	5-years*(2)*
Maximum Initial Lattice Average Enrichment:	4.0 wt. % U-235
Maximum Pellet Enrichment:	4.4 wt. % U-235
Minimum Initial Bundle Average Enrichment:	2.0 wt. % U-235
Maximum Initial Uranium Content:	198 kg/assembly
Maximum Decay Heat:	300 W/assembly
Group 2: Maximum Burnup:	35,000 MWd/MTU
Minimum Cooling Time:	8-years*(2)*
Maximum Initial Lattice Average Enrichment:	4.0 wt. % U-235
Maximum Pellet Enrichment:	4.4 wt. % U-235
Minimum Initial Bundle Average Enrichment:	2.65 wt. % U-235
Maximum Initial Uranium Content:	198 kg/assembly
Maximum Decay Heat:	300 W/assembly
Group 3: Maximum Burnup:	37,200 MWd/MTU
Minimum Cooling Time:	6.5-years*(2)*
Maximum Initial Lattice Average Enrichment:	4.0 wt. % U-235
Maximum Pellet Enrichment:	4.4 wt. % U-235
Minimum Initial Bundle Average Enrichment:	3.38 wt. % U-235
Maximum Initial Uranium Content:	198 kg/assembly
Maximum Decay Heat:	300 W/assembly

(1) Fuel assemblies fully complying with any of the four groups of parameters or alternate radiological parameters are suitable for storage in the NUHOMS®-61BT DSC. No interpolation of Radiological Parameters is permitted between groups 1 to 4 *when the OS197L TC is employed, apply the requirements of Table W.2-4 and Table W.2-5 and Figure W.2-1.*

(2) For fuel assemblies containing BLEU fuel pellets, add 3 years additional cooling time to the minimum values shown in this table.

Table K.3.7-16
61BT Basket, Enveloping Stress Results – High Seismic Loading

Load Cases	Storage Loading	Stress Level	Component	Material	Max.Stress Intensity (ksi)		Allowable Stress Intensity (ksi)	
					P_m (ksi)	P_m+P_b (ksi)	P_m (ksi)	P_m+P_b (ksi)
61BT	1.6g Axial +2g Transverse +3.646g Vertical	D	Fuel Compartment (750 °F)	SA-240 Type 304	16.0	22.9	44.3	57.0
			Transition Rail (550 °F)	SA-240 Type 304	7.5	23.7	44.4	57.1
			Canister Shell (500 °F)	SA-240 Type 304	2.0	32.1	44.4	57.1

M.2.1 Spent Fuel To Be Stored

There are four design configurations for the NUHOMS®-32PT DSC, two "short" canister configurations (the 32PT-S100 and 32PT-S125), and two "long" canister configurations (the 32PT-L100 and 32PT-L125). The main difference between the –S100/-L100 and –S125/-L125 configuration designs *is* the thicknesses of shield plugs and DSC cover plates. The basket layout for these two configurations is identical except for the length of the components. Each of the DSC configurations is designed to store 32 intact *(including reconstituted)* standard PWR *high burnup* fuel assemblies. The 32PT-L100 and 32PT-L125 are also designed to store 32 intact standard PWR fuel assemblies with or without *Control Components (*CCs*)*. The NUHOMS®-32PT DSCs can store intact PWR fuel assemblies and CCs with the characteristics described in Table M.2-1. *The fuel to be stored is limited to a maximum assembly average enrichment of 5.0 wt. %.* ^{235}U*. The maximum allowable assembly average burn up is limited to 55 GWd/MTU and the minimum cooling time is 5.0 years.* The CCs include *b*urnable *p*oison *r*od *a*ssemblies (BPRAs), *t*himble *p*lug *a*ssemblies (TPAs), *c*ontrol rod *a*ssemblies (CRAs), *r*od *c*luster *c*ontrol *a*ssemblies (RCCAs), *a*xial *p*ower *s*haping *r*od *a*ssemblies (APSRAs), *o*rifice *r*od *a*ssemblies (ORAs), *v*ibration *s*uppression *i*nserts (VSIs), *n*eutron *s*ource *a*ssemblies (NSAs), and *n*eutron sources. *Furthermore, non-fuel hardware that are positioned within the fuel assembly after the fuel assembly is discharged from the core such as Guide Tube or Instrument Tube Tie Rods or Anchors, Guide Tube Inserts, BPRA Spacer Plates or devices that are positioned and operated within the fuel assembly during reactor operation such as those listed above are also considered as CCs.* The NUHOMS®-32PT DSC may store PWR fuel assemblies arranged in any of three alternate heat zoning configurations with a maximum decay heat of 1.2 kW per assembly and a maximum heat load of 24 kW per canister. The heat load zoning configurations are shown in Figure M.2-1 through Figure M.2-3. The NUHOMS®-32PT DSC is inerted and backfilled with helium at the time of loading. The maximum fuel assembly weight with a CC is 1682 lb which is the same as the NUHOMS®-24P DSC design.

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 of the NUHOMS®-32PT DSC per *the guidance provided in NUREG-1536*, Revision *1* [2.1]. In addition, *NUREG-1536* does 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 maximum fuel cladding temperature limit of 570 °C (1058 °F) is applicable to accidents or off-normal thermal transients [2.1].

Calculations were performed to determine the fuel assembly type which was most limiting for each of the analyses including shielding, criticality, heat load and confinement. These evaluations are performed in *Appendices* M.5, M.6, M.4 and M.7. The fuel assembly types considered are listed in Table M.2-2. It was determined that the B&W 15x15 is the enveloping fuel design for the shielding source term calculation because of its total assembly weight and highest initial heavy metal loading. For criticality safety, the B&W 15x15 assembly is the most reactive assembly type for a given enrichment. This assembly is used to determine the most reactive configuration in the DSC. Using this most reactive configuration, criticality analysis for all other fuel assembly classes is performed to determine the maximum enrichment allowed as a function of number of *p*oison *r*od *a*ssemblies (PRAs). For thermal analysis, the WE 14x14 fuel assembly is limiting, since it results in the lowest fuel conductivity. The confinement analys*is* is

Table M.2-1
Intact PWR Fuel Assembly Characteristics

PHYSICAL PARAMETERS:	
Fuel Class	Only intact (including reconstituted) B&W 15x15, WE 17x17, CE 15x15, WE 15x15, CE 14x14 and WE 14x14 class PWR assemblies or equivalent reload fuel manufactured by other vendors that are enveloped by the fuel assembly design characteristics listed in Table M.2-2.
Reconstituted Fuel Assemblies	≤ 32 assemblies per DSC with up to 56 stainless steel rods per assembly or unlimited number of lower enrichment UO_2 rods per assembly.
Fuel Cladding Material	Zircaloy
Fuel Damage	Cladding damage in excess of pinhole leaks or hairline cracks is not authorized to be stored as "Intact PWR Fuel."
Control Components (CCs)	• Up to 32 CCs are authorized for storage in 32PT DSC. • Authorized CCs include Burnable poison Rod Assemblies (BPRAs), Thimble Plug Assemblies, (TPAs), Control Rod Assemblies (CRAs), Rod Cluster Control Assemblies (RCCAs), Axial Power Shaping Rod Assemblies (APSRAs), Orifice Rod Assemblies (ORAs), Vibration Suppression Inserts (VSIs), Neutron Source Assemblies (NSAs), and Neutron Sources. *Non-fuel hardware that are positioned within the fuel assembly after the fuel assembly is discharged from the core such as Guide Tube or Instrument Tube Tie Rods or Anchors, Guide Tube Inserts, BPRA Spacer Plates or devices that are positioned and operated within the fuel assembly during reactor operation such as those listed above are also considered as CCs.* • Design basis thermal and radiological characteristics for the CCs are listed in Table M.2-2a.
Maximum Assembly plus CC Weight	-1365 lbs for 32PT-S100 & 32PT-L100 DSC System -1682 lbs for 32PT-S125 & 32PT-L125 DSC System
CC Damage	CCs with cladding failures are acceptable for loading.
THERMAL/RADIOLOGICAL PARAMETERS:	
Fuel Burnup and Cooling Time *with or* without CCs[(1)]	Per Table M.2-5, Table M.2-6, Table M.2-7, Table M.2-8, Table M.2-9; and Figure M.2-1 or Figure M.2-2 or Figure M.2-3, *except that for a 32PT DSC contained in an OS197L TC, see Appendix W, Tables W.2-7, W.2-8, and Figure W.2-2.*
Initial Enrichment	Per Table*s* M.2-3, *M.2-3a,* and Figure M.2-4 or Figure M.2-5 or Figure M.2-6, as applicable.

In Amd. 11, under NRC review

(1) BPRAs are considered as being representative of all CCs.

Table M.3.7-14
NUHOMS®-32PT DSC Components Stress Results for Deadweight + Internal Pressure(1) + High Seismic Load Combination [Bottom End]
32PT DSC Components Stress Results (Bottom)

Component	*Stress Category*	*Stress Intensity (ksi)*	*Allowable Stress (ksi)*	*Stress Ratio*
Shell	P_m	35.31	45.50	0.78
	P_L+P_b	48.90	58.60	0.83
Outer Cover	P_m	22.94	45.50	0.50
	P_L+P_b	26.32	58.60	0.45
Inner Cover	P_m	15.15	45.50	0.33
	P_L+P_b	19.69	58.60	0.34
Shield Plug	P_m	5.37	40.60	0.13
	P_L+P_b	10.04	52.20	0.19
Grapple Ring Support	P_m	1.56	45.50	0.03
	P_L+P_b	3.69	58.60	0.06
Grapple Ring	P_m	0.76	45.50	0.02
	P_L+P_b	1.10	58.60	0.02

Note: *1. Internal Pressure of 15 psig is used for high seismic load combination.*

Orifice Rod Assemblies (ORAs), Vibration Suppression Inserts (VSIs), Neutron Source Assemblies (NSAs) and Neutron Sources. Non-fuel hardware that are positioned within the fuel assembly after the fuel assembly is discharged from the core such as Guide Tube or Instrument Tube Tie Rods or Anchors, Guide Tube Inserts, BPRA Spacer Plates or devices that are positioned and operated within the fuel assembly during reactor operation such as those listed above are also considered as CCs.

- *Intact* WE 17x17, WE 15x15, CE 14x14 and WE 14x14 fuel assemblies, all with*out CC*s.

The NUHOMS®-24PHB DSC is also authorized to store fuel assemblies containing Blended Low Enriched Uranium (BLEU) fuel material. Fuel pellets containing BLEU fuel material are no different than UO_2 fuel pellets except for the presence of a higher quantity of cobalt impurity. The consideration of cobalt impurity only affects the gamma source terms for fuel assemblies located in the DSC periphery. This does not affect any criticality, thermal or structural analysis inputs for evaluation of fuel assemblies with BLEU material. The qualification of fuel assemblies containing BLEU fuel pellets will require an additional cooling time of three years to ensure that the source terms calculated with UO_2 material are bounding.

Fuel assemblies that contain fixed integral non-fuel rods are also considered as intact fuel assemblies. These fuel assemblies are different than reconstituted assemblies because fuel rods are not "replaced" by non-fuel rods, rather the non-fuel rods are part of the initial fuel design. The non-fuel rods displace the same amount of moderator, with zirconium-alloy (or aluminum) cladding and typically contain burnable absorber (or other non-fuel) material. The radiatnion and thermal source terms for the non-fuel rods are significantly lower than those of the fuel rods since there is no significant radioactive decay source. The internal pressure of the non-fuel rods after irradiation is lower than those of the fuel rods since there is no fission gas generation. The reactivity of the fuel rods (from a criticality standpoint) is significantly higher than that of non-fuel rods. In summary, the mechanical, thermal, shielding and criticality evaluations for these rods are bounded by those of the regular rods. Therefore, no further evaluations are required for the qualification of these fuel assemblies.

Calculations are performed to determine the fuel assembly type which is most limiting for each of the analyses including shielding, criticality, heat load and confinement. Analyses performed demonstrated that limiting features associated with the design basis B&W 15x15 fuel assembly are bounding for assembly types listed in Table N.2-1.

N.2.1.1 General Operating Functions

No change *to Chapter 3, Section 3.1.2.*

Table N.2-1
PWR Fuel Specifications for Fuel to be Stored in the NUHOMS®-24PHB DSC

Title or Parameter	Specifications
Fuel *Class*	*Intact or damaged,* unconsolidated B&W 15x15 (with or without *CCs*), *intact* WE 17x17, *intact* WE 15x15, *intact* CE 14x14 and *intact* WE 14x14 Class PWR fuel assemblies (all without *CCs*) or equivalent reload fuel manufactured by other vendor, with the following requirements:
Maximum No. of Reconstituted Assemblies per DSC with Stainless Steel Rods	4
Maximum No. of Stainless Steel Rods per Reconstituted Assembly	10
Maximum No. of Reconstituted Assemblies per DSC with Low Enriched Uranium Oxide Rods	24 *Damaged fuel assemblies beyond the definition contained below are not authorized for storage.*
Fuel Damage	*Damaged PWR fuel assemblies are assemblies containing missing or partial fuel rods or fuel rods with known or suspected cladding defects greater than hairline cracks or pinhole leaks. The extent of damage in the fuel assembly is to be limited such that a fuel assembly is being able to be handled by normal means. Missing fuel rods are allowed. Damaged fuel assemblies shall also contain top and bottom end fittings or nozzles or tie plates depending on the fuel type.*
Control Components	• *Up to 24 CCs are authorized for storage in 24PHBL DSCs only.* • *Authorized CCs include Burnable Poison Rod Assemblies (BPRAs), Thimble Plug Assemblies (TPAs), Control Rod Assemblies (CRAs), Rod Cluster Control Assemblies (RCCAs), Axial Power Shaping Rod Assemblies (APSRAs), Orifice Rod Assemblies (ORAs), Vibration Suppression Inserts (VSIs), Neutron Source Assemblies (NSAs), and Neutron Sources. Non-fuel hardware that are positioned within the fuel assembly after the fuel assembly is discharged from the core such as Guide Tube or Instrument Tube Tie Rods or Anchors, Guide Tube Inserts, BPRA Spacer Plates or devices that are positioned and operated within the fuel assembly during reactor operation such as those listed above are also considered as CCs.* • *Design basis thermal and radiological characteristics for the CCs are listed in Table N.2-2a.*
Physical Parameters (without *CCs*)	
Maximum Assembly Length (unirradiated, *intact assembly with Maximum Burnup ≤ 55 GWd/MTU)*	165.785 in (24PHBS DSC) 171.*23* in (24PHBL DSC)
Maximum Assembly Length (unirradiated, damaged assembly with Maximum Burnup ≤ 45 GWd/MTU)	*165.785 in (24PHBS DSC)* *171.23 in (24PHBL DSC)*
Maximum Assembly Length (unirradiated, damaged assembly with Maximum Burnup> 45 GWd/MTU and ≤ 55 GWd/MTU)	*164.785 in (24PHBS DSC)* *170.23 in (24PHBL DSC)*

- WE 17x17, WE 15x5, CE 14x14 and WE 14x14 class PWR fuel assemblies, all with no BPRAs.

Note that while the B&W fuel types are specifically listed, storing reload fuel designed by other manufacturers is also allowed provided an analysis is performed to demonstrate that the limiting features listed in Table N.2-1 bound the specific manufacturers replacement fuel.

The NUHOMS®-24PHB DSC is also authorized to store fuel assemblies containing Blended Low Enriched Uranium (BLEU) fuel material.

Proprietary information withheld pursuant to 10 CFR 2.390

The design basis fuel source terms for this evaluation bound the source terms from fuel with the burnup/initial enrichment/cooling time combination given in Table N.2-3 through Table N.2-5 (with or without BPRAs and with or without reconstituted fuel assemblies) and located in the basket as shown in Figure N.2-1 and Figure N.2-2. This evaluation bounds the maximum dose rate on the surface of the HSM (Model 102) and Standard, OS197 or OS197H Transfer Cask (TC). The source terms for the BPRAs are the same as given in Appendix J for the B&W 15x15 BPRAs. The approach used to assure that the neutron and gamma source spectrum and the source terms used, bound the fuel allowed per the fuel qualification tables is consistent with that used for the Standardized NUHOMS®-24P and -52B canister designs as described in Section 7.3.2.

The BPRA terminology employed throughout this Chapter is generic and refers to Control Components (CCs) inserted into the guide tubes of fuel assemblies during irradiation. The CCs include Burnable Poison Rod Assemblies (BPRAs), Thimble Plug Assemblies (TPAs), Control Rod Assemblies (CRAs), Rod Cluster Control Assemblies (RCCAs), Axial Power Shaping Rod Assemblies (APSRAs), Orifice Rod Assemblies (ORAs), Vibration Suppression Inserts (VSIs), Neutron Source Assemblies (NSAs), and neutron sources. Furthermore, non-fuel hardware that are positioned within the fuel assembly after the fuel assembly is discharged from the core such as Guide Tube or Instrument Tube Tie Rods or Anchors, Guide Tube Inserts, BPRA Spacer Plates or devices that are positioned and operated within the fuel assembly during reactor operation such as those listed above are also considered as CCs. The list of authorized BPRAs include but are not limited to the B&W Burnable Absorber Assemblies, Pyrex Burnable Absorber Assemblies, Wet Annular Burnable Absorbers. In order to qualify for loading, the source terms (or the dose equivalence) from any eligible BPRA design must be bounded by the design basis source terms employed in Table N.5-11.

P.2.1 Spent Fuel To Be Stored

As described in *Appendix* P.1, there are three design configurations for the NUHOMS®-24PTH DSC; S, L and S-LC. Each of the DSC configurations is designed to store intact (including reconstituted) and/or damaged *and/or failed* PWR fuel assemblies as specified in Table P.2-1 and Table P.2-3. The fuel to be stored is limited to a maximum assembly average initial enrichment of 5.0 wt. % ^{235}U. The maximum allowable assembly average burnup is limited to 62 GWd/MTU and the minimum cooling time is 3 years. The 24PTH-L and 24PTH-S-LC DSCs are also designed to store Control Components (CCs) with thermal and radiological characteristics as listed in Table P.2-2. The CCs include Burnable Poison Rod Assemblies (BPRAs), Thimble Plug Assemblies (TPAs), Control Rod Assemblies (CRAs), Rod Cluster Control Assemblies (RCCAs), Axial Power Shaping Rod Assemblies (APSRAs), Orifice Rod Assemblies (ORAs), Vibration Suppression Inserts (VSIs) Neutron Source Assemblies (NSAs) and Neutron Sources. *Non-fuel hardware that are positioned within the fuel assembly after the fuel assembly is discharged from the core such as Guide Tube or Instrument Tube Tie Rods or Anchors, Guide Tube Inserts, BPRA Spacer Plates or devices that are positioned and operated within the fuel assembly during reactor operation such as those listed above are also considered as CCs.*

Partial Length Shield Assemblies (PLSAs) for the Westinghouse 15x15 class, where part of the active fuel is replaced with steel are also included as authorized.

The NUHOMS®-24PTH DSC is also authorized to store fuel assemblies containing Blended Low Enriched Uranium (BLEU) fuel material. Fuel pellets containing BLEU fuel material are no different than UO_2 fuel pellets except for the presence of a higher quantity of cobalt impurity. The consideration of cobalt impurity only affects the gamma source terms for fuel assemblies located in the DSC periphery. This does not affect any criticality, thermal or structural analysis inputs for evaluation of fuel assemblies with BLEU material. The qualification of fuel assemblies containing BLEU fuel pellets will require an additional cooling time of three years to ensure that the source terms calculated with UO_2 material are bounding.

Fuel assemblies that contain fixed integral non-fuel rods are also considered as intact fuel assemblies. These fuel assemblies are different than reconstituted assemblies because fuel rods are not "replaced" by non-fuel rods, rather the non-fuel rods are part of the initial fuel design. The non-fuel rods displace the same amount of moderator, with zirconium-alloy (or aluminum) cladding and typically contain burnable absorber (or other non-fuel) material. The radiation and thermal source terms for the non-fuel rods are significantly lower than those of the fuel rods since there is no significant radioactive decay source. The internal pressure of the non-fuel rods after irradiation is lower than those of the fuel rods since there is no fission gas generation. The reactivity of the fuel rods (from a criticality standpoint) is significantly higher than that of non-fuel rods. In summary, the mechanical, thermal, shielding and criticality evaluations for these rods are bounded by those of the regular rods. Therefore, no further evaluations are required for the qualification of these fuel assemblies.

Reconstituted assemblies containing up to 10 replacement stainless steel rods per assembly or unlimited number of lower enrichment UO_2 rods are acceptable for storage in 24PTH DSC as intact fuel assemblies. The stainless steel rods are assumed to have two-thirds the irradiation time as the remaining fuel rods of the assembly. The reconstituted UO_2 rods are assumed to have the same irradiation history as the entire fuel assembly. The reconstituted rods can be at any location in the fuel assemblies. The maximum number of reconstituted fuel assemblies per DSC is four.

Table P.2-1
PWR Fuel Specification for the Fuel to be Stored in the NUHOMS®-24PTH DSC

PHYSICAL PARAMETERS: Fuel Class	Intact or damaged *or failed* unconsolidated B&W 15x15, WE 17x17, CE 15x15, WE 15x15, CE 14x14 and WE 14x14 class PWR assemblies (with or without control components) that are enveloped by the fuel assembly design characteristics listed in Table P.2-3. Equivalent reload fuel manufactured by other vendors but enveloped by the design characteristics listed in Table P.2-3 is also acceptable. *Damaged and/or failed fuel assemblies beyond the definition contained below are not authorized for storage.*
Fuel Damage	Damaged PWR fuel assemblies are assemblies containing missing or partial fuel rods or fuel rods with known or suspected cladding defects greater than hairline cracks or pinhole leaks. The extent of cladding damage in the fuel rods is to be limited such that a fuel *assembly needs to be handled by normal means.*
Partial Length Shield Assemblies (PLSAs)	WE 15x15 class PLSAs with *the* following characteristics are authorized: • Maximum burnup, 40 GWd/MTU • Minimum cooling time, 6.5 years • Maximum Decay Heat, 900 watts
Failed Fuel	*Failed fuel is defined as ruptured fuel rods, severed fuel rods, loose fuel pellets, or fuel assemblies that cannot be handled by normal means. Failed 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 can not 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 and it is located at a position of at least 10" above the top of the bottom shield plug of the DSC.* *Fuel debris may be associated with any type of UO_2 fuel provided that the maximum uranium content and initial enrichment limits are met. The total weight of each failed fuel plus all its contents shall be less than 1682 lb.*
Reconstituted Fuel Assemblies: • Maximum No. of Reconstituted Assemblies per DSC With *Irradiated* Stainless Steel Rods • Maximum No. of *Irradiated* Stainless Steel Rods per Reconstituted Fuel Assembly • Maximum No. of Reconstituted Assemblies per DSC with *Unlimited No. of* Low Enriched UO_2 Rods *and/or Unirradiated Stainless Steel Rods and/or Zr Rods or Zr Pellets.*	4 10 24

Table P.2-1
PWR Fuel Specification for the Fuel to be Stored in the NUHOMS®-24PTH DSC
(Continued)

Control Components (CCs)	• Up to 24 CCs are authorized for storage in 24PTH-L, *24PTH-S* and 24PTH-S-LC DSCs only. • Authorized CCs include Burnable Poison Rod Assemblies (BPRAs), Thimble Plug Assemblies (TPAs), Control Rod Assemblies (CRAs), Rod Cluster Control Assemblies (RCCAs), Axial Power Shaping Rod Assemblies (APSRAs), Orifice Rod Assemblies (ORAs), Vibration Suppression Inserts (VSIs), Neutron Source Assemblies (NSAs) and Neutron Sources. *Non-fuel hardware that are positioned within the fuel assembly after the fuel assembly is discharged from the core such as Guide Tube or Instrument Tube Tie Rods or Anchors, Guide Tube Inserts, BPRA Spacer Plates or devices that are positioned and operated within the fuel assembly during reactor operation such as those listed above are also considered as CCs.* • Design basis thermal and radiological characteristics for the CCs are listed in Table P.2-2.
Nominal Assembly Width *for Intact and Damaged Fuel Only*	8.536 inches
No. of Intact Assemblies	≤24
No. and Location of Damaged Assemblies	Up to 12 damaged fuel assemblies. Balance may be intact fuel assemblies, empty slots, or dummy assemblies depending on the specific heat load zoning configuration. Damaged fuel assemblies are to be placed in Locations A and/or B as shown in Figure P.2-6. The DSC basket cells which store damaged fuel assemblies are provided with top and bottom end caps to assure retrievability.
No. and Location of Failed Assemblies	*Up to 8 failed fuel assemblies. Balance may be intact and/or damaged fuel assemblies, empty slots, or dummy assemblies depending on the specific heat load zoning configuration.* *Failed fuel assemblies are to be placed in Location A as shown in Figure P.2-6. Failed fuel assembly/fuel debris is to be encapsulated in an individual failed fuel can (FFC) provided with a welded bottom closure and a removable top closure.*
Maximum Assembly plus CC Weight	1682 lbs

NUHOMS®-24PTH system described in Section P.2.1). A summary of the three system configurations analyzed in this chapter are summarized below:

System Configuration	DSC Type	Aluminum Inserts in Basket	Fuel Type	Total Heat Load per DSC, kW	Transfer Cask	Storage Module
1	24PTH-S or 24PTH-L	With inserts	All Fuels	40.8	OS197FC/*OS200FC*	HSM-H/*HSM-HS*
				31.2	OS197/ OS197H/*OS200*	HSM-H/*HSM-HS*
2	24PTH-S or 24PTH-L	No inserts	All Fuels	31.2	OS197FC/*OS200FC*	HSM-H/*HSM-HS*
3	24PTH-S-LC(1)	No inserts	B&W 15x15	24	Standardized TC	HSM-H/*HSM-HS* or HSM Model 102 *or 202*

(1) The maximum heat load allowed in the 24PTH-S-LC DSC is 24 kW. The HSM Model 102 is designed for a maximum heat load of 24 kW from a NUHOMS® 24P DSC as described in Section 8.1.3. Therefore no additional analysis of HSM Model 102 is required with 24PTH-S-LC DSC.

The thermal evaluations presented herein include steady state and transient analyses of the thermal response of the NUHOMS®-24PTH system components to a defined set of thermal loading conditions. These loading conditions envelope the thermal conditions expected during all normal, off-normal, and postulated accident loading, transfer and dry storage operations for the design basis thermal conditions as defined in Section P.2. The applicable allowable temperatures are presented and comparisons are made with calculated temperatures as the basis for acceptance.

The analyses conservatively apply a uniform maximum peaking factor of 1.11 [4.1] along the active fuel length to bound the effect of the decay heat flux varying axially along the active fuel length.

A description of the detailed analyses performed for the storage of NUHOMS®-24PTH DSC under normal, off-normal, and accident conditions is provided in Sections P.4.4 and for transfer is provided in Section P.4.5. Section P.4.6 describes the 24PTH DSC basket and fuel cladding analysis for storage and transfer conditions. *Section P.4.6.8 describes thermal analysis of the OS200 TC with the 24PTH DSC and Section P.4.6.9 describes evaluation of the 24PTH DSC with damaged/failed fuel assemblies (FAs).* The DSC cavity internal pressures are also calculated in Section P.4.6 for all conditions of storage and transfer. Section P.4.7 describes the evaluation performed for loading/unloading conditions. The thermal evaluation concludes that each of the three NUHOMS®-24PTH systems configurations listed above meets all the design criteria.

The effective thermal conductivity of the fuel assemblies used in the 24PTH DSC thermal analysis is based on the conservative assumption of radiation and conduction heat transfer only, where any convection heat transfer is neglected. In addition, the lowest effective thermal conductivity among the fuel assemblies to be stored using 24PTH-S DSC, -L DSC, and -S-LC DSC is selected as the basis for the thermal analysis. Section P.4.8 presents the calculations that determined the fuel assembly effective thermal conductivity in a helium or vacuum environment. The thermal analysis model conservatively neglects convection heat transfer in the basket regions.

The DSC basket and fuel cladding temperature calculation methodology has been benchmarked [4.20] against experimental data [4.21] obtained for the TN-24 cask.

Table T.2-1
BWR Fuel Specification for the Fuel to be Stored in the NUHOMS®-61BTH DSC

PHYSICAL PARAMETERS: Fuel Class	 Intact or damaged *or failed* 7x7, 8x8, 9x9 or 10x10 BWR assemblies manufactured by General Electric or Exxon/ANF or FANP or reload fuel manufactured by other vendors that are enveloped by the fuel assembly design characteristics listed in Table T.2-2. Damaged *and/or failed* fuel assemblies beyond the definition contained below are not authorized for storage.
Fuel Damage	Damaged BWR fuel assemblies are assemblies containing fuel rods with known or suspected cladding defects greater than hairline cracks or pinhole leaks. The extent of cladding damage in the fuel rods is to be limited such that a fuel *assembly needs to be handled by normal means.* Missing fuel rods are allowed.
Failed Fuel	*Failed fuel is defined as ruptured fuel rods, severed fuel rods, loose fuel pellets, or fuel assemblies that cannot be handled by normal means. Failed 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 can not 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 and it is located at a position of at least 10" above the top of the bottom shield plug of the DSC.* *Fuel debris may be associated with any type of UO_2 fuel provided that the maximum uranium content and initial enrichment limits are met. The total weight of each failed fuel can plus all its contents shall be less than 705 lb.*
RECONSTITUTED FUEL ASSEMBLIES: • Maximum No. of Reconstituted Assemblies per DSC with Irradiated Stainless Steel Rods • Maximum No. of Irradiated Stainless Steel Rods per Reconstituted Fuel Assembly • Maximum No. of Reconstituted Assemblies per DSC with unlimited number of low enriched UO2 rods or Zr Rods or Zr Pellets or Unirradiated Stainless Steel Rods	 4 10 61
No. of Intact Assemblies	≤61

Using the same loading as Appendix U, Section U.3.7.2, and based on NRC Reg. Guide 1.61, a damping value of three (3) percent is used for the 61BTH DSC high seismic load analysis. Based on the evaluation of the frequency content of the loaded HSM-HS, the amplified accelerations associated with the design basis seismic response spectra are determined and used for the structural evaluation of the 61BTH DSC.

T.3.7.2.1 DSC Seismic Evaluation

Standard Seismic Criteria

For the DSC inside the HSM, the results presented in *Appendix M,* Section M.3.7.3 are bounding. For the DSC inside the HSM-H, *the finite element models presented in Section T.3.6 are used and* the results *are* presented in *Table T.3-14 and Table T.3-15.*

High Seismic Criteria

The stability evaluation of 32PTH1 DSC inside the high- seismic HSM-HS presented in Appendix U, Section U.3.7.2 is applicable to the HSM-HS loaded with 61BTH Type 1 and Type 2 DSC.

In addition based on the frequency analysis of HSM-HS with the bounding 32PTH1 DSC, the maximum calculated accelerations for the 61BTH Type 1 and Type 2 DSC inside the HSM-HS when considering the higher seismic criteria are 2.0g transverse and 1.6g axial in the horizontal directions and 1.0g in the vertical direction.

The stresses in 61BTH Type 1 and Type 2 DSC shell due to vertical and horizontal high seismic loads criteria are determined and included in appropriate load combinations.

As stated in Section T.3.7.3.1, the maximum calculated seismic accelerations for the 61BTH Type 1 and Type 2 DSC inside the HSM-HS, when considering the higher seismic criteria, are 2.0g transverse and 1.6g axial in the horizontal directions and 1.0g in the vertical direction. See Table T.3.7-20 through Table T.3.7-25 for the high seismic case stress evaluations results for the 61BTH Type 1 and Type 2 DSC shell assembly components.

T.3.7.2.2 Basket Seismic Evaluation

Standard Seismic Criteria

The basket seismic analysis is performed using the models which were developed for normal and off-normal evaluations. A description of the seismic models, applied loads and associated result

Table T.3.7-18
DSC Enveloping Load Combination Table Notes

(1) See Table T.2-11 for load combination nomenclature.

(2) See Table T.2-12 for allowable stress criteria. Material properties were obtained from Table 8.1-3 at a design temperature of 750°F or as noted.

(3) In accordance with the ASME Code, thermal stresses need not be included in Service Level D load combinations.

(4) Not used.

(5) The maximum side drop membrane + bending stress is highly localized near the cask rail, at the outer bottom cover plate. The maximum temperature in this region is less than 240°F (temperature case 2).

(6) The maximum side drop membrane + bending stress is highly localized over the cask rail. The maximum temperature in this region is less than 300°F.

Table T.3.7-19
61BTH Basket, Enveloping Stress Results – High Seismic Loading

Design Option	Storage Loading	Stress Level	Component	Material	Max.Stress Intensity (ksi)		Allowable Stress Intensity (ksi)	
					P_m (ksi)	P_m+P_b (ksi)	P_m (ksi)	P_m+P_b (ksi)
61BTH Type 1	1.6g Axial +2g Transverse +3.646g Vertical	D	Fuel Compartment (750 °F)	SA-240 Type 304	16.0	22.9	44.3	57.0
			Transition Rail (550 °F)	SA-240 Type 304	7.5	23.7	44.4	57.1
			Canister Shell (500 °F)	SA-240 Type 304	2.0	32.1	44.4	57.1
61BTH Type 2	1.6g Axial +2g Transverse +3.646g Vertical	D	Fuel Compartment (750 °F)	SA-240 Type 304	11.1	22.0	44.3	57.0
			Transition Rail R45 (550 °F)	SA-240 Type 304	9.9	23.7	44.4	57.1
			Canister Shell (500 °F)	SA-240 Type 304	2.3	31.5	44.4	57.1

U.2.1 Spent Fuel To Be Stored

As described in Chapter U.1, there are three alternate design configurations for the NUHOMS®-32PTH1 DSC depending on the canister length: a short (185.75") DSC designated as 32PTH1-S, a medium (193.00") DSC designated as 32PTH1-M, and a long (198.5") DSC designated as 32PTH1-L DSC. Each of the DSC configurations is designed to store intact (including reconstituted) and/or damaged PWR fuel assemblies as specified in Table U.2-1 and Table U.2-3. The fuel to be stored is limited to a maximum assembly average initial enrichment of 5.0 wt. % U-235. The maximum allowable assembly average burnup is limited to 62 GWd/MTU and the minimum cooling time is 3 years. Each of the DSC types is designed to store Control Components (CCs) with thermal and radiological characteristics as listed in Table U.2-2. The CCs include Burnable Poison Rod Assemblies (BPRAs), Thimble Plug Assemblies (TPAs), Control Rod Assemblies (CRAs), Rod Cluster Control Assemblies (RCCAs), Axial Power Shaping Rod Assemblies (APSRAs), Orifice Rod Assemblies (ORAs), Vibration Suppression Inserts (VSIs), Neutron Source Assemblies (NSAs) and Neutron Sources. *Furthermore, non-fuel hardware that are positioned within the fuel assembly after the fuel assembly is discharged from the core such as Guide Tube or Instrument Tube Tie Rods or Anchors, Guide Tube Inserts, BPRA Spacer Plates or devices that are positioned and operated within the fuel assembly during reactor operation such as those listed above are also considered as CCs.*

The NUHOMS®-32PTH1 DSC is also authorized to store fuel assemblies containing Blended Low Enriched Uranium (BLEU) fuel material. Fuel pellets containing BLEU fuel material are no different than UO_2 fuel pellets except for the presence of a higher quantity of cobalt impurity. The consideration of cobalt impurity only affects the gamma source terms for fuel assemblies located in the DSC periphery. This does not affect any criticality, thermal or structural analysis inputs for evaluation of fuel assemblies with BLEU material. The qualification of fuel assemblies containing BLEU fuel pellets will require an additional cooling time of three years to ensure that the source terms calculated with UO_2 material are bounding.

Fuel assemblies that contain fixed integral non-fuel rods are also considered as intact fuel assemblies. These fuel assemblies are different than reconstituted assemblies because fuel rods are not "replaced" by non-fuel rods, rather the non-fuel rods are part of the initial fuel design. The non-fuel rods displace the same amount of moderator, with zirconium-alloy (or aluminum) cladding and typically contain burnable absorber (or other non-fuel) material. The radiation and thermal source terms for the non-fuel rods are significantly lower than those of the fuel rods since there is no significant radioactive decay source. The internal pressure of the non-fuel rods after irradiation is lower than those of the fuel rods since there is no fission gas generation. The reactivity of the fuel rods (from a criticality standpoint) is significantly higher than that of non-fuel rods. In summary, the mechanical, thermal, shielding, and criticality evaluations for these rods are bounded by those of the regular fuel rods. Therefore, no further evaluations are required for the qualification of these fuel assemblies.

Reconstituted assemblies containing up to 10 replacement irradiated stainless steel rods per assembly or 32 lower enrichment UO_2 rods instead of Zircaloy clad enriched UO_2 rods, or 32 Zr rods or Zr pellets, or unirradiated stainless steel rods are acceptable for storage in 32PTH1 DSC as intact fuel assemblies with a slightly longer cooling time than that required for a standard assembly. The stainless steel rods are assumed to have two-thirds the irradiation time as the

remaining fuel rods of the assembly. The reconstituted UO_2 rods are assumed to have the same irradiation history as the entire fuel assembly. The reconstituted rods can be at any location in the fuel assemblies. The maximum number of reconstituted fuel assemblies per DSC is four with irradiated stainless steel replacement rods or 32 with UO_2 replacement rods.

The NUHOMS®-32PTH1 DSCs can also accommodate up to a maximum of 16 damaged fuel assemblies placed in the center cells of the DSC as shown in Figure U.2-1 through Figure U.2-3. Damaged PWR fuel assemblies are assemblies containing missing or partial fuel rods, or fuel rods with known or suspected cladding defects greater than hairline cracks, or pinhole leaks. The extent of damage in the fuel assembly is to be limited such that a fuel assembly is being able to be handled by normal means and retrievability is assured following normal and off-normal conditions. The DSC basket cells which store damaged fuel assemblies are provided with top and bottom end caps to assure retrievability.

A 32PTH1 DSC containing less than 32 fuel assemblies may contain dummy fuel assemblies in the empty slots. The dummy assemblies are unirradiated, stainless steel encased structures that approximate the weight and center of gravity of a fuel assembly.

The 32PTH1 DSC basket is designed with 2 alternate options: Type 1 basket with solid aluminum transition rails and Type 2 basket with steel transition rails including aluminum inserts. Type 1 basket is the preferred option for canisters with high decay heat loads, since the solid aluminum rails allow a more direct heat conduction path from the basket edge to the DSC shell.

The NUHOMS®-32PTH1 DSCs may store up to 32 PWR fuel assemblies arranged in any of the three alternate heat load zoning configurations (HLZC) as shown in Figure U.2-1 through Figure

Table U.2-1
PWR Fuel Specification for the Fuel to be Stored in the NUHOMS®-32PTH1 DSC

PHYSICAL PARAMETERS: Fuel Class	Intact or damaged unconsolidated B&W 15x15, WE 17x17, CE 15x15, WE 15x15, CE 14x14, WE 14x14 and CE 16x16 class PWR assemblies (with or without control components) that are enveloped by the fuel assembly design characteristics listed in Table U.2-3. Reload fuel manufactured by other vendors but enveloped by the design characteristics listed in Table U.2-3 is also acceptable. Damaged fuel assemblies beyond the definition contained below are not authorized for storage.
Fuel Damage	Damaged PWR fuel assemblies are assemblies containing missing or partial fuel rods or fuel rods with known or suspected cladding defects greater than hairline cracks or pinhole leaks. The extent of damage in the fuel assembly is to be limited such that a fuel assembly is being able to be handled by normal means. *Missing fuel rods are allowed.* *Damaged fuel assemblies shall also contain top and bottom end fittings or nozzles or tie plates depending on the fuel type.*
RECONSTITUTED FUEL ASSEMBLIES: • Maximum No. of Reconstituted Assemblies per DSC with Irradiated Stainless Steel Rods • Maximum No. of Irradiated Stainless Steel Rods per Reconstituted Fuel Assembly • Maximum No. of Reconstituted Assemblies per DSC with Unlimited Number of Low Enriched UO_2 Rods, or Zr Rods or Zr Pellets or Unirradiated Stainless Steel Rods	4 10 32
Control Components (CCs)	• Up to 32 CCs are authorized for storage in 32PTH1-S, 32PTH1-M and 32PTH1-L DSCs. • Authorized CCs include Burnable Poison Rod Assemblies (BPRAs), Thimble Plug Assemblies (TPAs), Control Rod Assemblies (CRAs), Rod Cluster Control Assemblies (RCCAs), Axial Power Shaping Rod Assemblies (APSRAs), Orifice Rod Assemblies (ORAs), Vibration Suppression Inserts (VSIs), Neutron Source Assemblies (NSAs), and Neutron Sources. *Non-fuel hardware that are positioned within the fuel assembly after the fuel assembly is discharged from the core such as Guide Tube or Instrument Tube Tie Rods or Anchors, Guide Tube Inserts, BPRA Spacer Plates or devices that are positioned and operated within the fuel assembly during reactor operation such as those listed above are also considered as CCs.* • Design basis thermal and radiological characteristics for the CCs are listed in Table U.2-2.
No. of Intact Assemblies	≤32
No. and Location of Damaged Assemblies	Up to 16 damaged fuel assemblies. Balance may be intact fuel assemblies, or dummy assemblies which are authorized for storage in 32PTH1 DSC. Damaged fuel assemblies are to be placed in the center 16 locations as shown in Figure U.2-1, Figure U.2-2 and Figure U.2-3. The DSC basket cells which store damaged fuel assemblies are provided with top and bottom end caps.
Maximum Assembly plus CC Weight	1715 lbs

Table U.3.1-4
NUHOMS® DSCs System Configurations for On-Site Transfer and Storage

DSC	Max Heat Load (Storage)	DSC Max. Wt. (kips)	On-Site Transfer	On-Site Storage
32PTH1	24.0 (Type 2)	110	OS200/OS200FC	HSM-H/HSM-HS
	31.2 (Type 1) (Type 2)	110	OS200 (Type 2)/OS200FC (Type 1)	HSM-H/HSM-HS
	40.8 (Type 1)	110	OS200FC	HSM-H/HSM-HS
61BT	18.3	88.4	OS197 OS200	HSM 80/102/152/202/HSM-HS
32PT	24	101.4	OS197 OS200	HSM 80/102/152/202/HSM-HS
24PTH	24.0 (24PTH-S-LC only)	89.5	Standardized TC (solid neutron shield)	HSM 102/152/152/202/HSM-H/HSM-HS
	40.8	93.1	OS197 (up to 31.2 kW w/ aluminum inserts in basket rails) OS197FC/OS200/OS200FC (up to 40.8 kW)	HSM-H HSM-HS
61BTH	22.0 (Type 1)	88.7	OS197/OS197FC-B/ OS200/OS200FC	HSM 80/102/152/202/HSM-H/HSM-HS
	31.2 (Type 2)	83.1	OS197/OS197FC-B/ OS200/ OS200FC (up to 22.0 kW) OS197FC-B/OS200FC (22.0 to 31.2 kW)	HSM-H/HSM-HS
69BTH	24.0	103.9	OS200/OS200FC	HSM-H/HSM-HS
	35.0	103.9	OS200FC	HSM-H/HSM-HS
37PTH	24.0	108.1	OS200/OS200FC	HSM-H/HSM-HS
	30.0	108.1	OS200FC	HSM-H/HSM-HS

Notes:

(1) Reference to Type 1 and Type 2 means basket types as described in the applicable Appendix for each DSC in the UFSAR.

Proprietary information on pages Y.3-36 through Y.3-61 withheld pursuant to 10 CFR 2.390

$\lambda = i\pi$; for lowest natural frequency, i = 1.

Substituting yields: f_1 = 43.7 Hertz.

The DSC spectral accelerations at this frequency correspond to the zero period acceleration.

Y.3.7.2.1.2 DSC Seismic Stress Analysis

For the DSC seismic stress analysis, the DSC is conservatively assumed to be resting on a single support rail inside the HSM-H. The seismic load combination for the DSC shell include stresses due to deadweight and pressure loads in addition to the standard seismic loads (load combinations. *Table Y.3.7-10* summarizes the 69BTH DSC resulting stress intensities due to the seismic load combination and compares the results against the Level C allowable stress intensities at temperature. The high seismic loads stress evaluation results are evaluated and compared against Level D allowable stress criteria intensities, and are bounded by the Level D load combination results as shown in *Table Y.3.7-11*.

A single axial retainer is included in the design of the DSC support structure inside the HSM-H. Two axial seismic retainers are used in the design of the HSM-HS to prevent sliding of the DSC in the axial direction during a postulated seismic event. In addition, for the HSM-HS design, it is required that the gap between the axial retainer and the DSC be limited to 1/8 inch after loading the 69BTH DSC into the HSM-HS. The stresses induced in the DSC shell and bottom cover plate due to the restraining action of these retainers for the horizontal seismic load, applied along the axis of the DSC, are included in the seismic response evaluation of the DSC shell assembly.

Y.3.7.2.2 Basket Seismic Evaluation

The basket seismic analysis is performed using the models which were developed for normal and off-normal evaluations. A description of the seismic models, applied loads and associated results is presented in Section Y.3.6.1.3.4B. The basket natural frequency is also discussed in Section Y.3.6.1.3.4C.

Y.3.7.2.3 HSM-H and HSM-HS Seismic Evaluation

The seismic results of the 69BTH stored in an HSM-H are bounded by results presented in Appendix *U*, Section *U*.3.7.2.3. The seismic results of 69BTH stored in an HSM-HS are bounded by results presented in Appendix U, Section U.3.7.2.4.

Y.3.7.2.4 DSC Support Structure Seismic Evaluation

The seismic results of the 69BTH DSC support structure inside the HSM-H are bounded by those presented in Appendix P, Section P.3.7.11.6.4. The seismic results of the 69BTH DSC support structure inside the HSM-HS are bounded by those presented in Appendix U, Section U.3.7.11.6.4.

Y.3.7.2.5 DSC Axial Retainer Seismic Evaluation

The HSM-H axial retainer is qualified for a maximum DSC weight of 110 kips in Appendix P, whereas the maximum 69BTH DSC weight is 104 kips. Therefore, Appendix P, Section

3.50 Barsoum, R.S., "on the Use of Isoparametric Finite Elements in Linear Elastic Fracture Mechanics," International Journal for Numerical Methods in Engineering, Vol. 10, 1976.

3.51 Safety Analysis Report, "NUHOMS® HD Horizontal Modular Storage System for Irradiated Nuclear Fuel," NRC Docket No. 72-1030.

3.52 LSDYNA Version 971 R2 Rev. 7600 1224 "LS-DYNA Keyword User's Manual", Livermore Software Technology Corporation (LSTC).

3.53 ANP-2899NP, Revision 0, "Fuel Design Evaluation for ATRIUM 10XM BWR Reload Fuel," NRC ADAMS: ML101100643

3.54 Ledergerber, et al., "Fuel Performance Beyond Design - Exploring the Limits." Proceedings of 2010 LWR Fuel Performance/TopFuel/WRFMP Orlando, Florida, USA, September 26-29, 2010. pp. 513-524.

3.55 GE Energy, Nuclear (GEEN, Global Nuclear Fuel (GNF), "BWR Control Rod Drop Accident", NRC RIA Workshop, Bethesda, MD, 2006, NRC ADAMS: ML063190108

3.56 USNRC, NUREG-0800, Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants, Chapter 4, Section 4.2, "Fuel System Design," Revision 3.

3.57 *Fuel Material Technology Report 2007, Volume II, Advanced Nuclear Technology International, Sweden. http://www.antinternational.com/fileadmin/Products_and_handbooks/FMTR/FMTR_First_chapter.pdf*

10. Helium [4.4]

Temperature (K)	Thermal conductivity (W/m-K)	Temperature (°F)	Thermal conductivity (Btu/hr-in-°F)
300	0.1499	80	0.0072
400	0.1795	260	0.0086
500	0.2115	440	0.0102
600	0.2466	620	0.0119
800	0.3073	980	0.0148
1000	0.3622	1340	0.0174
1050	0.3757	1430	0.0181

The above data are calculated based on the following polynomial function from [4.4].

$k = \sum C_i T^i$ for conductivity in (W/m-K) and T in (K)

For 300 < T < 500 K		For 500< T < 1050 K	
C0	-7.761491E-03	C0	-9.0656E-02
C1	8.66192033E-04	C1	9.37593087E-04
C2	-1.5559338E-06	C2	-9.13347535E-07
C3	1.40150565E-09	C3	5.55037072E-10
C4	0.0E+00	C4	-1.26457196E-13

No density or specific heat is considered for helium for conservatism.

11. Air [4.4]

Temperature (K)	Thermal conductivity (W/m-K)	Temperature (°F)	Thermal conductivity (Btu/hr-in-°F)
250	0.02228	-10	0.0011
300	0.02607	80	0.0013
400	0.03304	260	0.0016
500	0.03948	440	0.0019
600	0.04557	620	0.0022
800	0.05698	980	0.0027
1000	0.06721	1340	0.0032

The above data are calculated based on the following polynomial function from [4.4].

$k = \sum C_i T^i$ for conductivity in (W/m-K) and T in (K)

For 250 < T < 1050 K	
C0	-2.2765010E-03
C1	1.2598485E-04
C2	-1.4815235E-07
C3	1.7355064E-10
C4	-1.0666570E-13
C5	2.4766304E-17

No density or specific heat is considered for air in the thermal models for conservatism.

Y.4.4 Thermal Evaluation of HSM-H/HSM-HS with 69BTH DSC

The maximum decay heat load considered for the NUHOMS®-69BTH DSC is described in Section Y.4.1 is 35 kW. The maximum heat load of 35 kW is used in this analysis to determine the bounding maximum temperatures. As described in Appendix U, Section U.1.2.1.2, the HSM-H and HSM-HS modules are thermally identical. Therefore, the thermal evaluation of HSM-H is also applicable to HSM-HS.

Y.4.4.1 Ambient Temperature Specification

Ambient temperatures in the range of 0–100°F are considered as normal storage conditions. A maximum day temperature of 117°F is considered an off-normal, hot storage condition. A 24-hour average ambient temperature of 105°F is conservatively used for the off-normal steady state analysis, based on the 102°F calculated in Appendix M, Section M.4.5. The lowest off-normal ambient temperature is considered to be -40°F.

In addition to the above ambient temperatures, an extreme ambient temperature (24-hour average ambient temperature) of 117°F corresponding to maximum day temperature of 132°F is considered as an accident condition for storage when 69BTH DSC is loaded within HSM-H.

Y.4.4.2 Thermal Analysis of HSM-H with 69BTH DSC

The HSM-H is designed to provide an independent, passive system with substantial structural capacity to ensure safe storage of spent fuel assemblies in NUHOMS®-69BTH DSCs.

The analyses in this section determine the temperature distribution on the HSM-H walls and on the 69BTH DSC shell. The DSC shell temperatures are used to calculate the basket and peak fuel cladding temperature in a detailed model of the 69BTH DSC and basket described in Section Y.4.6. For the analytical purpose of calculating the maximum temperatures, an HSM-H centered in a group of HSM-Hs, each loaded with a 69BTH DSC, is considered for the analysis. Rows of modules are assumed to exist back-to-back for this model.

The HSM-H has been designed to accommodate the DSCs with a maximum decay heat load of 40.8 kW (see Appendix P, Section P.4.4 and Appendix U, Section U.4.4). The maximum decay heat load that can be stored in the 69BTH DSC is 35 kW, which is significantly less than the design heat load of the HSM-H. The thermal analysis results for decay heat load of 40.8 kW are used for structural evaluation of HSM-H.

For thermal evaluations, the temperature distributions in the HSM-H are calculated at -40°F (off-normal cold) and 117°F (off-normal hot) ambient temperatures with a maximum decay heat load of 35 kW to bound the temperatures for normal conditions (0°F and 100°F ambient). The temperature distributions for these conditions are determined using steady-state models.

Since the HSM-H is located outdoors, there is a remote probability that the air inlet or outlet openings will become blocked by debris from events such as flooding, high wind, and tornados. The perimeter security fence around ISFSI and the location of the air inlet and outlet openings reduce the probability of such an accident. A complete blockage of all air inlets and outlets

simultaneously is not a credible event. However, to bound this scenario, analysis is carried out assuming *50% blockage of the HSM-H inlet vents under off-normal storage conditions and* complete blockage of the inlet and outlet vents as an accident case.

In addition to the complete blockage of the inlet and outlet vents, the HSM-H loaded with 69BTH DSC is evaluated for an extreme hot ambient temperature of 117 °F (24-hour average ambient temperature) as an accident condition.

Y.4.4.3 HSM-H Air Flow Analysis (Stack Effect Calculations)

The methodology used in the HSM-H airflow analysis (stack effect calculations) is presented in Appendix P, Section P.4.4.3. Different equations for computing the total pressure loss due to flow losses, air mass flow rate, temperature rise from air inlet to outlet, and the stack average temperature are also provided in Appendix P, Section P.4.4.3. *For the HSM-H with 50% blocked inlet vents, the air flow analysis includes the additional resistance and loss coefficients due to the decrease in the inlet surface area.*

Using these loss coefficients in an equation for ΔT_{HSM-H}, the exit and stack air temperature for the normal and off-normal cases are calculated. A summary of the calculation results for mass flow rates, total loss coefficients, exit and mean air temperatures for normal and off-normal storage conditions is provided in Table Y.4-1. These bulk air temperatures are used in the subsequent HSM-H analyses to calculate the temperatures throughout the HSM-H and the 69BTH DSC shell.

The accident blocked vents condition conservatively assumes no closed cavity convection.

Y.4.4.4 Description of the Thermal Model of HSM-H with 69BTH DSC

A half symmetric, three dimensional, ANSYS [4.18] finite element model of the HSM-H loaded with a 69BTH DSC is shown in Figure Y.4-1. Except for the dimensions of the DSC and its heat load, this model is identical to the HSM-H model described in Appendix U, Section U.4.4.4 used for analysis of 32PTH1 DSC.

The thermal model consists of SOLID70 conduction elements that simulate concrete and steel support structures of the HSM-H. SHELL57 elements superimposed on SOLID70 elements, as required, for generation of radiating surfaces for the MATRIX50 super elements. Radiation between the DSC shell, heat shields, and HSM-H walls is modeled using the ANSYS /AUX12 methodology. The SHELL57 elements used as radiation surfaces are unselected prior to solving the model. To reduce the number of nodes associated with the model's super-elements, the web of the supporting beam is modeled using only SHELL57 elements. As such, conservatively, radiation is not applied on the web of the supporting beam. This methodology is valid since the supporting beam's web greatly shields the support steel from the DSC radiation via its own flanges. The properties and dimensions of the support beam, such as the thickness of the web, are given as real constants to the appropriate SHELL57 elements.

During storage, up to 7" of the bottom part of the 69BTH DSC resides within the access port. Convection is conservatively omitted in the access port between the DSC and the concrete module.

Y.4.4.4.1 Boundary Conditions

The boundary conditions for HSM-H model are applied using the same methodology described in Appendix P, Section P.4.4.4. Ambient, exit and mean bulk air temperatures listed in Table Y.4-1 are used to apply the boundary conditions.

The correlation for convection coefficients over the HSM-H surfaces, including the HSM-H vertical flat surfaces, horizontal surfaces, the side heat shield, the top heat shield and the horizontal DSC cylinder surface are discussed in detail in Appendix P, Section P.4.9. Convection

As seen in Table Y.4-3, the normalized area under peaking factor curve is smaller than 1.0. To avoid any degradation of decay heat load, a correction factor of 1.00697 calculated as follows is used when applying the peaking factors.

$$\text{Nomalized Area under Curve} = \frac{\text{Area under Axial Heat Profile}}{\text{Active Fuel Length}} = 0.99308$$

Active fuel length = 144"

$$\text{Correction Factor} = \frac{1}{\text{Normalized Area under Curve}} = 1.00697$$

The heat generating rates for the elements representing the active fuel are calculated based on the HLZCs for each basket type. The HLZCs and their restrictions for 69BTH basket are shown in Chapter Y.2, Figures Y.2-1 through Y.2-6.

DSC Shell Temperatures

The DSC shell temperatures for normal, off-normal, and accident conditions are retrieved from the HSM-H and TC models described in Sections Y.4.4 and Y.4.5 and transferred to the DSC/basket model. The load cases considered for thermal evaluation of the 69BTH DSC model are summarized in Table Y.4-4.

Load case S3 in Table Y.4-4, with the maximum off-normal temperature of 117°F in HSM-H, bounds the normal storage cases (load cases S1 and S2). The DSC shell temperatures from load case S3 and S4 are chosen to determine the bounding maximum fuel cladding and basket component temperatures for all normal and off-normal storage conditions.

The DSC shell temperatures from load case S3A in Table Y.4-4, corresponding to off-normal storage condition with 50% blockage of HSM-H inlet vents, is chosen to demonstrate that the maximum fuel cladding and basket component temperatures remain below the allowable limit.

The DSC shell temperatures from load case S5 in Table Y.4-4, corresponding to HSM-H blocked vent accident condition, are chosen to determine the maximum fuel cladding and basket component temperatures for storage accident conditions. *The DSC shell temperatures from load case S6 in Table Y.4-4, corresponding to HSM-H accident condition with extreme hot ambient temperature, are chosen to demonstrate that the maximum fuel cladding and basket component temperatures for storage accident conditions are bounded by blocked vent accident case (load case S5).*

The DSC shell temperatures from load case T6 in Table Y.4-4, with the maximum normal temperature of 100°F in OS200/OS200FC TC, are chosen to determine the maximum fuel cladding and basket component temperatures for all normal and off-normal transient transfer conditions. The evaluations in Appendix U, Section U.4.5 shows that the normal transfer condition with insolation bounds the off-normal transfer due to the requirement of providing sunshade during off-normal transfer.

The DSC shell temperatures from load case T13 in Table Y.4-4, with the maximum normal temperature of 117°F in OS200/OS200FC TC, are chosen to determine the maximum fuel cladding and basket component temperatures for all normal and off-normal transient transfer conditions when air circulation option is used. Since the entrance air temperature of 117°F is higher than all other cases, load case T13 results in the bounding maximum temperatures for load cases T11 and T12.

Based on evaluations in Appendix U, Section U.4.5.4.2, the accident case of 'loss of neutron shield', corresponding to load case T14 in Table Y.4-4, results in the highest maximum

Table Y.4-1
Summary of Air-Flow Calculation Results

Operating Conditions	T_{amb} (°F)	Mass Flow Rate (lb_m/s)	Total Loss Coefficient (ft^{-4})	ΔT_{HSM} (°F)	T_{exit} (°F)	T_{mean} (°F)
Normal	0	1.973	0.1049	70	70	35
	100	1.592	0.1070	86	186	143
Off-Normal	-40	2.183	0.1039	63	23	-8
	117	1.578	0.1071	87	192	149
Off-Normal (50% Blockage of HSM-H Inlet Vents)	*117*	*1.435*	*0.1400*	*96*	*201*	*153*
Accident (Extreme Hot Ambient Temperature)	*117*	*1.545*	*0.1074*	*89*	*206*	*161*

Table Y.4-2
Maximum HSM-H Components Temperatures

Components	Normal Hot/ Normal Cold T_{max} (°F)	Off-Normal Hot T_{max} (°F)	*Off-Normal Hot with 50% Blockage of HSM-H Inlet Vents* T_{max} (°F)	Off-Normal Cold T_{max} (°F)	Blocked Vent @ 40 Hours T_{max} (°F)	*Extreme Hot Ambient Temperature* T_{max} (°F)
Concrete	<275	275	*277*	130	433	*286*
DSC shell	<454	454	*455*	355	590	*463*
Side heat shield (SHS)	<242	242	*247*	61	454	*256*
Support structure	<330	330	*331*	190	526	*342*
Top heat shield (THS)	<246	246	*253*	62	404	*261*

Table Y.4-4
Summary of Load Cases for Thermal Analysis of the 69BTH DSC

Load Case	Heat Load (kW)	Operation Condition	Description	Ambient Temperature (°F)	Insolation
S1	35	Normal storage	Normal hot, steady state	100	Yes
S2	35	Normal storage	Normal cold, steady state	0	No
S3	35	Off-normal storage	Off-normal hot, steady state	117	Yes
S3A	*35*	*Off-Normal Storage*	*Off-normal hot, steady-state with 50% blockage of HSM-H inlet vents*	*117*	*Yes*
S4	35	Off-normal storage	Off-normal cold, steady state	-40	No
S5	35	Accident storage	Blocked vents @ 40 hours transient	117	Yes
S6	*35*	*Accident storage*	*Extreme hot ambient temperature, steady state*	*132*(1)	*Yes*
T1	24	Normal transfer	Normal hot, steady state	100	Yes
T2	24	Normal transfer	Normal cold, steady state	0	No
T3	24	Normal transfer	Vertical operations, cold, steady state	0	No
T4	24	Normal transfer	Vertical operations, hot, steady state	140	No
T5	24	Off-normal transfer	Off-normal hot, steady state	117	No
T6	35	Normal transfer	Normal hot, transient	100	Yes
T7	35	Normal transfer	Normal cold, transient	0	No
T8	35	Normal transfer	Vertical operations cold, transient	0	No
T9	35	Normal transfer	Vertical operations hot, transient	140	No
T10	35	Off-normal transfer	Off-normal hot, transient	117	No
T11	35	Normal transfer	Normal hot with air circulation, steady state	100	Yes
T12	35	Normal transfer	Normal cold with air circulation, steady state	0	No
T13	35	Off-normal transfer	Off-normal hot with air circulation, steady state	117	No
T14	35	Accident transfer	Loss of air circulation, loss of neutron shield, and no sunshade	117	Yes
T15	35	Accident transfer	Loss of air circulation	117	No

(1) *The maximum day temperature of 132°F corresponds to a 24-hour average temperature of 117°F.*

Table Y.4-9
Maximum Fuel Cladding Temperatures for Storage/Transfer Conditions

Load Case No.	Load Case Description	Fuel Cladding (°F)	Limit (°F)
S1[1]	Normal storage, 35 kW, 100°F ambient with insolation	<728	752
S3	Off-normal storage, 35 kW, 117°F ambient with insolation	728	1058
S3A	*Off-normal storage, 35 kW, 117°F ambient with insolation and 50% blockage of HSM-H inlet vent*	*731*	
S4	Off-normal storage, 35 kW, -40°F ambient, no insolation	618	
S5	Accident storage, blocked vents @ 40 hours	859	
S6	*Accident storage, extreme hot ambient temperature*	*737*	
T1	Normal transfer, 24 kW, 100°F ambient with insolation	628	752
T6	Normal transfer, 35 kW, 100°F ambient with insolation, transient at operation time 15.75 hr	731	
T7	Normal transfer, 35 kW, 0°F ambient, no insolation, transient at operation Time 24.75 hr	723	
T12	Normal transfer, 35 kW, 0°F ambient, no insolation, Steady State with Air Circulation	595	
T13	Off-normal transfer, 35 kW, 117°F ambient with insolation, steady state with air circulation	693	
T14	Accident transfer, 35 kW, loss of air circulation, loss of neutron shield, and no sunshade	918	1058

(1) The results for normal storage condition (S1) with 100°F and insolation are bounded by the results for off-normal storage condition (S3) with 117°F and insolation.

Table Y.4-10
Maximum Basket Component Temperatures for Storage/Transfer Conditions

Load Case No.[1]	Fuel Compartment (°F)	Al/Poison Plates (°F)	Basket Rails (°F)	Top Shield Plug (°F)	Bottom Shield Plug (°F)	DSC Shell (°F)
S1[2]	<713	<713	<479	<204	<365	<451
S3	713	713	479	204	365	451
S3A	*716*	*716*	*482*	*209*	*368*	*455*
S4	601	600	357	40	237	351
S5	846	845	628	320	498	590
S6	*722*	*722*	*489*	*218*	*376*	*459*
T1	614	614	463	336	379	429
T6	714	714	496	308	390	451
T7	706	706	492	300	380	451
T12	573	573	373	254	222	340
T13	676	675	473	361	352	443
T14	901	901	694	499	569	651

(1) See Table Y.4-4 for a description of the load cases.

(2) The results for normal storage condition (S1) with 100°F and insolation are bounded by the results for off-normal storage condition (S3) with 117°F and insolation.

Table Y.4-11
Average Temperatures for Storage/Transfer Conditions

Load Case No.[1]	Fuel Assembly (°F)	Helium Backfill Gas (°F)	Basket Compartment (°F)	R90 Rail, Bottom (°F)	R90 Rail, Top (°F)	DSC Shell (°F)
S1 [2]	<580	<392	<601	<466	<476	<406
S3	580	392	601	466	476	406
S3A	*583*	*395*	*604*	*468*	*479*	*410*
S4	466	261	486	354	346	277
S5	714	528	739	596	626	562
S6	*589*	*403*	*611*	*476*	*486*	*417*
T1	515	407	530	382	460	398
T6	590	432	612	419	500	421
T7	581	421	604	400	498	413
T12	436	307	454	348	260	243
T13	553	426	573	450	399	377
T14	767	613	795	630	688	626

(1) See Table Y.4-4 for a description of the load cases.
(2) The results for normal storage condition (S1) with 100°F and insolation are bounded by the results for off-normal storage condition (S3) with 117°F and insolation.

Y.5.2 Source Specification

Thermal and radiological source terms are calculated with the SAS2H/ORIGEN-S modules of SCALE 4.4 [5.1] for the fuel. The SAS2H/ORIGEN-S results are used to develop the fuel qualification tables listed in Tables Y.2-5 through Y.2-16 of Appendix Y.2 and the design basis fuel source terms suitable for use in the shielding calculations.

The GE-2,3 7x7 Type G2A assembly is the bounding fuel assembly design for shielding purposes because it has the highest initial heavy metal loading in the fuel and ^{59}Co content in the hardware regions as compared to the 8x8, other 9x9, and 10x10 fuel assemblies which are also authorized contents of the NUHOMS®-69BTH DSC. The neutron flux during reactor operation is peaked in the active fuel region of the fuel assembly and drops off rapidly outside the active fuel region. Much of the fuel assembly hardware is outside of the *active fuel or* in-core region of the fuel assembly. To account for this reduction in neutron flux, the fuel assembly is divided into four exposure "regions." The four axial regions used in the source term calculation are: the bottom (nozzle) region, the *active fuel* region, the (gas) plenum region, and the top (nozzle) region. The GE 7x7 fuel assembly masses for each irradiation region are listed in Table Y.5-5. The light elements that make up the various materials for the various fuel assembly materials are taken from Reference [5.5] and are listed in Table Y.5-6. The design basis heavy metal loading is 0.198 MTU. These masses are irradiated in the appropriate fuel assembly region in the SAS2H/ORIGEN-S models. To account for the reduction in neutron flux outside the *active fuel* regions, neutron flux (fluence) correction factors are applied to the light element composition for each region. The neutron flux correction factors which are from Reference [5.13] are given in Table Y.5-7.

Evaluations of the existing light water reactor (LWR) fuel data with SAS2H and the 44-group ENDF/B-V library used in the calculation of the design basis source terms are documented in Reference [5.11, 5.12]. These comparisons all show generally good agreement between the calculations and measurements, and show no trend as a function of burnup in the data that would suggest that the isotopic predictions, and therefore neutron and gamma source terms, would not be in good agreement. A similar conclusion is also reached by the results documented in JAERI report [5.9]. In fact, for the case with 46,460 MWd/MTU burnup, the isotopic predictions are all within 2% of those measured. There are ongoing efforts, some of which are documented in [5.12] to obtain more data for burnups above 45 GWd/MTU.

There are cross-section data on about 1600 isotopes in the cross section libraries available for SAS2H. Only about 20 isotopes are primary concern when dealing with high burnup spent fuel [5.8, 5.10]. According to Reference [5.10], 95% of the decay heat is dominated by fewer than 10 nuclides for LWR assemblies at 5 years of cooling. Eight-five percent of the decay heat would be contributed by only 4 isotopes after 100 years.

Applicability of SAS2H for prediction of isotopic content in BWR assemblies was analyzed in [5.8]. A UO_2 sample was burned to 57 GWd/MTU in a BWR reactor. The sample ^{235}U enrichment was 4.97 wt%. Also, the isotopic content of the discharged sample was measured experimentally. Measured content was reported for actinides and fission products. Among concentrations of 16 nuclides investigated, 5 agreed with the measured values to within ±5%.

used to calculate the minimum required cooling time to the nearest 0.1 year as a function of fuel assembly initial enrichment and burnup for each decay heat limit. These cooling times are rounded up to the nearest 0.5 year increment in the final fuel qualification tables. Because the decay heat generally increases slightly with decreasing enrichment for a given burnup, it is conservative to assume that the required cooling time for a higher enrichment assembly is the same as that for a lower enrichment assembly with the same burnup. The required cooling time for initial enrichments that fall between any two SAS2H runs are assumed to be that of the lower enrichment case results.

Parameters that influence the source term calculations are fuel assembly specific power (expressed in MW/fuel assembly (MW/FA)) and the total time between cycles. Other depletion parameters like cycle length and number of cycles are derived from the target burnup, MTU loading and specific power. The most important parameter for the calculation of source terms is the specific power. Specific power for typical US-BWR fuel assemblies are ≤ 5 MW/FA. The source terms for this evaluation are calculated using a specific power that ranges from 7 MW/FA (at lower burnups) to 14 MW/FA (at higher burnups) and results in a conservative estimation of the source terms. The time between cycles utilized is 73 days and represents a typical downtime for US BWRs (60 to 90 days).

The design basis source terms for shielding evaluation are calculated using a conservatism of approximately 20% in the total decay heat load per DSC. The uncertainty associated with SAS2H/ORIGEN-S methodology for high burnup fuel is completely covered using this conservatism.

The design basis source terms are defined as the burnup/initial enrichment/cooling time combination given in the fuel qualification tables that result in the maximum dose rate on the surface of the HSM or TC. Note that for a given DSC design, the design basis HSM source will not necessarily be the same as the corresponding design basis TC source. For the HSM, the roof can be selected as the dose location, and for the TC the cask side is selected as the dose location. This approach is consistent with the method used to determine the fuel qualification tables for the NUHOMS® *61BTH* described in Appendix *T*.5. The radiological source terms generated in the SAS2H/ORIGEN-S runs are used to calculate the surface dose rates in the MCNP models for the locations of interest.

A sample SAS2H/ORIGEN-S input file is listed in Section Y.5.6.1.

Y.5.2.1 Gamma Source Term for MCNP Models

Y.5.2.1.1 Design Basis Gamma Fuel Assembly Source Terms

Once the design basis burnup/enrichment/cooling time combinations have been determined for each shielding configuration of interest, four SAS2H/ORIGEN-S runs are required for each combination to determine gamma source terms for the four fuel assembly regions (i.e., bottom, *active fuel*, plenum and top). The only difference between the runs is in Block #10 "Light Elements" of the SAS2H input and the 82$$ card in the ORIGEN-S input. Each run includes the appropriate Light Elements for the region being evaluated and the 82$$ card is adjusted to have ORIGEN-S output the total gamma source for the *active fuel* region and only the light element source for the plenum, bottom, and top regions. Gamma source terms for the *active fuel* region include contributions from actinides, fission products, and activation products. The bottom, plenum and top nozzle regions include the contribution from the activation products in the specified region only. The SAS2H/ORIGEN-S gamma ray source is output in the CASK-81 energy group structure.

The bounding shielding configuration corresponds to an artificial heat zoning configuration. It contains 25 FAs in zones 1 through 3 at 0.49 kW/FA and 44 FAs in zones 4 and 5 at 0.70 kW/FA. The total load is *43.05* kW/DSC, greater than what is allowed. Note, for a given decay heat, the design basis TC and HSM source may or may not be the same. Furthermore, the use of multiple heat zone loading patterns could also result in the design basis source for the TC side and ends being different because multiple heat zone loading patterns are used. Sources are presented in Tables Y.5-9 through Y.5-13.

Design basis source terms used for the shielding analysis of the 69BTH design basis DSC are shown in Tables Y.5-9 through Y.5-13. Bounding radiological sources for the shielding analysis of the 69BTH design basis DSC in a transfer cask are also shown in Tables Y.5-12 and Table Y.5-13. Design basis source terms used for the shielding analysis of the 69BTH design basis DSC in HSM-H are shown in Table Y.5-9, Table Y.5-10 and Table Y.5-11.

Y.5.2.1.2 Uncertainty in Gamma Source Terms

Almost 100% of the gamma spectrum from light elements is in the range of 0.70 to 1.33 MeV which corresponds exactly to the two most prominent lines of ^{60}Co. As for fission products, the main contributors after six years with a fraction greater then 5% in the range of 0.01 to 0.90 MeV are: ^{90}Sr, ^{90}Y, ^{106}Rh, ^{137}Cs, ^{144}Pr, ^{154}Eu, and ^{155}Eu. Contributions from ^{90}Y, ^{106}Rh, ^{137}Cs, ^{144}Pr, and ^{154}Eu are dominant in the range of 0.90 to 1.50 MeV. ^{106}Rh, ^{147}Sm, and ^{142}Ce are the strongest emitters at energies greater then 2.0 MeV. The accuracy of the gamma spectrum is dependent upon the energy. Photon rates computed for fission products tend to be more accurate than those for actinides because the calculation of their inventory has less uncertainty [5.1].

Shortly after discharge the emission at higher energies is dominated by actinides. This is true for energies >4 MeV at all cooling times and energy above 3.5 MeV for cooling times greater than 10 years [5.1]. The major part of this emission comes from ^{244}Cm. Thus the uncertainty for energy groups of order 3.0 MeV and greater is bounded with the precision with which the inventory of ^{244}Cm is calculated. Per SCALE 4.4 [5.1], reported experimental ^{244}Cm densities are

accurate within ± 20%. The gamma emission intensity from ^{244}Cm, which is proportional to the quantity of ^{244}Cm in the actinide inventory, is bounded by this value. Uncertainty in the source strength in the gamma energy range 0.5 to 2.5 MeV is in the vicinity of 10 to 15% [5.1].

Y.5.2.2 Neutron Source Term for MCNP Models

One SAS2H/ORIGEN-S run is required for each burnup/initial enrichment/cooling time combination to determine the total neutron source term for the *active fuel* regions. At discharge the neutron source is almost equally produced from ^{242}Cm and ^{244}Cm. The other strong contributor is ^{252}Cf, which is approximately 10% of the Cm intensity, but its share vanishes after 6 years of cooling time because the half-life of ^{252}Cf is 2.65 years. The half-lives of ^{242}Cm and ^{244}Cm are 163 days and 18 years, respectively. Contributions from the next strongest emitters, ^{238}Pu and ^{240}Pu, are lower by a factor of 1000 and 100, respectively, relative to ^{244}Cm. For the ranges of exposures, enrichments, and cooling times in the fuel qualification tables, ^{244}Cm represents more than 90% of the total neutron source. The neutron spectrum is, therefore, relatively constant for the fuel parameters addressed herein.

The magnitude of the neutron source is provided as the final row in the gamma source term tables; see Tables Y.5-9 through Y.5-13. Neutron source terms for use in the MCNP shielding models are calculated by multiplying the fuel assembly source by the number of assemblies in the *active fuel* region and summing the terms from all the radial zones. The magnitude of the neutron source is also increased to account for the axial distribution in the fuel, as explained in Appendix Y.5.2.3.

The fixed source spectrum in MCNP is assumed to follow a ^{244}Cm spontaneous fission spectrum for all of the calculations in Appendix Y.5. It is based on the following relationship:

$$f(E) \propto \exp\left[\left(\frac{-E}{a}\right)\sinh(bE)^{1/2}\right] \tag{5.1}$$

where input parameters a = 0.906 MeV and b = 3.848 $(\mathrm{MeV})^{-1}$, as given in the MCNP manual [5.2, 5.3].

Y.5.2.3 Axial Peaking

The axial peaking factors for both neutron and gamma sources in BWR fuel are provided in Appendix K, Section K.5.2.3. The same peaking factors are used in the MCNP analysis presented herein. The peaking factors for both neutron and gamma sources as a function of active fuel height are listed in Appendix K, Table K.5-14. These factors are directly applied to MCNP source input for the fuel region.

These factors in Appendix K, Table K.5-14 are determined based on typical axial burnup distributions for BWR assemblies and based on typical axial water density distribution that occurs during core operation. Using the base SAS2H/ORIGEN-S input for the 7x7 BWR, selected as the design basis assembly for this application, neutron and gamma source terms are generated for axial zones as a function of burnup and moderator density. This estimates both the non-linear behavior of the neutron source with burnup and the core operating moderator density

Y.5.3 Material Densities

The material weights given in Table Y.5-5 for the fuel are used to calculate material densities for *active fuel*, plenum, top, and bottom regions of the fuel assembly.

In order to account for sub-critical multiplication, an initial enrichment of 4.0 wt% U-235 is used to calculate the amount of U-235 in the shielding models.

Material densities used in the various MCNP models are summarized in Table Y.5-8.

Y.5.4 Shielding Evaluation

Dose rate contributions from the bottom, *active fuel*, plenum and top regions, as appropriate, from 69 BWR fuel assemblies loaded into NUHOMS® -69BTH DSC are calculated with the MCNP 5 v1.40 [5.3] code at various locations on and around the transfer casks (TC) and horizontal storage modules (HSM), respectively. The following evaluation specifically addresses the NUHOMS®-69BTH DSC in HSM-H or OS200 TC using the design basis source terms determined in Section Y.5.2.

Y.5.4.1 Computer Program

MCNP 5 [5.3] is a general-purpose Monte Carlo N-Particle codes that can be used for neutron, photon, electron, or coupled neutron/photon/electron transport. The code treats an arbitrary three-dimensional configuration of materials in geometric cells bounded by first- and second-degree surfaces and some special fourth-degree surfaces. Pointwise (continuous energy) cross-section data are used. For neutrons, all reactions given in a particular cross-section evaluation are accounted for in the cross section set. For photons, the code takes account of incoherent and coherent scattering, the possibility of fluorescent emission after photoelectric absorption, absorption in pair production with local emission of annihilation radiation, and bremsstrahlung. Important standard features that make MCNP 5 very versatile and easy to use include a powerful general source; an extensive collection of cross-section data; and an extensive collection of variance reduction techniques that can be employed to track particles through very complex deep penetration problems. MCNP 5 was employed to take advantage of its mesh tallies capabilities in calculating dose rates distributed over the surface of the HSM. It also allows more point detectors to be used in a single run that substantially reduces the number of input/out decks needed to perform ISFSI site dose rate calculations described in Appendix Y.10.

Y.5.4.2 Spatial Source Distribution

The source components are:

- the neutron sources due to the active fuel region,
- the gamma source due to the active fuel region,
- the gamma source due to the plenum,
- the gamma source due to the top region, and
- the gamma source due to the bottom region.

Axial peaking is accounted for in the active fuel region by inputting an axial shape, as discussed in Section Y.5.2.3.

Y.5.4.3 Cross Section Data

The cross-section data used is the continuous energy ENDF/B-VI provided with the MCNP code. The cross-section data allows coupled neutron/gamma-ray dose rate evaluation to be made to account for the contributions from secondary gamma radiation (n,γ), if desired. All of the transfer cask dose rate calculations account for the dose rate due to secondary gamma radiation. For the HSM-H dose rate calculation, the dose rate contribution from the secondary gamma

Table Y.5-5
BWR Fuel Assembly Material Mass

Hardware Item	**Material**	**Average Mass, (kg/FA)**	**Comments**
***Active Fuel* Zone, (144.00** inch long, **4.73** g/FA total cobalt content)			
Cladding	Zircaloy-2	49.2	
Fuel channel sleeve	Zircaloy-4	37.1	
Grid spacers	Zircaloy-4	1.95	7 spacers*~0.28 kg/spacer
Spacer springs	Inconel X-750	0.36	7 springs*0.051 kg/spring
Channel spring & bolt	Inconel X-750	0.13	
Channel fastener guard	stainless steel	0.46	
Channel spacer & rivet	stainless steel	0.13	
Fuel	uranium	198	wt. of UO_2=224.643 kg.=0.198 mtu/0.8814
Gas Plenum Zone, (12.93 inch long, **0.89** g/FA total cobalt content)			
Cladding	Zircaloy-2	4.89	
Fuel channel	Zircaloy-4	0.00	
Plenum springs	stainless steel	1.05	
Top End Fitting Zone, (12.62 inch long, **4.51** g/FA total cobalt content)			
Upper tie plate	stainless steel	2.08	
Lock tab washers & nuts	stainless steel	0.05	
Expansion springs	Inconel X-750	0.43	
End plugs	zircaloy	1.26	
Bottom End Fitting Zone, (6.65 inch long, **4.10** g/FA total cobalt content)			
Finger springs	Inconel	0.05	
End plugs	zircaloy	1.26	
Lower tie plate	stainless steel	4.7	
Total, kg.[1]		329.7	
Total, lb.[1]		726.3	

Note: This mass is conservative for the source term calculation because the maximum weight of fuel assembly with or without channel is limited to 705 lb per Appendix T.2.

Table Y.5-7
Flux Scaling Factors by Fuel Assembly Region

Fuel Assembly Region	Flux Factor
Bottom	0.15
Active Fuel	1.00
Plenum	0.20
Top	0.10

Table Y.5-9
Bounding Radiological Source Term for the HSM-HS–Zone 1 through Zone 3

*Representative Radiological Source, γ/(sec*FA): 62 GWD/MTU, 2.6 wt%, after 14.0 years of cooling. Generates 0.49 kWt/FA of decay heat.*						
E_{min}, MeV	to	E_{max}, MeV	Bottom Nozzle	*Active Fuel*	Plenum	Top Nozzle
0.00e+00	to	5.00e-02	5.85E+10	6.044E+14	2.935E+10	4.767E+10
5.00e-02	to	1.00e-01	7.12E+09	1.142E+14	2.398E+09	5.838E+09
1.00e-01	to	2.00e-01	1.90E+09	8.155E+13	1.533E+09	1.531E+09
2.00e-01	to	3.00e-01	9.62E+07	2.397E+13	8.596E+07	7.729E+07
3.00e-01	to	4.00e-01	1.56E+08	1.535E+13	2.660E+08	1.210E+08
4.00e-01	to	6.00e-01	9.89E+08	4.264E+13	5.107E+09	6.601E+08
6.00e-01	to	8.00e-01	5.27E+08	9.044E+14	2.644E+09	4.041E+08
8.00e-01	to	1.00e+00	1.57E+08	2.447E+13	4.519E+07	1.520E+08
1.00e+00	to	1.33e+00	2.07E+12	3.475E+13	6.823E+11	1.702E+12
1.33e+00	to	1.66e+00	5.86E+11	6.380E+12	1.927E+11	4.806E+11
1.66e+00	to	2.00e+00	1.10E+01	4.131E+10	5.706E+01	7.352E+00
2.00e+00	to	2.50e+00	1.39E+07	3.939E+09	4.572E+06	1.141E+07
2.50e+00	to	3.00e+00	2.16E+04	3.890E+08	7.090E+03	1.769E+04
3.00e+00	to	4.00e+00	1.39E-11	8.711E+07	6.975E-15	7.607E-11
4.00e+00	to	5.00e+00	0.00E+00	2.193E+07	1.401E-45	0.000E+00
5.00e+00	to	6.50e+00	0.00E+00	8.800E+06	0.000E+00	0.000E+00
6.50e+00	to	8.00e+00	0.00E+00	1.726E+06	0.000E+00	0.000E+00
8.00e+00	to	1.00e+01	0.00E+00	3.666E+05	0.000E+00	0.000E+00
Total gamma, g/(sec*FA)			2.729E+12	1.852E+15	9.163E+11	2.239E+12
Total neutrons, n/(sec*FA)			6.370E+08			

Table Y.5-10
Bounding Radiological Source Term for the HSM-HS–Zone 4

*Representative Radiological Source, γ/(sec*FA): 62 GWD/MTU, 2.6 wt%, after 7.1 years of cooling. Generates 0.70 kWt/FA of decay heat.*						
E_{min}, MeV	**to**	**E_{max}, MeV**	**Bottom Nozzle**	*Active Fuel*	**Plenum**	**Top Nozzle**
0.00e+00	to	5.00e-02	1.520E+11	8.731E+14	1.124E+11	1.223E+11
5.00e-02	to	1.00e-01	1.768E+10	1.652E+14	6.150E+09	1.450E+10
1.00e-01	to	2.00e-01	5.325E+09	1.286E+14	6.992E+09	4.203E+09
2.00e-01	to	3.00e-01	2.753E+08	3.661E+13	4.037E+08	2.160E+08
3.00e-01	to	4.00e-01	5.312E+08	2.405E+13	1.414E+09	3.966E+08
4.00e-01	to	6.00e-01	5.681E+09	3.320E+14	2.946E+10	3.788E+09
6.00e-01	to	8.00e-01	2.954E+09	1.353E+15	1.525E+10	2.022E+09
8.00e-01	to	1.00e+00	1.346E+10	1.537E+14	4.065E+09	4.319E+09
1.00e+00	to	1.33e+00	5.143E+12	8.113E+13	1.692E+12	4.219E+12
1.33e+00	to	1.66e+00	1.452E+12	2.223E+13	4.777E+11	1.191E+12
1.66e+00	to	2.00e+00	1.324E+01	2.674E+11	6.771E+01	8.831E+00
2.00e+00	to	2.50e+00	3.446E+07	3.480E+11	1.134E+07	2.827E+07
2.50e+00	to	3.00e+00	5.344E+04	1.958E+10	1.758E+04	4.384E+04
3.00e+00	to	4.00e+00	1.603E-11	2.526E+09	8.974E-15	8.800E-11
4.00e+00	to	5.00e+00	0.000E+00	2.925E+07	1.401E-45	0.000E+00
5.00e+00	to	6.50e+00	0.000E+00	1.174E+07	0.000E+00	0.000E+00
6.50e+00	to	8.00e+00	0.000E+00	2.303E+06	0.000E+00	0.000E+00
8.00e+00	to	1.00e+01	0.000E+00	4.890E+05	0.000E+00	0.000E+00
Total gamma, g/(sec*FA)			6.793E+12	3.170E+15	2.346E+12	5.562E+12
Total neutrons, n/(sec*FA)			8.580E+08			

Table Y.5-11
Bounding Radiological Source Term for the HSM-HS-Zone 5

*Representative Radiological Source, γ/(sec*FA): 31 GWD/MTU, 0.9 wt%, after 3.1 years of cooling. Generates 0.70 kWt/FA of decay heat.*						
E_{min}, MeV	**to**	**E_{max}, MeV**	**Bottom Nozzle**	***Active Fuel***	**Plenum**	**Top Nozzle**
0.00e+00	to	5.00e-02	1.995E+11	1.440E+15	2.205E+11	1.542E+11
5.00e-02	to	1.00e-01	2.020E+10	3.223E+14	7.295E+09	1.665E+10
1.00e-01	to	2.00e-01	6.493E+09	2.816E+14	1.018E+10	5.096E+09
2.00e-01	to	3.00e-01	3.413E+08	7.962E+13	6.061E+08	2.657E+08
3.00e-01	to	4.00e-01	7.053E+08	6.103E+13	2.131E+09	5.200E+08
4.00e-01	to	6.00e-01	8.702E+09	6.056E+14	4.509E+10	5.801E+09
6.00e-01	to	8.00e-01	4.538E+09	1.098E+15	2.351E+10	3.058E+09
8.00e-01	to	1.00e+00	1.761E+11	2.115E+14	5.315E+10	5.438E+10
1.00e+00	to	1.33e+00	5.862E+12	9.727E+13	1.941E+12	4.837E+12
1.33e+00	to	1.66e+00	1.655E+12	3.240E+13	5.482E+11	1.366E+12
1.66e+00	to	2.00e+00	1.974E+05	1.948E+12	6.288E+04	1.443E+05
2.00e+00	to	2.50e+00	3.929E+07	4.574E+12	1.301E+07	3.242E+07
2.50e+00	to	3.00e+00	6.092E+04	1.598E+11	2.017E+04	5.027E+04
3.00e+00	to	4.00e+00	3.469E-12	1.995E+10	1.021E-15	1.906E-11
4.00e+00	to	5.00e+00	0.000E+00	8.637E+06	0.000E+00	0.000E+00
5.00e+00	to	6.50e+00	0.000E+00	3.467E+06	0.000E+00	0.000E+00
6.50e+00	to	8.00e+00	0.000E+00	6.801E+05	0.000E+00	0.000E+00
8.00e+00	to	1.00e+01	0.000E+00	1.444E+05	0.000E+00	0.000E+00
Total gamma, g/(sec*FA)			7.934E+12	4.236E+15	2.852E+12	6.443E+12
Total neutrons, n/(sec*FA)			2.503E+08			

Table Y.5-12
Bounding Radiological Source Term for the OS200–Zone 1 through Zone 3

*Representative Radiological Source, γ/(sec*FA): 62 GWD/MTU, 2.6 wt%, after 14.0 years of cooling. Generates 0.490 kWt/FA of decay heat.*						
E_{min}, MeV	to	E_{max}, MeV	Bottom Nozzle	*Active Fuel*	Plenum	Top Nozzle
0.00e+00	to	5.00e-02	5.85E+10	6.044E+14	2.935E+10	4.767E+10
5.00e-02	to	1.00e-01	7.12E+09	1.142E+14	2.398E+09	5.838E+09
1.00e-01	to	2.00e-01	1.90E+09	8.155E+13	1.533E+09	1.531E+09
2.00e-01	to	3.00e-01	9.62E+07	2.397E+13	8.596E+07	7.729E+07
3.00e-01	to	4.00e-01	1.56E+08	1.535E+13	2.660E+08	1.210E+08
4.00e-01	to	6.00e-01	9.89E+08	4.264E+13	5.107E+09	6.601E+08
6.00e-01	to	8.00e-01	5.27E+08	9.044E+14	2.644E+09	4.041E+08
8.00e-01	to	1.00e+00	1.57E+08	2.447E+13	4.519E+07	1.520E+08
1.00e+00	to	1.33e+00	2.07E+12	3.475E+13	6.823E+11	1.702E+12
1.33e+00	to	1.66e+00	5.86E+11	6.380E+12	1.927E+11	4.806E+11
1.66e+00	to	2.00e+00	1.10E+01	4.131E+10	5.706E+01	7.352E+00
2.00e+00	to	2.50e+00	1.39E+07	3.939E+09	4.572E+06	1.141E+07
2.50e+00	to	3.00e+00	2.16E+04	3.890E+08	7.090E+03	1.769E+04
3.00e+00	to	4.00e+00	1.39E-11	8.711E+07	6.975E-15	7.607E-11
4.00e+00	to	5.00e+00	0.00E+00	2.193E+07	1.401E-45	0.000E+00
5.00e+00	to	6.50e+00	0.00E+00	8.800E+06	0.000E+00	0.000E+00
6.50e+00	to	8.00e+00	0.00E+00	1.726E+06	0.000E+00	0.000E+00
8.00e+00	to	1.00e+01	0.00E+00	3.666E+05	0.000E+00	0.000E+00
Total gamma, g/(sec*FA)			2.729E+12	1.852E+15	9.163E+11	2.239E+12
Total neutrons, n/(sec*FA)			6.370E+08			

Table Y.5-13
Bounding Radiological Source Term for the OS200–Zone 4 and 5

*Representative Radiological Source, γ/(sec*FA): 62 GWD/MTU, 2.6 wt%, after 7.1 years of cooling. Generates 0.700 kWt/FA of decay heat.*						
E_{min}, MeV	to	E_{max}, MeV	Bottom Nozzle	*Active Fuel*	Plenum	Top Nozzle
0.00e+00	to	5.00e-02	1.520E+11	8.731E+14	1.124E+11	1.223E+11
5.00e-02	to	1.00e-01	1.768E+10	1.652E+14	6.150E+09	1.450E+10
1.00e-01	to	2.00e-01	5.325E+09	1.286E+14	6.992E+09	4.203E+09
2.00e-01	to	3.00e-01	2.753E+08	3.661E+13	4.037E+08	2.160E+08
3.00e-01	to	4.00e-01	5.312E+08	2.405E+13	1.414E+09	3.966E+08
4.00e-01	to	6.00e-01	5.681E+09	3.320E+14	2.946E+10	3.788E+09
6.00e-01	to	8.00e-01	2.954E+09	1.353E+15	1.525E+10	2.022E+09
8.00e-01	to	1.00e+00	1.346E+10	1.537E+14	4.065E+09	4.319E+09
1.00e+00	to	1.33e+00	5.143E+12	8.113E+13	1.692E+12	4.219E+12
1.33e+00	to	1.66e+00	1.452E+12	2.223E+13	4.777E+11	1.191E+12
1.66e+00	to	2.00e+00	1.324E+01	2.674E+11	6.771E+01	8.831E+00
2.00e+00	to	2.50e+00	3.446E+07	3.480E+11	1.134E+07	2.827E+07
2.50e+00	to	3.00e+00	5.344E+04	1.958E+10	1.758E+04	4.384E+04
3.00e+00	to	4.00e+00	1.603E-11	2.526E+09	8.974E-15	8.800E-11
4.00e+00	to	5.00e+00	0.000E+00	2.925E+07	1.401E-45	0.000E+00
5.00e+00	to	6.50e+00	0.000E+00	1.174E+07	0.000E+00	0.000E+00
6.50e+00	to	8.00e+00	0.000E+00	2.303E+06	0.000E+00	0.000E+00
8.00e+00	to	1.00e+01	0.000E+00	4.890E+05	0.000E+00	0.000E+00
Total gamma, g/(sec*FA)			6.793E+12	3.170E+15	2.346E+12	5.562E+12
Total neutrons, n/(sec*FA)			8.580E+08			

Y.9.1.3 Leak Tests

The DSC canister confinement boundary is tested using two procedures described below. Personnel performing the leakage test are qualified in accordance with SNT-TC-1A [9.2].

Procedure 1 is accomplished during fabrication:

Upon completion of all canister shell welding and attachment of the inner bottom cover plate to the shell, a temporary seal plate is placed over the open end of the DSC. A bag is placed around the outside of the entire DSC and it is filled with helium. The DSC cavity is evacuated and a helium leakage test is performed using a port in the seal plate. This test is used to show that the entire DSC confinement boundary tested is leak tight (1x10-7 ref cm3/s).

Procedure 2 of the testing occurs after the DSC has been loaded with fuel assemblies:

The DSC cavity has been dried, back filled with helium and the inner top cover plate and the vent and drain port cover plates have been welded in place. After these welds are completed, a temporary test cover is installed or the outer top cover plate is welded in place. The cavity between inner and outer top cover plates is evacuated and a helium leakage test is performed using the test port in the outer top cover plate. The leakage test thus includes the weld attaching the inner top cover plate to the canister shell, the vent and drain port cover plate welds and the base metal of the inner top cover plate and port cover plates. The ports are filled with helium prior to welding the port covers. This test verifies that the tested welds and covers are leak tight (1x10-7 ref cm3/s).

Y.9.1.4 Components

The Standardized NUHOMS® system does not include any components such as valves, rupture discs, pumps, or blowers. The gaskets in the Transfer Cask do not require acceptance testing other than the leak testing cited above. No other components of the NUHOMS® system require testing, except as discussed in this chapter.

Y.9.1.5 Shielding Integrity

The Transfer Cask poured lead shielding integrity will be confirmed via gamma scanning prior to first use. The detector and examination grid will be matched to provide coverage of the entire lead-shielded surface area. For example, for a 6" × 6" grid, the detector will encompass a 6" × 6" square. The acceptance criterion is attenuation greater than or equal to that of a test block matching the cask through-wall configuration with lead and steel thicknesses equal to the design minima less 5%.

The radial neutron shielding is provided by filling the neutron shield shell with water during operations. No testing is necessary. The neutron shield material in the lid and bottom end is a proprietary polymer resin. The shielding performance of the resin will be assured by written procedures controlling temperature, measuring, and mixing of the components, degassing of the resin, and verification of the mass or volume of resin installed.

The gamma and neutron shielding materials of the storage system itself are limited to concrete HSM components and steel shield plugs in the DSC. The integrity of these shielding materials is ensured by the control of their fabrication in accordance with the appropriate ASME, ASTM or ACI criteria. No additional acceptance testing is required.

Y.9.1.6 Thermal Acceptance

No thermal acceptance testing is required to verify the performance of each storage unit other than that specified in the Technical Specifications for initial loading.

The heat transfer analysis for the basket includes credit for the thermal conductivity of neutron-absorbing materials, as specified in Section Y.4.3. Because these materials do not have publicly documented values for thermal conductivity, testing of such materials will be performed in accordance with Section Y.9.1.7.6.

Y.11.2.10.4 Corrective Actions

Evaluation of HSM-H or TC neutron shield damage as a result of a fire is to be performed to assess the need for temporary shielding (for HSM-H or cask, if fire occurs during transfer operations) and repairs to restore the TC and HSM-H to pre-fire design conditions.

Y.11.2.11 <u>Accident Temperatures</u>

For this accident condition, very high ambient temperature of 132 °F is postulated.

Y.11.2.11.1 Cause of Accident

At some locations, there is a possibility of very high ambient temperatures, higher than the off-normal conditions used in Section Y.11.1. Therefore, to envelope these high temperatures, a 132 °F ambient temperature is selected for this accident evaluation.

Y.11.2.11.2 Accident Analysis

The analysis of the 69BTH DSC in HSM-H with 132 °F ambient temperatures is bounded by the blocked vent accident analysis documented in Section Y.11.2.7 because the inlet and outlet vents are functioning normally for this accident. Similarly, the analysis of the 69BTH DSC in OS200 TC is bounded by the accidental cask drop analysis documented in Section Y.11.2.5 because the neutron shield in the TC is functioning normally. The 69BTH DSC temperatures and internal pressures, HSM-H and TC temperatures and hence stresses are all bounded by the blocked vent analysis or the accidental cask drop analysis. Therefore, the NUHOMS® 69BTH system will withstand these very high ambient temperatures without breach of the confinement boundary.

Y.11.2.11.3 Accident Dose Calculations

There are no radiation dose consequences for this accident. The NUHOMS® 69BTH DSC is designed and tested as a leak-tight containment boundary. Very high ambient temperature of 132 °F does not breach the containment boundary. Therefore radioactive material inside the DSC will remain sealed in the DSC and, therefore, will not contaminate the environment.

Y.11.2.11.4 Corrective Actions

None required.

Z.2.1 Spent Fuel to be Stored

Z.2.1.1 Intact or Damaged UO_2 Fuel

As described in Appendix Z.1, there are two alternate design configurations for the NUHOMS®-37PTH DSC depending on the canister length: a short (182.00") DSC designated as 37PTH-S, and a medium (189.25") DSC designated as 37PTH-M. Each of the DSC configurations is designed to store intact (including reconstituted) and/or damaged PWR fuel assemblies as specified in Table Z.2-1 and Table Z.2-3. The fuel to be stored is limited to a maximum assembly average initial enrichment of 5.0 wt. % U-235. The maximum allowable assembly average burnup is limited to 62 GWd/MTU and the minimum cooling time is 3 years. Each of the DSC types is designed to store control components (CCs) with thermal and radiological characteristics as listed in Table Z.2-2. The CCs include burnable poison rod assemblies (BPRAs), neutron source assemblies (NSAs), thimble plug assemblies (TPAs), control rod assemblies (CRAs), rod cluster control assemblies (RCCAs), axial power shaping rod assemblies (APSRAs), orifice rod assemblies (ORAs), vibration suppression inserts (VSIs), and neutron sources. *Non-fuel hardware that are positioned within the fuel assembly after the fuel assembly is discharged from the core such as Guide Tube or Instrument Tube Tie Rods or Anchors, Guide Tube Inserts, BPRA Spacer Plates or devices that are positioned and operated within the fuel assembly during reactor operation such as those listed above are also considered as CCs.*

The NUHOMS®-37PTH DSC is also authorized to store fuel assemblies containing Blended Low Enriched Uranium (BLEU) fuel material. Fuel pellets containing BLEU fuel material are no different than UO_2 fuel pellets except for the presence of a higher quantity of cobalt impurity. The consideration of cobalt impurity affects only the gamma source terms for fuel assemblies located in the DSC periphery. This does not affect any criticality, thermal or structural analysis inputs for evaluation of fuel assemblies with BLEU material. The qualification of fuel assemblies containing BLEU fuel pellets will require an additional cooling time of three years to ensure that the source terms calculated with UO_2 material are bounding.

Fuel assemblies that contain fixed integral non-fuel rods are also considered as intact fuel assemblies. These fuel assemblies are different than reconstituted assemblies because fuel rods are not "replaced" by non-fuel rods, rather the non-fuel rods are part of the initial fuel design. The non-fuel rods displace the same amount of moderator, with zirconium-alloy (or aluminum) cladding and typically contain burnable absorber (or other non-fuel) material. The radiation and thermal source terms for the non-fuel rods are significantly lower than those of the fuel rods since there is no significant radioactive decay source. The internal pressure of the non-fuel rods after irradiation is lower than those of the fuel rods since there is no fission gas generation. The reactivity of the fuel rods (from a criticality standpoint) is significantly higher than that of non-fuel rods. In summary, the mechanical, thermal, shielding and criticality evaluations for these rods are bounded by those of the regular rods. Therefore, no further evaluations are required for the qualification of these fuel assemblies.

Reconstituted assemblies containing up to 10 replacement irradiated stainless steel rods per assembly or unlimited lower enrichment UO_2 rods instead of zircaloy clad enriched UO_2 rods, or Zr rods or Zr pellets, or unirradiated stainless steel rods are acceptable for storage in the 37PTH DSC as intact fuel assemblies with a slightly longer cooling time than that required for a standard assembly. The stainless steel rods are assumed to have two-thirds the irradiation time as the remaining fuel rods of the assembly. The reconstituted UO_2 rods are assumed to have the same

Table Z.2-1
PWR Fuel Specification for the Fuel to be Stored in the NUHOMS®-37PTH DSC (UO_2 Fuel)

(Part 1 of 2)

PHYSICAL PARAMETERS: Fuel Class	Intact or damaged unconsolidated WE 17x17, CE 16X16, CE 15x15, WE 15x15, CE 14x14, and WE 14x14 class PWR assemblies (with or without control components) that are enveloped by the fuel assembly design characteristics listed in Table Z.2-3. Reload fuel manufactured by other vendors but enveloped by the design characteristics listed in Table Z.2-3 is also acceptable. Damaged fuel assemblies beyond the definition contained below are not authorized for storage.
Fuel Damage	Damaged PWR fuel assemblies are assemblies containing missing or partial fuel rods or fuel rods with known or suspected cladding defects greater than hairline cracks or pinhole leaks. The extent of damage in the fuel assembly is to be limited such that a fuel assembly is being able to be handled by normal means. Missing fuel rods are allowed. Damaged fuel assemblies shall also contain top and bottom end fittings or nozzles or tie plates depending on the fuel type.
Reconstituted Fuel Assemblies:	
• Maximum No. of Reconstituted Assemblies per DSC with Irradiated Stainless Steel Rods	4
• Maximum No. of Irradiated Stainless Steel Rods per Reconstituted Fuel Assembly	10
• Maximum No. of Reconstituted Assemblies per DSC with Unlimited Number of Low Enriched UO_2 Rods, or Zr Rods or Zr Pellets or Unirradiated Stainless Steel Rods	37
Control Components (CCs)	• Up to 37 CCs are authorized for storage in 37PTH-S, and 37PTH-M DSCs. • Authorized CCs include Burnable Poison Rod Assemblies (BPRAs), Thimble Plug Assemblies (TPAs), Control Rod Assemblies (CRAs), Rod Cluster Control Assemblies (RCCAs), Axial Power Shaping Rod Assemblies (APSRAs), Orifice Rod Assemblies (ORAs), Neutron Source Assemblies (NSAs), Vibration Suppression Inserts (VSIs) and Neutron Sources. *Non-fuel hardware that are positioned within the fuel assembly after the fuel assembly is discharged from the core such as Guide Tube or Instrument Tube Tie Rods or Anchors, Guide Tube Inserts, BPRA Spacer Plates or devices that are positioned and operated within the fuel assembly during reactor operation such as those listed above are also considered as CCs.* • Design basis thermal and radiological characteristics for the CCs are listed in Table Z.2-2.

5. Aluminum, Type 6061 [4.3]

Temperature (°F)	k (Btu/hr-in-°F)	ρ (lb_m/in^3)	C_p (Btu/lb_m-°F)
70	8.008	0.098	0.213
100	8.075		0.215
150	8.167		0.218
200	8.250		0.221
250	8.317		0.223
300	8.383		0.226
350	8.442		0.228
400	8.492		0.230

6. Neutron Absorber Thermal Conductivity (See Section Z.4.3)

7. Helium [4.4]

Temperature (K)	Thermal conductivity (W/m-K)	Temperature (°F)	Thermal conductivity (Btu/hr-in-°F)
300	0.1499	80	0.0072
400	0.1795	260	0.0086
500	0.2115	440	0.0102
600	0.2466	620	0.0119
800	0.3073	980	0.0148
1000	0.3622	1340	0.0174
1050	0.3757	1430	0.0181

The above data are calculated based on the following polynomial function from [4.4].
$k = \sum C_i \, T^i$ for conductivity in (W/m-K) and T in (K)

For 300 < T < 500 K		For 500< T < 1050 K	
C0	-7.761491E-03	C0	-9.0656E-02
C1	8.66192033E-04	C1	9.37593087E-04
C2	-1.5559338E-06	C2	-9.13347535E-07
C3	1.40150565E-09	C3	5.55037072E-10
C4	0.0E+00	C4	-1.26457196E-13

No density or specific heat is considered for helium for conservatism.

8. Air [4.4]

Temperature (K)	Thermal conductivity (W/m-K)	Temperature (°F)	Thermal conductivity (Btu/hr-in-°F)
250	0.02228	-10	0.0011
300	0.02607	80	0.0013
400	0.03304	260	0.0016
500	0.03948	440	0.0019
600	0.04557	620	0.0022
800	0.05698	980	0.0027
1000	0.06721	1340	0.0032

The above data are calculated based on the following polynomial function from [4.4].
$k = \sum C_i T^i$ for conductivity in (W/m-K) and T in (K)

For 250 < T < 1050 K	
C0	-2.2765010E-03
C1	1.2598485E-04
C2	-1.4815235E-07
C3	1.7355064E-10
C4	-1.0666570E-13
C5	2.4766304E-17

No density or specific heat is considered for air in the thermal models for conservatism.

9. MOX Fuel Pellet

(See Section Z.4.9.1 for evaluation of irradiated conductivities for MOX fuel pellets)

Temperature (°F)	200	300	400	500	600	700	800
$k_{MOX\ irrd}$ (Btu/hr-in-°F)	0.1201	0.1186	0.1167	0.1143	0.1113	0.1092	0.1067

Z.4.4 Thermal Evaluation of HSM-H/HSM-HS with 37PTH DSC

The HSM-H or HSM-HS modules are used to store a loaded 37PTH DSC at the ISFSI. As described in Appendix U, Section U.1.2.1.2, the HSM-H and HSM-HS modules are thermally identical. Therefore, the thermal evaluation of the HSM-H is also applicable to the HSM-HS.

The design of HSM-H was first described and evaluated in Appendix P for 24PTH DSC with maximum heat load of 40.8 kW. The same design was also evaluated in Appendix U, Section U.4.4 for 32PTH1 DSC for heat loads of 40.8 kW and 31.2 kW. The maximum decay heat load considered for the NUHOMS®-37PTH DSC as described in Section Z.4.1 is 30 kW. Therefore, the thermal performance of HSM-H with 37PTH DSC is bounded by the evaluations presented in Appendix P, Section P.4.4 and Appendix U, Section U.4.4. This condition is justified in the following sections.

Z.4.4.1 Ambient Temperature Specification

Ambient temperatures in the range of 0-100°F are considered as normal storage conditions. A maximum day temperature of 117°F is considered as off-normal, hot storage condition, *which corresponds to a* 24-hour average ambient temperature of 102°F calculated in Appendix M, Section M.4.5. *A 24-hour average ambient temperature of 117°F is conservatively used for evaluation of the off-normal hot storage condition. In addition to the above ambient temperatures, an extreme ambient temperature (24-hour average ambient temperature) of 117 °F corresponding to maximum day temperature of 132°F is considered as an accident condition for storage when 37PTH DSC is loaded within the HSM-H. Based on the above description, the 24-hour average ambient temperatures of 117 °F considered in the analyses for the off-normal hot storage condition is identical to the ambient temperature considered for the extreme hot ambient accident conditions.* The lowest off-normal ambient temperature is considered to be -40°F.

Z.4.4.2 Description of Loading Cases for Storage of 37PTH DSC

The loading cases considered for storage of the 37PTH DSC include the normal, off-normal, and blocked vent accident conditions. All of these conditions were evaluated for the HSM-H loaded with 32PTH1 DSC and presented in Appendix U, Section U.4.4. The thermal evaluations in Section U.4.4 included heat loads of 31.2 kW and 40.8 kW as described in Appendix U, Section U.4.4.2. The differences between the boundary conditions considered for 37PTH DSC and 32PTH1 DSC under storage conditions are highlighted below.

DSC Type	37PTH	32PTH1
Maximum Heat Load (kW)	30	31.2
Normal Ambient Temperature (°F)	100	106
Extreme Ambient Temperature (°F)	*117*	133

In addition to load cases considered in Appendix U, Section U.4.4 for the HSM-H storage module, an additional load case is considered for off-normal storage condition with 50% blockage of the HSM-H inlet vents. This load case is designated as load case S4A and described in Table Z.4-2.

For off-normal storage condition with 50% blockage of the HSM-H inlet vents (load case S4A), the thermal performance of the 37PTH DSC in HSM-H is determined using the same methodology described in Appendix U, Section U.4.4 with the blockage considered to occur over the bottom half of the 30" high inlet vents reducing the area of the inlet by half.

The effective density and specific heat for 37PTH basket are calculated using the same methodology discussed in Appendix U, Section U.4.8.3. A comparison of the effective properties between 37PTH basket and 32PTH1 basket shows that the overall heat capacity of the 32PTH1-S basket (density × specific heat) is about 4% lower than that of the 37PTH-S basket as listed in the following table.

Basket Type	$\rho_{eff,basket}$ **(lbm/in³)**	Cp_{eff} **(Btu/lbm-°F)**	$(\rho \times Cp)_{basket}$ **(Btu/in³-°F)**	$\frac{(\rho \times Cp)_{32PTH1-S}}{(\rho \times Cp)_{37PTH-S}}$
32PTH1-S	0.125 (Section U.4.8.3)	0.095 (Section U.4.8.3)	0.011875	0.96
37PTH-S	0.133	0.093	0.012369	

Based on the above table, the heat up rate for the 37PTH DSC is lower than that for the 32PTH1 DSC for the same heat load.

The inner and outer shell diameters (68.75" and 69.75") and the basket length (162") of the 37PTH DSC are identical to 32PTH1 DSC as used in the thermal analyses. Therefore, the decay heat flux and the decay heat generation rate used in the thermal analyses of HSM-H loaded with 32PTH1 DSC with 31.2 kW heat load are bounding for the 37PTH DSC with 30 kW heat load.

The above comparison shows that the boundary conditions for load cases considered in the thermal analysis of HSM-H with 32PTH1 DSC and 31.2 kW heat load are bounding for the 37PTH DSC. Therefore, the maximum HSM-H component temperatures and the DSC shell temperature profiles resulting from the steady state and transient thermal analyses of the 32PTH1 DSC in the HSM-H with 31.2 kW heat load discussed in Appendix U, Section U.4.4 are bounding for the HSM-H loaded with the 37PTH DSC. These DSC shell temperatures are used as boundary conditions for the 37PTH DSC model to determine the maximum fuel cladding temperatures for storage conditions in the 37PTH DSC conservatively.

Z.4.4.3 HSM-H Thermal Analysis Results

Based on the discussions in Section Z.4.4.2, the DSC shell temperatures from the thermal analyses of HSM-H with 32PTH1 DSC and 31.2 kW heat load result in conservative predictions of the temperatures for 37PTH DSC in HSM-H. To maintain this conservatism, the same time limit of 40 hours is considered for blocked vent accident condition for HSM-H loaded with 37PTH DSC as it was established in Appendix U, Section U.4.4.7 for 32PTH1 DSC. The HSM-H air flow analysis and thermal model were described in Appendix U, Sections U.4.4.3 and U.4.4.4, respectively.

Based on the methodology described in Appendix U, Section U.4.4.3 the exit air temperature is computed assuming a 50% blockage of the HSM-H inlet vents with a 24 hour average ambient temperature of 105°F. The considered average ambient temperature of 105°F is greater than the 24 hour average ambient temperature of 102°F, which corresponds to a maximum daily ambient ***temperature of 117°F as described in Appendix Z, Section Z.4.4.1. Further, the maximum heat load of 30 kW allowed in 37PTH DSC is used instead of the conservative heat load of 31.2 kW considered for other load cases.***

The resulting exit air temperature is 192°F (load case S4A) and is lower than the 199°F reported in Appendix U, Table U.4-1 for 32PTH1 DSC in HSM-H with 31.2 kW for off-normal condition with 24 hour average ambient temperature of 117°F ambient. The exit air temperature for load case S4A is lower due to the removal of the conservatisms described above.

Further, as discussed in Appendix Z, Section Z.4.4.2, the 32PTH1 DSC has a higher heat load and lower heat capacity compared to the 37PTH DSC. This coupled with the higher airflow exit temperatures shows the thermal analysis performed for 37PTH DSC off-normal storage condition (load case S4) with DSC shell temperature profiles from 32PTH1 DSC remains bounding for the off-normal condition with 50% blockage of the HSM-H inlet vents (load case S4A).

As described in the Section Z.4.4.1, a 24-hour average ambient temperature of 117°F is conservatively used for evaluation of the off-normal hot storage condition. This average temperature is applicable to the extreme hot ambient accident storage condition with the same

average ambient temperature of 117°F. Therefore, the thermal analysis performed for the extreme hot ambient accident storage condition (load case S6) is equivalent to the off-normal storage condition (load case S4).

The resultant DSC shell temperatures from the analyses performed in Appendix U, Section U.4.4 for 31.2 kW heat load are used as boundary conditions in the analysis of the 37PTH DSC with 30 kW heat load, which is presented in Section Z.4.6.

The summary of the HSM-H component and DSC shell maximum temperatures achieved under the blocked vent accident conditions for the 32PTH1 DSC with decay heat load of 31.2 kW were listed in Table U.4-3. These temperatures are bounding for accident storage conditions of the HSM-H loaded with the 37PTH DSC with heat loads up to 30 kW (all HLZCs).

Temperature distributions for the blocked vent accident case for the 32PTH1 DSC with 31.2 kW decay heat load at 40 hours after complete blockage were presented in Appendix U, Figure U.4-12. The predicted transient temperature responses (heat up rates) for the HSM-H with the 32PTH1 DSC and 31.2 kW decay heat load were illustrated in Appendix U, Figure U.4-14 for blocked vent accident conditions. These responses are bounding for the blocked vent accident condition when the HSM-H is loaded with the 37PTH DSC. The resultant DSC shell temperatures from the blocked vent accident analysis performed in Appendix U, Section U.4.4

$$\text{Nomalized Area under Curve} = \frac{\text{Area under Axial Heat Profile}}{\text{Active Fuel Length}} = 0.998$$

Active fuel length = 144"

$$\text{Correction Factor} = \frac{1}{\text{Normalized Area under Curve}} = 1.002$$

The heat generating rates for the elements representing the active fuel are calculated based on the HLZC for the 37PTH DSC. The HLZC and its restrictions for the 37PTH basket are shown in Appendix Z.2, Figures Z.2-1 through Z.2-3.

Z.4.6.5 DSC Shell Temperatures

The DSC shell temperatures for normal, off-normal, and accident conditions are retrieved from the HSM-H and TC models described in Sections Z.4.4 and Z.4.5 and transferred to the 37PTH DSC model. The load cases considered for thermal evaluation of the 37PTH DSC model are summarized in Table Z.4-2.

Load cases S2 and S4 in Table Z.4-2, with the hot ambient temperatures in the HSM-H bound the maximum component temperatures for load case S1 and S3 with cold ambient temperatures. *As described in Section Z.4.4.3, the DSC shell temperature profiles for load case S4 are bounding for the load case S4A.* The DSC shell temperatures from load case S2 and S4 are chosen to determine the bounding maximum fuel cladding and basket component temperatures for all normal and off-normal storage conditions.

The DSC shell temperatures from load case S5 in Table Z.4-2, corresponding to the HSM-H blocked vent accident condition, are chosen to determine the maximum fuel cladding and basket component temperatures for storage accident conditions. *As described in Section Z.4.4.3, the extreme hot ambient accident storage condition (load case S6) is equivalent to the off-normal hot storage condition (load case S4).*

The DSC shell temperatures from load case T3 in Table Z.4-2, with the maximum normal temperature of 100°F in the OS200/OS200FC TC, are chosen to determine the maximum fuel cladding and basket component temperatures for all normal and off-normal transient transfer conditions. The evaluations in Appendix U, Section U.4.5 showed that the hot normal transfer condition with insolation bounds the hot off-normal transfer condition due to the requirement of providing sunshade during hot off-normal transfer with ambient temperature higher than 100°F.

The DSC shell temperatures from load case T7 in Table Z.4-2, with the maximum off-normal temperature of 117°F in the OS200/OS200FC TC, are chosen to determine the maximum fuel cladding and basket component temperatures for all normal and off-normal transient transfer conditions when air circulation option is used. Since the entrance air temperature of 117°F is higher than all other cases, load case T7 results in the bounding maximum DSC shell temperatures for load cases T2 and T4 as shown in Appendix U, Table U.4-12.

Based on evaluations in Appendix U, Section U.4.5.4.2, the accident case of "loss of neutron shield," corresponding to load case T8 in Table Z.4-2, results in the highest maximum component temperatures within the OS200FC TC. The DSC shell temperatures from this load case are chosen to determine the maximum fuel cladding and basket component temperatures for transfer accident conditions.

Table Z.4-2
Summary of Load Cases for Thermal Analysis of the 37PTH DSC

Load Case	Heat Load (kW)	Operation Condition	Description	Ambient Temperature (°F)	Insolation
S1	30	Normal storage	Normal cold, steady state	0	no
S2	30	Normal storage	Normal hot, steady state	100	yes
S3	30	Off-normal storage	Off-normal cold, steady state	-40	no
S4	30	Off-normal storage	Off-normal hot, steady state	117	yes
S4A	*30*	*Off-Normal Storage*	*Off-normal hot, steady-state with 50% blockage of HSM-H inlet vents*	*117(1)*	*Yes*
S5	30	Accident storage	Blocked vents @ 40 hours, transient	117	yes
S6	*30*	*Accident storage*	*Extreme hot ambient temperature, steady state*	*132(2)*	*Yes*
T1	30	Normal transfer	Normal cold, no air circulation, steady state	0	no
T2	30	Normal transfer	Normal cold with air circulation, steady state	0	no
T3	30	Normal transfer	Normal hot, no air circulation, transient	100	yes
T4	30	Normal transfer	Normal hot with air circulation, steady state	100	yes
T5	30	Normal transfer	Vertical operations hot, no water in TC/DSC annulus, transient	140	no
T6	30	Off-normal transfer	Off-normal hot, no air circulation transient	117	no
T7	30	Off-normal transfer	Off-normal hot, with air circulation, steady state	117	no
T8	30	Accident transfer	Loss of neutron shield, loss of sunshade, and no air circulation	117	yes
T9	30	Vacuum drying	Vertical operations, with water in TC/DSC annulus, steady state	N/A	no
T10	24	Normal transfer	Normal hot, no air circulation, steady state	100	yes
T11	24	Off-normal transfer	Off-normal hot, no air circulation, steady state	117	no
T12	24	Normal transfer	Vertical operations hot, no water in TC/DSC annulus, steady state	140	no

Notes

(1) A 24 hour average ambient temperature of 105°F is used. The 105°F considered is greater than the 24 hour average ambient temperature of 102°F corresponding to a maximum daily ambient temperature of 117°F as described in Appendix Z, Section Z.4.4.1.

(2) The maximum day temperature of 132 °F corresponds to a 24-hour average ambient temperature of 117 °F.

Table Z.4-3
Maximum Fuel Cladding Temperatures for Storage/Transfer Conditions

Load Case No.	Load Case Description	Fuel Cladding (°F)	Limit (°F)
S2	Normal storage, 30 kW, 100°F ambient with insolation	708	752
S3	Off-normal storage, 30 kW, -40°F ambient, no insolation	602	1058
S4	Off-normal storage, 30 kW, 117°F ambient with insolation	716	
S5	Accident storage, blocked vents @ 40 hours	839	
S6 (1)	*Accident storage, extreme hot ambient temperature*	*716*	
T1	Normal transfer, 30 kW, 0°F ambient, no insolation, no air circulation, steady state	713	752
T2	Normal transfer, 30 kW, 0°F ambient, no insolation, with air circulation, steady state	566	
T3	Normal transfer, 30 kW, 100°F ambient with insolation, transient at operation time of 23 hours	712	
T5	Vertical loading, 30 kW, 140°F ambient, no water in TC/DSC annulus, transient at operation time of 16 hours	715	
T9	Vacuum drying, 30 kW, steady state	594	
T7	Off-normal transfer, 30 kW, 117°F ambient with insolation, with air circulation, steady state	655	
T8	Accident transfer, 30 kW, loss of air circulation, loss of neutron shield, no sunshade, 117°F ambient with insolation	841	1,058
T10	Normal transfer, 24 kW, 100°F ambient with insolation, no air circulation, steady state	713	752
T12	Vertical loading, 24 kW, 140°F ambient, no water in TC/DSC annulus, steady state	739	

Table Z.4-5
Average Temperatures for Storage/Transfer Conditions

Load Case No.[1]	Condition	Fuel Assembly (°F)	Helium[2] (°F)	Basket Compartment (°F)	Large Rail, Top (°F)	Large Rail, Bottom (°F)	DSC Shell (°F)
S2	Normal	556	397	559	440	438	383
T1	Normal	560	414	564	467	374	393
T2	Normal	402	268	397	207	281	196
T3	Normal	560	414	563	460	385	390
T5	Normal	564	415	567	443	438	395
T9	Normal	440	304	426	281	284	224
T10	Normal	550	423	554	458	381	398
T12	Normal	**578**	**451**	583	467	462	431
S3	Off-normal	441	268	440	306	308	247
S4	Off-normal	**566**	**408**	568	450	447	393
T7	Off-normal	500	375	499	327	381	314
S5	Accident	693	537	706	602	598	552
S6 [3]	*Accident*	*566*	*408*	*568*	*450*	*447*	*393*
T8	Accident	**695**	**555**	707	610	557	555

(1) See Table Z.4-2 for a description of the load cases.
(2) Helium in the DSC cavity outside fuel assemblies along active fuel length.
(3) Based on discussion in Section Z.4.6.5, the load case S6 is equivalent to load case S4.

Z.5 Shielding Evaluation

The radiation shielding evaluation for the standardized NUHOMS® system (during loading, transfer, and storage) for the other NUHOMS® canisters is discussed in other sections and appendices of the UFSAR. The following radiation shielding evaluation specifically addresses the shielding performance of the NUHOMS®-37PTH system with design-basis PWR fuel assembly (FA) containing UO_2 or MOX fuel and control components (CCs) loaded in a NUHOMS®-37PTH DSC.

The radiation shielding evaluation described below is for the NUHOMS®-37PTH DSC transferred in a NUHOMS® OS200/OS200 FC transfer cask and stored in an HSM-H/HSM-HS module. There are two alternate configurations depending on the canister length applicable to the 37PTH system: (1) 37PTH-S, and (2) 37PTH-M. Each DSC has a different length. The basket layout for these configurations is identical except for the length of the compartments. Note that HSM and HSM-H (or HSM-HS) as well as OS200 and OS200 FC are used interchangeably throughout this appendix. Also, 37PTH, 37PTH-S, and 37PTH-M nomenclatures are used interchangeably. For each 37PTH DSC there are three heat load zoning configurations (HLZCs) described in Appendix Z.2, Figures Z.2-1 through Figure Z.2-3. The OS200 TC is the same as described in Appendix U.

Each DSC configuration is designed to store up to 37 intact (and up to 4 damaged in the corner fuel compartments, with remaining intact) PWR fuel assemblies with or without control components (CCs). The authorized 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). Furthermore, *non-fuel hardware that are positioned within the fuel assembly after the fuel assembly is discharged from the core such as Guide Tube or Instrument Tube Tie Rods or Anchors, Guide Tube Inserts, BPRA Spacer Plates or devices that are positioned and operated within the fuel assembly during reactor operation such as those listed above are also considered as CCs.* Based on the results of fuel qualification calculations described in Section Z.5.2, fuel with CC requires up to one additional year of cooling time for fuel assemblies with 10 or fewer years of cooling time. To assure that this evaluation is conservative, the design basis source terms are not adjusted to account for the additional cooling time required to accommodate the CC.

The design-basis PWR fuel source terms are derived from the bounding fuel assembly design (B&W 15x15 Mark B assembly design), as described in Section Z.5.2; however, B&W 15x15 Mark B assemblies are not authorized for storage or transfer in the 37PTH DSC. The SAS2H\ORIGEN-S depletion model of these assemblies is utilized because it results in radiological sources that are bounding for all PWR assemblies described in Appendix Z.2.

The NUHOMS®-37PTH DSCs are designed to store PWR fuel assemblies with or without CCs with the characteristic sources for CCs described in Table Z.5-13. The 37PTH DSCs have a maximum decay heat of 1.2 kW per assembly and a maximum heat load of 30.0 kW per canister. Note that while the specific fuel designs are listed in Appendix Z.2, storing reload fuel designed by other manufacturers is also allowed provided an analysis is performed to demonstrate that the limiting features listed in Appendix Z.2 bound the specific manufacturer's replacement fuel. The limiting parameters are the design basis radiological and decay heat source terms.

Z.5.2 Source Specification

Thermal and radiological source terms are calculated with the SAS2H/ORIGEN-S modules of SCALE 4.4 [5.1] for the fuel. The SAS2H/ORIGEN-S results are used to develop the fuel qualification tables listed in Appendix Z.2, Tables Z.2-6 through Z.2-10 and the design-basis fuel source terms suitable for use in the shielding calculations. The thermal and radiological source terms for the CCs, which are taken from Appendix J, are shown in Table Z.5-13.

The B&W 15x15 assembly is the bounding fuel assembly design for shielding purposes because it has the highest initial heavy metal loading and ^{59}Co content of the hardware regions as compared to fuel assemblies listed in Appendix Z.2 which are authorized contents of the NUHOMS®-37PTH DSC. The neutron flux during reactor operation is peaked in the *active fuel or* in-core region of the fuel assembly and drops off rapidly outside the *active fuel* region. Much of the fuel assembly hardware is outside of the *active fuel* region of the fuel assembly. To account for this reduction in neutron flux, the fuel assembly is divided into four exposure "regions." The four axial regions used in the source term calculation are: the bottom (nozzle) region, the fuel (*active fuel*) region, the (gas) plenum region, and the top (nozzle) region. The B&W 15x15 fuel assembly masses for each irradiation region are listed in Table Z.5-4. The light elements that make up the various materials for the various fuel assembly materials are taken from Reference [5.4] and are listed in Table Z.5-5. The design-basis heavy metal weight is 0.490 MTU. These masses are irradiated in the appropriate fuel assembly region in the SAS2H/ORIGEN-S models. To account for the reduction in neutron flux outside the *active fuel* regions neutron flux (fluence) scaling factors are applied to light element composition for each region. The neutron flux scaling factors which are from Reference [5.4] are given in Table Z.5-6.

Evaluations of the existing data with SAS2H and the 44-group ENDF/B-V library used in the analysis are documented in References [5.11] and [5.12]. These comparisons all show generally good agreement between the calculations and measurements, and show no trend as a function of burnup in the data that would suggest that the isotopic predictions, and therefore neutron and gamma source terms, would not be in good agreement. A similar conclusion is also reached by the results documented in JAERI report [5.13]. In fact, for the case with 46,460 *MWd*/MTU burnup, the isotopic predictions are all within 2% of those measured. There are ongoing efforts, some of which are documented in Reference [5.10], to obtain more data for burnups above 45 GWD/MTU. There is no reason to expect that the ongoing evaluations of the higher burnup fuel will result in less favorable comparisons. Therefore, the uncertainty in the gamma source term, and associated dose rates, is estimated to be within ±5 %.

As noted in References [5.14] and [5.10], there is no public data for the neutron component currently available that bounds a fuel burnup of up to 62 GWD/MTU. However, as documented in Reference [5.14] and confirmed in the SAS2H analysis, the total neutron source with increasing burnup is more and more dominated by spontaneous fission neutrons. Reviewing the output from the SAS2H runs, the neutron source term is due almost entirely to the spontaneous fission of ^{244}Cm (~94% of all neutrons both spontaneous fission and (α, n)). After reviewing the measured ^{244}Cm content compared to the ^{244}Cm content predicted by SAS2H and the 44-group ENDF/B-V library documented in References [5.11] and [5.12] for burnups up to 46,460 *MWd*/MTU, it is readily apparent that the calculated values are within ±11 % of the measured values, with most of the predicted values within ±5% of the measured. Finally, there is no observed trend as a function of burnup in the data that would indicate that the predicted ^{244}Cm

As discussed previously, reconstituted and/or damaged fuel is also acceptable for the DSC payload. Reconstituted fuel may contain up to 10 fuel rods replaced with irradiated solid stainless steel rods. Reconstituted fuel has a rather small effect on the dose rate such that for cooling times less than 10 years, 1 year of cooling time is added if reconstituted irradiated stainless steel rods are present. If the cooling time is greater than 10 years, no additional cooling time is needed. Under normal conditions, damaged fuel has essentially no impact on the dose rate as the source term would not be impacted and gross axial source redistribution is not likely. Damaged fuel under accident conditions is addressed by assuming the fuel turns to rubble. This assumption is only applicable to the transfer casks shielding analysis.

Parameters that influence the source term calculations are fuel assembly specific power (expressed in MW/fuel assembly (MW/FA)) and the total time between cycles. Other depletion parameters like cycle length and number of cycles are derived from the target burnup, MTU loading and specific power. The most important parameter for the calculation of source terms is the specific power. Specific power for typical US-PWR fuel assemblies are ≤ 18 MW/FA. For the transfer cask, radiological source terms in the heat zones that essentially contribute to the dose rates near the wet cask are calculated using a specific power of 25 MW/FA and results in a conservative estimation of the source terms. The time between cycles utilized is 30 days and is adequately bounding.

The design-basis source terms are defined as the burnup/initial enrichment/cooling time combination given in the fuel qualification tables that result in the maximum dose rate on the surface of the HSM (HSM-H) or TC (OS200). Note that for a given HLZC, the design basis HSM source will not necessarily be the same as the corresponding design basis TC source. For the HSM, the middle of the roof centerline is selected as the dose location, and for the middle of the TC the cask side is selected as the dose location. This approach is consistent with the method used to determine the fuel qualification tables for the Standardized NUHOMS® canister designs described in Chapter 7, Section 7.2.3 and Appendices M.5, P.5, and U.5.

HLZC 3 (Figure Z.2-3 in Appendix Z.2) produced the bounding total surface dose rate for both the HSM-H and OS200 TC containing the 37PTH DSC. The enveloping HLZC selected for the shielding analysis (shown in Figure Z.5-1) of the 37PTH DSC bounds the actual heat load configuration shown in Figure Z.2-1 and Z.2-2 because 1.2 kW fuel is assumed in all 16 peripheral locations.

A sample SAS2H/ORIGEN-S input file for the *active fuel* region for the 26 GWD/MTU, 1.5 wt. % U-235, and 3.1-years cooling case is listed in Section Z.5.6.1. Input for reconstituted fuel is similar, except for a reduced number of fuel pins from 208 to 198, light element masses that reflect reconstituted rods, and slightly different power input to maintain the same burnup for a reduced fuel mass.

Z.5.2.1 Gamma Source Term for MCNP

Z.5.2.1.1 Design Basis Gamma Fuel Assembly Source Terms

Once the design basis burnup/enrichment/cooling time combinations have been determined for each shielding configuration of interest, four SAS2H/ORIGEN-S runs are required for each combination to determine gamma source terms for the four fuel assembly regions (i.e., bottom,

active fuel, plenum and top). The only difference between the runs is in Block #10 "Light Elements" of the SAS2H input and the 82$$ card in the ORIGEN-S input. Each run includes the appropriate light elements for the region being evaluated and the 82$$ card is adjusted to have ORIGEN-S output the total gamma source for the *active fuel* region and only the light element source for the plenum, bottom, and top regions. Gamma source terms for the *active fuel* region include contributions from actinides, fission products, and activation products. The bottom, plenum and top nozzle regions include the contribution from the activation products in the specified region only. The SAS2H/ORIGEN-S gamma radiation source is output in the CASK-81 energy group structure.

A design basis source is developed for each decay heat (0.4, 0.7, and 1.2 kW) and shielding structure combination used in the shielding analysis. The enveloping configuration evaluated in the shielding analyses is based on four radial zones. Radial zone 1 is comprised of the center fuel compartments of the 37PTH DSC, radial zone 2 is comprised of the inner middle 8 assemblies, and the outer middle 12 assemblies are of radial zone 3. The remaining 16 outer assemblies define radial zone 4. Source terms are generated for the following enveloping (hypothetical) configuration.

(1) Radial zone 1 and 2: 0.4 kW fuel
(2) Radial zone 3: 0.7 kW fuel
(3) Radial zone 4: 1.2 kW fuel

The source terms for radial zone 1 and 2 fuel in a 37PTH DSC loaded in the HSM-H (0.4 kW, 40 GWD/MTU, 1.5 wt. % U-235, and 40.2-years cooling) are shown in Table Z.5-7. The source terms for radial zone 3 fuel in a 37PTH DSC loaded in the HSM-H (0.7 kW, 16 GWD/MTU, 0.7 wt. % U-235 and 3.1-years cooling) are shown in Table Z.5-8. The source terms for radial zone 4 fuel in a 37PTH DSC loaded in the HSM-H (1.2 kW, 26 GWD/MTU, 1.5 wt. % U-235, 3.1-years cooling) are shown in Table Z.5-9. The bounding radiological source terms for 37PTH DSC in the cask from assemblies in the similar zones are shown in Table Z.5-7, Table Z.5-10 through Table Z.5-12. Again, dose rates on and around the cask containing 37PTH DSC from those sources are bounded by the dose rates presented in Appendix U.5. Dose rates on and around HSM containing 37PTH DSC from sources in Table Z.5-7 through Table Z.5-9 are bounded by the dose rates presented in Appendix U.5, Table U.5-1. Dose rates near the bounding cask and the bounding HSM are due to radiological sources in Table U.5-7 through Table U.5-9.

Z.5.2.1.2 Design Basis CC Source Terms

The design basis CC source terms are taken from Appendix J and are listed in Table Z.5-13. All CCs to be stored in the 37PTH DSC must be bounded by this source. The source terms from the fuel assembly and the CCs are utilized in the MCNP shielding models.

Z.5.2.1.3 Uncertainty in Gamma Source Terms

Almost 100% of the gamma spectrum from light elements is in the range of 0.70 to 1.33 MeV which corresponds exactly to two of the most prominent lines of ^{60}Co. As for fission products, the main contributors after six years with a fraction greater then 5% in the range of 0.01 to 0.90 MeV are: ^{90}Sr, ^{90}Y, ^{106}Rh, ^{137}Cs, ^{144}Pr, ^{154}Eu, and ^{155}Eu. Contributions from ^{90}Y, ^{106}Rh, ^{137}Cs, ^{144}Pr, and ^{154}Eu are dominant in the range of 0.90 to 1.50 MeV. ^{106}Rh, ^{147}Sm, and ^{142}Ce are the

strongest emitters at energies greater then 2.0 MeV. The accuracy of gamma spectrum is dependent upon the energy. Photon rates computed for fission products tend to be more accurate then those for actinides because the calculation of their inventory has less uncertainty [5.1].

Shortly after discharge the emission at higher energies is dominated by actinides. This is true for energies >4 MeV at all cooling times and energy above 3.5 MeV for cooling times after 10 years [5.1]. The major part of this emission comes from ^{244}Cm. Thus the uncertainty for energy groups of order 3.0 MeV and greater is bounded with the precision with which the inventory of 244Cm is calculated. Per SCALE 4.4 [5.1], reported experimental ^{244}Cm densities are accurate within ± 20%. The gamma emission intensity from Cm, which is proportional to the quantity of Cm in the actinide inventory, is bounded by this value. Uncertainty in the source strength in the gamma energy range 0.5 to 2.5 MeV is approximately 10 to 15 % [5.1].

Z.5.2.2 Neutron Source Term for MCNP

One SAS2H/ORIGEN-S run is required for each burnup/initial enrichment/cooling time combination to determine the total neutron source term for the *active fuel* regions. At discharge the neutron source is almost equally produced from ^{242}Cm and ^{244}Cm. The other strong contributor is ^{252}Cf, which is approximately 1/10 of the Cm intensity, but its share vanishes after 6 years of cooling time because the half-life of ^{252}Cf is 2.65 years. The half-lives of ^{242}Cm and ^{244}Cm are 163 days and 18 years, respectively. Contributions from the next strongest emitters, ^{238}Pu and ^{240}Pu, are lower by a factor of 1000 and 100, respectively, relative to ^{244}Cm. For the ranges of exposures, enrichments, and cooling times in the fuel qualification tables, ^{244}Cm represents more than 85% of the total neutron source. The neutron spectrum is, therefore, relatively constant for the fuel parameters addressed herein.

The magnitude of the neutron source is provided as the final row in the gamma source term tables; see Tables Z.5-7 through Table Z.5-12.. Neutron source terms for use in the MCNP shielding models are calculated by multiplying the assembly source by the number of assemblies in heat zones of interest. The magnitude of the neutron source is also increased to account for the axial distribution in the fuel, as explained in Section Z.5.2.3. The neutron source terms for 37PTH DSC in the cask from assemblies in the similar zones are shown in the last row of Table Z.5-7, Table Z.5-10 through Table Z.5-12. Dose rates near the cask containing 37PTH DSC from radiological sources in those tables are bounded by the dose rates presented in Appendix U.5.

The fixed source spectrum in MCNP is assumed to follow a ^{244}Cm spontaneous fission spectrum for all of the calculations in this chapter, except Section Z.5.5. It is based on the following relationship:

$$f(E) = C\exp\left[\left(\frac{-E}{a}\right)\sinh(bE)^{1/2}\right]$$

where input parameters $a = 0.906$ MeV and $b = 3.848$ (MeV)$^{-1}$, as given in the MCNP manual [5.2] and E is energy (MeV).

Z.5.3 Material Densities

The material masses given in Table Z.5-4 for the fuel are used to calculate material densities for *active fuel*, plenum, top, and bottom regions of the fuel assembly.

In order to account for sub-critical multiplication, an initial enrichment of 5.0 wt. % ^{235}U is used to calculate the amount of ^{235}U in dry fuel for the shielding models. For the shielding analyses of wet fuel an enrichment of 3.5 wt. % is considered.

Material densities used in the various MCNP models are summarized in Table Z.5-15.

Z.5.4 Shielding Evaluation

Dose rate contributions from the bottom, *active fuel*, plenum, and top regions, as appropriate, from 37 fuel assemblies are calculated with the MCNP Code [5.2] at various locations on and around the NUHOMS®-37PTH DSCs within the HSM and OS200 TC.

The following shielding evaluation discussion specifically addresses the NUHOMS®-37PTH DSC in an OS200 TC and the NUHOMS®-37PTH DSC in HSM-H using the design-basis source terms described in the above sections.

Z.5.4.1 Computer Program

MCNP [5.2] is a general-purpose Monte Carlo N-Particle code that can be used for neutron, photon, electron, or coupled neutron/photon/electron transport. The code treats an arbitrary three-dimensional configuration of materials in geometric cells bounded by first- and second-degree surfaces and some special fourth-degree surfaces. Pointwise (continuous energy) cross-section data are used. For neutrons, all reactions given in a particular cross-section evaluation are accounted for in the cross section set. For photons, the code takes account of incoherent and coherent scattering, the possibility of fluorescent emission after photoelectric absorption, absorption in pair production with local emission of annihilation radiation, and bremsstrahlung. Important standard features that make MCNP very versatile and easy to use include a powerful general source; an extensive collection of cross-section data; and an extensive collection of variance reduction techniques that can be employed to track particles through very complex deep penetration problems. MCNP was employed to take advantage of its mesh tallies capabilities in calculating dose rates distributed over the surface of the HSM. It also allows more point detectors to be used in a single run that substantially reduces the number of input/out decks needed to perform ISFSI site dose rate calculations described in Appendix Z.10.

Z.5.4.2 Spatial Source Distribution

The source components are:

the neutron sources due to the active fuel region,
the gamma source due to the active fuel region,
the gamma source due to the plenum,
the gamma source due to the top region,
the gamma source due to the bottom region,
the gamma source due to the CC in the active fuel region,
the gamma source due to the CC in the plenum region, and
the gamma source due to the CC in the top region.

Axial peaking is accounted for in the active fuel region by inputting an axial shape, as discussed in Section Z.5.2.3.

Z.5.4.3 Cross Section Data

The cross-section data used is the continuous energy ENDF/B-VI provided with the MCNP code [5.2]. The cross-section data allows coupled neutron/gamma-ray dose rate evaluation to be made

Table Z.5-4
PWR Fuel Assembly Material Mass

Fuel Assembly Region, Length	Fuel Assembly Part	Material	Standard Fuel Assembly Mass (kg)	Reconstituted Fuel Assembly Mass (kg)
Top Nozzle, 6.23 in.	Top nozzle/misc steel	SS-304	9.2	9.2
	Hold down spring	Inconel-718	1.8	1.8
Plenum, 8.73 in.	Upper spring	Inconel-718	4.3	4.3
	Upper end cap	Zircaloy-4	1.0	1.0
	Encompassing clad.	Zircaloy-4	5.8	5.5
	Upper end grid	Inconel-718	1.1	1.1
	Stainless steel rods	SS304	NA	1.7
Active Fuel Region, 142.29 in.	Fuel stack	UO_2	490	466
	Encompassing clad.	Zircaloy-4	101.1	96.2
	Encompassing guide tube	Zircaloy-4	6.3	6.3
	Spacer grids	Inconel-718	5.0	5.0
	Grid supports	Zircaloy-4	0.64	0.64
	Stainless steel rods	SS304	NA	27.2
Bottom Nozzle, 8.38 in.	Lower end plug	Zircaloy-4	8.9	8.5
	Encompassing guide tube	Zircaloy-4	0.1	0.1
	Lower guide tube plugs	Zircaloy-4	1.4	1.4
	Lower end fitting	SS 304	8.2	8.2
	Lower end grid	Inconel-718	1.1	1.1
	Stainless steel rods	SS304	NA	0.5

Table Z.5-6
Flux Scaling Factors By Fuel Assembly Region

Fuel Assembly Region	Flux Factor
Bottom	0.20
Active Fuel	1.00
Plenum	0.20
Top	0.10

Table Z.5-7
Gamma and Neutron Source Term for 0.4 kW Fuel in HSM-H and Transfer Cask
(40 GWD/MTU, 1.5 wt. % U-235 and 40.2-Years Cooled)

E_{min}, MeV	to	E_{max}, MeV	Bottom Nozzle	*Active Fuel*	Plenum	Top Nozzle
0.00E+00	to	5.00E-02	7.2380E+09	4.7496E+14	1.8095E+10	5.0982E+09
5.00E-02	to	1.00E-01	5.2916E+08	1.0646E+14	1.2048E+09	3.6922E+08
1.00E-01	to	2.00E-01	1.3030E+08	5.6885E+13	3.0067E+08	9.0209E+07
2.00E-01	to	3.00E-01	6.9349E+06	1.7714E+13	1.7184E+07	4.8499E+06
3.00E-01	to	4.00E-01	8.5081E+06	1.1915E+13	1.8479E+07	5.6596E+06
4.00E-01	to	6.00E-01	8.7989E+06	8.3849E+12	6.5178E+06	3.5267E+05
6.00E-01	to	8.00E-01	1.3012E+09	7.4970E+14	6.5968E+09	1.0991E+09
8.00E-01	to	1.00E+00	1.2570E+09	3.3399E+12	6.3727E+09	1.0643E+09
1.00E+00	to	1.33E+00	1.4897E+11	5.7423E+12	3.2870E+11	1.0348E+11
1.33E+00	to	1.66E+00	4.2070E+10	7.8039E+11	9.2825E+10	2.9222E+10
1.66E+00	to	2.00E+00	2.9749E+01	3.0160E+10	1.9384E+01	2.9906E-03
2.00E+00	to	2.50E+00	9.9838E+05	1.5828E+09	2.2029E+06	6.9349E+05
2.50E+00	to	3.00E+00	1.5481E+03	1.5644E+08	3.4158E+03	1.0753E+03
3.00E+00	to	4.00E+00	1.4132E-11	2.4777E+07	7.2313E-11	1.1641E-11
4.00E+00	to	5.00E+00	0.0	8.3631E+06	0.0	0.0
5.00E+00	to	6.50E+00	0.0	3.3560E+06	0.0	0.0
6.50E+00	to	8.00E+00	0.0	6.5827E+05	0.0	0.0
8.00E+00	to	1.00E+01	0.0	1.3975E+05	0.0	0.0
Total Gamma, g/(sec*FA):			2.0152E+11	1.4359E+15	4.5414E+11	1.4043E+11
Total Neutrons, n/(sec*FA):			2.45E+08			

Table Z.5-8
Gamma and Neutron Source Term for 0.7 kW Fuel in HSM-H
(16 GWD/MTU, 0.7 wt. % U-235 and 3.1-Years Cooled)

E_{min}, MeV	to	E_{max}, MeV	Bottom Nozzle	*Active Fuel*	Plenum	Top Nozzle
0.00E+00	to	5.00E-02	4.4940E+11	1.5792E+15	7.2752E+11	2.0102E+11
5.00E-02	to	1.00E-01	3.6142E+10	3.5259E+14	7.8898E+10	2.4684E+10
1.00E-01	to	2.00E-01	1.7143E+10	3.0346E+14	2.4517E+10	5.9562E+09
2.00E-01	to	3.00E-01	9.4927E+08	8.5593E+13	1.2821E+09	2.9609E+08
3.00E-01	to	4.00E-01	2.5845E+09	6.5586E+13	2.5512E+09	3.8801E+08
4.00E-01	to	6.00E-01	4.5156E+10	5.1149E+14	2.9466E+10	4.6335E+07
6.00E-01	to	8.00E-01	2.4132E+10	1.1358E+15	1.8580E+10	5.5733E+08
8.00E-01	to	1.00E+00	1.8816E+11	1.6376E+14	3.5961E+10	1.0722E+11
1.00E+00	to	1.33E+00	1.0365E+13	1.6100E+14	2.2906E+13	7.2050E+12
1.33E+00	to	1.66E+00	2.9271E+12	4.7933E+13	6.4685E+12	2.0347E+12
1.66E+00	to	2.00E+00	4.9208E+05	1.9787E+12	1.0715E+06	3.3370E+05
2.00E+00	to	2.50E+00	6.9466E+07	4.8384E+12	1.5351E+08	4.8286E+07
2.50E+00	to	3.00E+00	1.0771E+05	1.6044E+11	2.3803E+05	7.4873E+04
3.00E+00	to	4.00E+00	7.0404E-11	1.9982E+10	3.5643E-10	5.9375E-11
4.00E+00	to	5.00E+00	0.0	2.6313E+06	0.0	0.0
5.00E+00	to	6.50E+00	0.0	1.0559E+06	0.0	0.0
6.50E+00	to	8.00E+00	0.0	2.0712E+05	0.0	0.0
8.00E+00	to	1.00E+01	0.0	4.3972E+04	0.0	0.0
Total Gamma, g/(sec*FA):			1.4056E+13	4.4134E+15	3.0293E+13	9.5799E+12
Total Neutrons, n/(sec*FA):			7.65E+07			

Table Z.5-9
Gamma and Neutron Source Term for 1.2 kW Fuel in HSM-H
(26 GWD/MTU, 1.5 wt. % U-235 and 3.1-Years Cooled)

E_{min}, MeV	to	E_{max}, MeV	Bottom Nozzle	*Active Fuel*	Plenum	Top Nozzle
0.00E+00	to	5.00E-02	5.9553E+11	2.5874E+15	9.2766E+11	2.5159E+11
5.00E-02	to	1.00E-01	4.5059E+10	5.7503E+14	9.8125E+10	3.0681E+10
1.00E-01	to	2.00E-01	2.2923E+10	4.9868E+14	3.1517E+10	7.4034E+09
2.00E-01	to	3.00E-01	1.2794E+09	1.3921E+14	1.6579E+09	3.6805E+08
3.00E-01	to	4.00E-01	3.5933E+09	1.0559E+14	3.4182E+09	4.8229E+08
4.00E-01	to	6.00E-01	6.4599E+10	9.5797E+14	4.2138E+10	5.9173E+07
6.00E-01	to	8.00E-01	3.4457E+10	2.0047E+15	2.6195E+10	7.3103E+08
8.00E-01	to	1.00E+00	2.8211E+11	3.3987E+14	5.3071E+10	1.6071E+11
1.00E+00	to	1.33E+00	1.2886E+13	2.2684E+14	2.8463E+13	8.9554E+12
1.33E+00	to	1.66E+00	3.6391E+12	7.0676E+13	8.0379E+12	2.5290E+12
1.66E+00	to	2.00E+00	6.4873E+05	3.0250E+12	1.4094E+06	4.3844E+05
2.00E+00	to	2.50E+00	8.6362E+07	7.9221E+12	1.9075E+08	6.0017E+07
2.50E+00	to	3.00E+00	1.3391E+05	2.4106E+11	2.9578E+05	9.3062E+04
3.00E+00	to	4.00E+00	4.9444E-10	2.9954E+10	2.5023E-09	4.1722E-10
4.00E+00	to	5.00E+00	0.0	7.1335E+06	0.0	0.0
5.00E+00	to	6.50E+00	0.0	2.8628E+06	0.0	0.0
6.50E+00	to	8.00E+00	0.0	5.6159E+05	0.0	0.0
8.00E+00	to	1.00E+01	0.0	1.1924E+05	0.0	0.0
Total Gamma, g/(sec*FA):			1.7575E+13	7.5172E+15	3.7685E+13	1.1936E+13
Total Neutrons, n/(sec*FA):			2.07E+08			

Table Z.5-10
Gamma and Neutron Source Term for 0.7 kW Fuel in Transfer Cask
(41 GWD/MTU, 1.5 wt. % U-235 and 15.3-Years Cooled)

E_{min}, MeV	to	E_{max}, MeV	Bottom Nozzle	*Active Fuel*	Plenum	Top Nozzle
0.00E+00	to	5.00E-02	1.2286E+11	9.0825E+14	2.6136E+11	7.8631E+10
5.00E-02	to	1.00E-01	1.3903E+10	1.7662E+14	3.0617E+10	9.6190E+09
1.00E-01	to	2.00E-01	4.2418E+09	1.2093E+14	7.9725E+09	2.3218E+09
2.00E-01	to	3.00E-01	2.2019E+08	3.5615E+13	4.0409E+08	1.1567E+08
3.00E-01	to	4.00E-01	4.3028E+08	2.2704E+13	6.1841E+08	1.5104E+08
4.00E-01	to	6.00E-01	4.7523E+09	4.3871E+13	3.1134E+09	9.5408E+06
6.00E-01	to	8.00E-01	3.7875E+09	1.4024E+15	8.3654E+09	1.1300E+09
8.00E-01	to	1.00E+00	1.4849E+09	2.7595E+13	6.9177E+09	1.2245E+09
1.00E+00	to	1.33E+00	4.0374E+12	7.0887E+13	8.9053E+12	2.8042E+12
1.33E+00	to	1.66E+00	1.1402E+12	1.4542E+13	2.5149E+12	7.9190E+11
1.66E+00	to	2.00E+00	5.7905E+01	5.9467E+10	3.7724E+01	5.8231E-03
2.00E+00	to	2.50E+00	2.7057E+07	4.2719E+09	5.9681E+07	1.8793E+07
2.50E+00	to	3.00E+00	4.1955E+04	3.4284E+08	9.2542E+04	2.9140E+04
3.00E+00	to	4.00E+00	2.8145E-11	8.0733E+07	1.4387E-10	2.3229E-11
4.00E+00	to	5.00E+00	0.0	2.2773E+07	0.0	0.0
5.00E+00	to	6.50E+00	0.0	9.1400E+06	0.0	0.0
6.50E+00	to	8.00E+00	0.0	1.7930E+06	0.0	0.0
8.00E+00	to	1.00E+01	0.0	3.8071E+05	0.0	0.0
Total Gamma, g/(sec*FA):			5.3293E+12	2.8235E+15	1.1740E+13	3.6893E+12
Total Neutrons, n/(sec*FA):			6.60E+08			

Table Z.5-11
Gamma and Neutron Source Term for 1.2 kW Fuel in Dry Transfer Cask and Dry DSC (41 GWD/MTU, 1.5 wt. % U-235 and 5.7-Years Cooled)

E_{min}, MeV	to	E_{max}, MeV	Bottom Nozzle	*Active Fuel*	Plenum	Top Nozzle
0.00E+00	to	5.00E-02	5.2208E+11	1.6711E+15	9.4885E+11	2.7127E+11
5.00E-02	to	1.00E-01	4.9591E+10	3.2893E+14	1.0834E+11	3.3981E+10
1.00E-01	to	2.00E-01	2.2112E+10	2.6241E+14	3.2747E+10	8.1996E+09
2.00E-01	to	3.00E-01	1.2027E+09	7.4633E+13	1.6962E+09	4.0752E+08
3.00E-01	to	4.00E-01	3.2061E+09	5.0767E+13	3.2805E+09	5.3396E+08
4.00E-01	to	6.00E-01	5.4315E+10	7.1099E+14	3.5400E+10	3.3782E+07
6.00E-01	to	8.00E-01	2.9444E+10	2.4115E+15	2.5071E+10	1.1392E+09
8.00E-01	to	1.00E+00	5.7782E+10	3.1229E+14	1.7355E+10	3.3292E+10
1.00E+00	to	1.33E+00	1.4280E+13	2.3696E+14	3.1485E+13	9.9178E+12
1.33E+00	to	1.66E+00	4.0328E+12	6.6938E+13	8.8913E+12	2.8008E+12
1.66E+00	to	2.00E+00	1.4750E+02	9.1671E+11	2.0957E+02	5.0427E+01
2.00E+00	to	2.50E+00	9.5704E+07	1.4733E+12	2.1100E+08	6.6466E+07
2.50E+00	to	3.00E+00	1.4840E+05	7.3378E+10	3.2718E+05	1.0306E+05
3.00E+00	to	4.00E+00	1.5839E-09	9.2897E+09	8.0179E-09	1.3365E-09
4.00E+00	to	5.00E+00	0.0	3.2853E+07	0.0	0.0
5.00E+00	to	6.50E+00	0.0	1.3186E+07	0.0	0.0
6.50E+00	to	8.00E+00	0.0	2.5868E+06	0.0	0.0
8.00E+00	to	1.00E+01	0.0	5.4926E+05	0.0	0.0
Total Gamma, g/(sec*FA):			1.9053E+13	6.1290E+15	4.1549E+13	1.3068E+13
Total Neutrons, n/(sec*FA):			9.53E+08			

Table Z.5-12
Gamma and Neutron Source Term for 1.2 kW Fuel in Wet Transfer Cask and Wet DSC (62 GWD/MTU, 3.4 wt. % U-235 and 13.7-Years Cooled)

E_{min}, MeV	to	E_{max}, MeV	Bottom Nozzle	*Active Fuel*	Plenum	Top Nozzle
0.00E+00	to	5.00E-02	1.7944E+11	1.5844E+15	3.7278E+11	1.1091E+11
5.00E-02	to	1.00E-01	1.9782E+10	3.0653E+14	4.3486E+10	1.3658E+10
1.00E-01	to	2.00E-01	6.5620E+09	2.1926E+14	1.1666E+10	3.2966E+09
2.00E-01	to	3.00E-01	3.4476E+08	6.4672E+13	5.9403E+08	1.6415E+08
3.00E-01	to	4.00E-01	7.3817E+08	4.1428E+13	9.6075E+08	2.1450E+08
4.00E-01	to	6.00E-01	9.5763E+09	1.0931E+14	6.2610E+09	1.3550E+07
6.00E-01	to	8.00E-01	6.5800E+09	2.2430E+15	1.1497E+10	1.3821E+09
8.00E-01	to	1.00E+00	1.9439E+09	6.4547E+13	8.5510E+09	1.5745E+09
1.00E+00	to	1.33E+00	5.7363E+12	1.1388E+14	1.2647E+13	3.9826E+12
1.33E+00	to	1.66E+00	1.6199E+12	2.2518E+13	3.5716E+12	1.1247E+12
1.66E+00	to	2.00E+00	9.4297E+01	1.1134E+11	6.1438E+01	1.1267E-02
2.00E+00	to	2.50E+00	3.8443E+07	1.0291E+10	8.4758E+07	2.6691E+07
2.50E+00	to	3.00E+00	5.9610E+04	1.0084E+09	1.3143E+05	4.1387E+04
3.00E+00	to	4.00E+00	8.5875E-11	1.7590E+08	4.3749E-10	7.1419E-11
4.00E+00	to	5.00E+00	0.0	4.1462E+07	0.0	0.0
5.00E+00	to	6.50E+00	0.0	1.6641E+07	0.0	0.0
6.50E+00	to	8.00E+00	0.0	3.2645E+06	0.0	0.0
8.00E+00	to	1.00E+01	0.0	6.9315E+05	0.0	0.0
Total Gamma, g/(sec*FA):			7.5812E+12	4.7697E+15	1.6674E+13	5.2385E+12
Total Neutrons, n/(sec*FA):			1.20E+09			

Z.9.1.3 Leak*age* Tests

The DSC canister confinement boundary is tested using two procedures described below. Personnel performing the leakage test are qualified in accordance with SNT-TC-1A [9.2].

Procedure 1 is accomplished during fabrication:

Upon completion of all canister shell welding and attachment of the inner bottom cover plate to the shell, a temporary seal plate is placed over the open end of the DSC. A bag or other enclosure is placed around the outside of the entire DSC and it is filled with helium. The DSC cavity is evacuated and a helium leakage test is performed using a port in the seal plate. This test is used to show that the entire DSC confinement boundary tested is leak tight ($1x10^{-7}$ ref cm^3/s).

Procedure 2 of the testing occurs after the DSC has been loaded with fuel assemblies:

The DSC cavity has been dried, back filled with helium and the inner top cover plate and the vent and drain port cover plates have been welded in place. After these welds are completed, a temporary test cover is installed or the outer top cover plate is welded in place with at least the root pass of the full weld. The cavity between inner top cover plate and the temporary test cover or outer top cover plate is evacuated and a helium leakage test is performed using a test port in the temporary test cover or in the outer top cover plate. The leakage test thus includes the weld attaching the inner top cover plate to the canister shell, the vent and drain port cover plate welds and the base metal of the inner top cover plate and vent and drain port cover plates. The vent and drain ports are filled with helium prior to welding the vent and drain port covers. This test verifies that the tested welds and cover plates are leak tight ($1x10^{-7}$ ref cm^3/s).

Z.9.1.4 Components

The Standardized NUHOMS® system does not include any components such as valves, rupture discs, pumps, or blowers. The gaskets in the transfer cask do not require acceptance testing other than the leak testing cited above. No other components of the NUHOMS® system require testing, except as discussed in this chapter.

Z.9.1.5 Shielding Integrity

The transfer cask poured lead shielding integrity will be confirmed via gamma scanning prior to first use. The detector and examination grid will be matched to provide coverage of the entire lead-shielded surface area. For example, for a 6" × 6" grid, the detector will encompass a 6" × 6" square. The acceptance criterion is attenuation greater than or equal to that of a test block matching the cask through-wall configuration with lead and steel thicknesses equal to the design minima less 5%.

The radial neutron shielding is provided by filling the neutron shield shell with water during operations. No testing is necessary. The neutron shield material in the lid and bottom end is a proprietary polymer resin. The shielding performance of the resin will be assured by written

procedures controlling temperature, measuring, and mixing of the components, degassing of the resin, and verification of the mass or volume of resin installed.

The gamma and neutron shielding materials of the storage system itself are limited to concrete HSM components and steel shield plugs in the DSC. The integrity of these shielding materials is ensured by the control of their fabrication in accordance with the appropriate ASME, ASTM or ACI criteria. No additional acceptance testing is required.

Z.9.1.6 Thermal Acceptance

No thermal acceptance testing is required to verify the performance of each storage unit other than that specified in the Technical Specifications for initial loading.

The heat transfer analysis for the basket includes credit for the thermal conductivity of neutron-absorbing materials, as specified in Section Z.4.3. Because these materials do not have publicly documented values for thermal conductivity, testing of such materials will be performed in accordance with Section Z.9.1.7.6.

Z.11.2.10.4 Corrective Actions

Evaluation of HSM-H or TC neutron shield damage as a result of a fire is to be performed to assess the need for temporary shielding (for HSM-H or cask, if fire occurs during transfer operations) and repairs to restore the TC and HSM-H to pre-fire design conditions.

Z.11.2.11 <u>Accident Temperatures</u>

For this accident condition, very high ambient temperature of 132 °F is postulated.

Z.11.2.11.1 Cause of Accident

At some locations, there is a possibility of very high ambient temperatures, higher than the off-normal conditions used in Section Z.11.1. Therefore, to envelope these high temperatures, a 132 °F ambient temperature is selected for this accident evaluation.

Z.11.2.11.2 Accident Analysis

The analysis of the 37PTH DSC in HSM-H with 132 °F ambient temperatures is bounded by the blocked vent accident analysis documented in Section Z.11.2.7 because the inlet and outlet vents are functioning normally for this accident. Similarly, the analysis of the 37PTH DSC in OS200 TC is bounded by the accidental cask drop analysis documented in Section Z.11.2.5 because the neutron shield in the TC is functioning normally. The 37PTH DSC temperatures and internal pressures, HSM-H and TC temperatures and hence stresses are all bounded by the blocked vent analysis or the accidental cask drop analysis. Therefore, the NUHOMS® 37PTH system will withstand these very high ambient temperatures without breach of the confinement boundary.

Z.11.2.11.3 Accident Dose Calculations

There are no radiation dose consequences for this accident. The NUHOMS® 37PTH DSC is designed and tested as a leak-tight containment boundary. Very high ambient temperature of 132 °F does not breach the containment boundary. Therefore radioactive material inside the DSC will remain sealed in the DSC and, therefore, will not contaminate the environment.

Z.11.2.11.4 Corrective Actions

None required.

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ALL DIMENSIONS ARE NOMINAL UNLESS A SPECIFIC TOLERANCE IS INDICATED WITH THE DRAWING DIMENSION

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INTERPRET WELD SYMBOLS PER ANSI / AWS 2.4

U.S. Patent No. 4,780,269 & 5,438,597
Transnuclear, Inc.

TRANSNUCLEAR
AN AREVA COMPANY

SAFETY ANALYSIS REPORT
NUHOMS®37PTH
TRANSPORTABLE CANISTER FOR PWR FUEL
BASKET SHELL ASSEMBLY

DRAWING NO.	SCALE	SHEET
NUH37PTH-72-1002	NONE	1 OF 5

PROPRIETARY AND SECURITY RELATED INFORMATION WITHHELD UNDER 10 CFR 2.390

DRAWING NO.	SHEET	REVISION
NUH37PTH-72-1002	2 OF 5	0B

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DRAWING NO.	SHEET	REVISION
NUH37PTH-72-1002	3 OF 5	0B

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NUHOMS®37PTH
TRANSPORTABLE CANISTER FOR PWR FUEL
ALTERNATE TOP CLOSURE

DRAWING NO.	SCALE	SHEET
NUH37PTH-72-1004	NONE	1 OF 6

PROPRIETARY AND SECURITY RELATED INFORMATION WITHHELD UNDER 10 CFR 2.390

DRAWING NO.	SHEET	REVISION
NUH37PTH-72-1004	2 OF 6	08

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SAFETY ANALYSIS REPORT
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BASKET ASSEMBLY

DRAWING NO.	SCALE	SHEET
NUH37PTH-72-1011	NONE	1 OF 7

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DAMAGED FUEL END CAPS

DRAWING NO.	SCALE	SHEET
NUH37PTH-72-1015	NONE	1 OF 1

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MAIN ASSEMBLY

DRAWING NO.	SCALE	SHEET
NUH69BTH-72-1001	NONE	1 OF 4

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DRAWING NO	SCALE	SHEET
NUH69BTH-72-1003	NONE	1 OF 4

PROPRIETARY AND SECURITY RELATED INFORMATION WITHHELD UNDER 10 CFR 2.390

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NUH69BTH-72-1004	NONE	1 OF 6

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DRAWING NO.	SHEET	REVISION
NUH69BTH-72-1011	5 OF 5	0B

NUH69BTH-72-1015 1 OF 1

PROPRIETARY AND SECURITY RELATED INFORMATION WITHHELD UNDER 10 CFR 2.390

REVISION	DESCRIPTION	DATE
0B	REVISED PER NRC RSI #1	07/21/11
0A	FIRST ISSUE	01/28/11

ALL DIMENSIONS ARE NOMINAL UNLESS A SPECIFIC TOLERANCE IS INDICATED WITH THE DRAWING DIMENSION

DIMENSIONS ARE IN INCHES AND DEGREES UNLESS OTHERWISE SPECIFIED. DIMENSIONING IN ACCORDANCE WITH ASME Y14.5M-1994.

INTERPRET WELD SYMBOLS PER ANSI / AWS 2.4

U.S. PATENT NO. 4,780,269

TRANSNUCLEAR, INC

TRANSNUCLEAR
AN AREVA COMPANY

SAFETY ANALYSIS REPORT
NUHOMS®69BTH
TRANSPORTABLE CANISTER FOR BWR FUEL
DAMAGED FUEL END CAPS

DRAWING NO.	SCALE	SHEET
NUH69BTH-72-1015	NONE	1 OF 1

Enclosure 9 to TN E-31217

Listing of Computer Files Contained in Enclosure 10

Listing of Computer Files Contained in Enclosure 10

Disk ID No. (size)	Discipline	System/Component	File Series (topics)	Number of files
Enclosure 10 DVD (1 of 3) **Structural Folder** (1.59 GB)	**Structural**	**Basket**	37PTH-Basket-0-Deg-Buckling-Analysis – **Folder** (Z.3.7.4.3 – input and output files for 37PTH-Basket 0-Deg Buckling Analysis – LS DYNA analysis)	8
			37PTH-Basket-Accident-180-Deg-Side Drop – **Folder** (Z.3.7.4.3 – input and output files for 37PTH-Basket Accident 180-Deg Side Drop – LS DYNA analysis)	8
		Fuel Rod	37PTH-14X14 PWR Fuel Corner Drop – **Folder** (Z.3.5.3 – input and output files for 37PTH 14X14 PWR Fuel Corner Drop - LSDYNA analysis)	7
			69BTH-7x7 BWR Fuel Corner Drop – **Folder** (Y.3.5.3 – input and output files for 69BTH 7x7 BWR Fuel Corner Drop – LS DYNA analysis)	7
Enclosure 10 DVD (2 of 3) **Thermal Folder** (2.58 GB)	**Thermal**	**69BTH DSC in HSM-H with Partial Vent Blockage**	001-HSM-H-Partial-Vent – **Directory**	
			1-35kW_117D19_50Block – **Folder** HSM-H Thermal Analysis with 50% Partial Blocked Vent and 69BTH DSC at 35 kW	32
			2-69BTH_HSM_35kW_117F_50Block – **Folder** Thermal analysis of 69BTH DSC/Basket Model at 35 kW with DSC shell temperatures from HSM-H with 50% Partial Blocked Vent	16
Enclosure 10 DVD (3 of 3) **Thermal Folder** (2.58 GB)	**Thermal**	**69BTH DSC in HSM-H under Extreme Accident Ambient Temperatures**	002-Extreme-Accident-Ambient – **Directory**	
			1-35kW_117D19_acc – **Folder** HSM-H Thermal Analysis with Extreme Accident Ambient Temperature and 69BTH DSC at 35 kW	32
			2-69BTH_HSM_35kW_117_acc – **Folder** Thermal analysis of 69BTH DSC/Basket Model at 35 kW with DSC shell temperatures from HSM-H with Extreme Accident Ambient Temperature	16

Enclosure 10 to TN E-31217

Computer Files Associated with CoC 1004 Amendment 13, Revision 1 and RSI responses on DVDs (Proprietary)

AFFIDAVIT PURSUANT
TO 10 CFR 2.390

Transnuclear, Inc.)
State of Maryland) SS.
County of Howard)

I, Jayant Bondre, depose and say that I am a Vice President of Transnuclear, Inc., duly authorized to execute this affidavit, and have reviewed or caused to have reviewed the information which is identified as proprietary and referenced in the paragraph immediately below. I am submitting this affidavit in conformance with the provisions of 10 CFR 2.390 of the Commission's regulations for withholding this information.

The information for which proprietary treatment is sought is contained in Enclosures 1, 7 and 10 and is listed below:

- Portions of the RSI responses
- Portions of CoC 1004 UFSAR Appendix Y, Sections Y.1.5 and Y.3
- Portions of CoC 1004 UFSAR Appendix Z, Section Z.1.5
- Computer files associated with the RSI responses provided on DVDs

These documents have been appropriately designated as proprietary.

I have personal knowledge of the criteria and procedures utilized by Transnuclear, Inc. in designating information as a trade secret, privileged or as confidential commercial or financial information.

Pursuant to the provisions of paragraph (b) (4) of Section 2.390 of the Commission's regulations, the following is furnished for consideration by the Commission in determining whether the information sought to be withheld from public disclosure, included in the above referenced document, should be withheld.

1) The information sought to be withheld from public disclosure involves drawings related to the design of dry spent fuel storage systems, portions of the safety analysis report related to the structural evaluation, which are owned and have been held in confidence by Transnuclear, Inc.

2) The information is of a type customarily held in confidence by Transnuclear, Inc. and not customarily disclosed to the public. Transnuclear, Inc. has a rational basis for determining the types of information customarily held in confidence by it.

3) Public disclosure of the information is likely to cause substantial harm to the competitive position of Transnuclear, Inc. because the information consists of descriptions of the design of dry spent fuel storage systems, the application of which provide a competitive economic advantage. The availability of such information to competitors would enable them to modify their product to better compete with Transnuclear, Inc., take marketing or other actions to improve their product's position or impair the position of Transnuclear, Inc.'s product, and avoid developing similar data and analyses in support of their processes, methods or apparatus.

Further the deponent sayeth not.

Jayant Bondre

Jayant Bondre
Vice President, Transnuclear, Inc.

Subscribed and sworn to me before this 22nd day of July, 2011.

Lauren McKee
Notary Public

My Commission Expires

Lauren McKee
NOTARY PUBLIC
Anne Arundel County, Maryland
My Commission Expires 2/12/2015

LAUREN MCKEE NOTARY PUBLIC ANNE ARUNDEL COUNTY MD