TRANSNUCLEAR
AN AREVA COMPANY

December 15, 2008
E-27377

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 1 to the NUHOMS® HD System, Response to Request for Additional Information (Docket No. 72-1030; TAC NO. L24153)

Reference: Letter from B. Jennifer Davis (NRC) to Donis Shaw (TN), "REQUEST FOR ADDITIONAL INFORMATION FOR REVIEW OF AMENDMENT 1 TO THE NUHOMS® HD SYSTEM (TAC NO. L24153), INCLUDING UPDATED REVIEW SCHEDULE," November 14, 2008

This submittal provides responses to the request for additional information (RAI) forwarded by the referenced letter. Enclosure 2 herein provides each of the NRC staff RAI followed by a TN response. Enclosure 3 provides a list of proposed NUHOMS® HD Technical Specifications (TS) and Updated Final Safety Analysis Report (UFSAR) pages that changed and are included herein.

Enclosure 4 provides TS and UFSAR Amendment 1 new and replacement pages. In both the TS and the UFSAR, Amendment 1 Revision 0 changes and Amendment 1 Revision 1 changes are shown, using italicized text and revision bars; however, Revision 1 changes are shaded to distinguish them from Revision 0 changes. For the UFSAR, page footers for new and replacement pages are annotated as "Amendment 1, Rev. 1, 12/08."

This submittal includes proprietary information in Enclosures 4 and 7 which may not be used for any purpose other than to support your staff's review of the application. In accordance with 10 CFR 2.390, I am providing an affidavit (Enclosure 1) specifically requesting that you withhold this proprietary information from public disclosure. This submittal also includes security-related information. Accordingly, Enclosure 5 provides a public version of the TS and UFSAR Amendment 1, Revision 1 changed pages.

Should the NRC staff require additional information to support review of this application, please do not hesitate to contact Mr. Don Shaw at 410-910-6878 or me at 410-910-6930.

Sincerely,

Robert Grubb
Senior Vice President - Engineering

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NMSS01
NMSS

cc: B. Jennifer Davis (NRC SFST) (six paper copies of this cover letter and Enclosures 1 through 4, plus one copy of Enclosures 6 and 7, all provided separately)

Enclosures:

1. Affidavit Pursuant to 10 CFR 2.390
2. RAI Responses
3. List of Changed Pages for CoC 1030 Amendment 1 Application Revision 1
4. NUHOMS® HD Amendment 1 Application Revision 1, Changed and New Proposed Technical Specifications and Proposed Updated Final Safety Analysis Report Pages (Proprietary Version)
5. NUHOMS® HD Amendment 1 Application Revision 1, Changed and New Proposed Technical Specifications and Proposed Updated Final Safety Analysis Report Pages (Public Version)
6. Listing of the Files Contained in Enclosure 7
7. Computer Disc Containing Input and Output Files (Proprietary)

AFFIDAVIT PURSUANT
TO 10 CFR 2.390

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

I, Robert Grubb, depose and say that I am Senior Vice President of Transnuclear, Inc. (TN), 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 4 and 7, and is listed below:

1. SAR Drawing 10494-72-5, Revision 3A
2. SAR Drawing 10494-72-17, Revision 3A
3. SAR Drawing 10494-72-2003-SAR, Revision 1A
4. Portions of SAR Chapter 5 which provide a SAS2H Input File for Fuel Qualification
5. Certain Computer Input and Output Files for Thermal Analyses

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 changed safety analysis report drawings related to the analysis of dry storage systems, changed safety analysis report pages related to the shielding analyses of dry storage systems, plus computer input and output files associated with thermal analyses 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) The information is being transmitted to the Commission in confidence under the provisions of 10 CFR 2.390 with the understanding that it is to be received in confidence by the Commission.

4) The information, to the best of my knowledge and belief, is not available in public sources, and any disclosure to third parties has been made pursuant to regulatory provisions or proprietary agreements which provide for maintenance of the information in confidence.

5) Public disclosure of the information is likely to cause substantial harm to the competitive

position of Transnuclear, Inc. because:

a) A similar product is manufactured and sold by competitors of Transnuclear, Inc.

b) Development of this information by Transnuclear, Inc. required expenditure of considerable resources. To the best of my knowledge and belief, a competitor would have to undergo similar expense in generating equivalent information.

c) In order to acquire such information, a competitor would also require considerable time and inconvenience related to the development of a design and analysis of a dry spent fuel storage system.

d) The information required significant effort and expense to obtain the licensing approvals necessary for application of the information. Avoidance of this expense would decrease a competitor's cost in applying the information and marketing the product to which the information is applicable.

e) The information consists of drawings and computer input and output files associated with the design and analysis of dry spent fuel storage systems, the application of which provides 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.

f) In pricing Transnuclear, Inc.'s products and services, significant research, development, engineering, analytical, licensing, quality assurance and other costs and expenses must be included. The ability of Transnuclear, Inc.'s competitors to utilize such information without similar expenditure of resources may enable them to sell at prices reflecting significantly lower costs.

Further the deponent sayeth not.

Robert Grubb
Senior Vice President, Transnuclear, Inc.

Subscribed and sworn to ~~me~~ before this 15th day of December, 2008.

Notary Public

My Commission Expires 10 / 14 / 2012

CHAPTER 1 - GENERAL INFORMATION

1.1 **(Technical Specifications (TS))** Expand and clarify the definition of damaged and intact fuel assemblies. Include separate definitions for intact and damaged fuel rods in the proposed TS.

The definitions for intact and damaged fuel assemblies provided in the SAR appear to be for intact and damaged fuel rods, not fuel assemblies. Interim Staff Guidance Document 1 (ISG-1), Rev. 2, "Classifying the Condition of Spent Nuclear Fuel for Interim Storage and Transportation Based on Function" provides guidance on acceptable definitions. In the provided definitions, list specific defects which would still permit an assembly to be classified as undamaged, and justify accordingly.

The current definitions are not sufficiently precise to categorize the large number of defects which may be observed in a fuel assembly. It is unclear how a fuel assembly with a missing grid spacer, for example, would be categorized under the current definition.

This information is needed to determine compliance with 10 CFR 72.236(a).

Response to 1.1

TN had responded to a similar question from the Staff during the initial review and approval of NUHOMS® HD System [1]. TN's response to RAI 2-2 [2] clarified the definition for damaged fuel which was reviewed and accepted by the Staff. That definition is included in the NUHOMS® HD Technical Specification 1-1. This definition was used to select and load fuel in the NUHOMS® HD System by various General Licensees.

TN therefore requests that the existing definition in the NUHOMS® HD Technical Specification should remain unchanged.

References for RAI Response 1.1

[1] Letter from Mary Jane Ross-Lee (NRC) to Michael Mason (TN), "REQUEST FOR ADDITIONAL INFORMATION REGARDING THE TRANSNUCLEAR NUHOMS® HD HORIZONTAL MODULAR STORAGE SYSTEM (TAC NO. L23738)," dated December 13, 2004

[2] Letter from Michael Mason (TN) to Mary Jane Ross-Lee (NRC), "RAI Response for the NUHOMS® HD Storage System Docket No. 72-1030 (TAC No. L23738)," dated February 18, 2005

1.2 **(Technical Specifications)** Clarify which portions of the SAR are incorporated by reference in the proposed TS.

It is unclear if certain portions of the SAR are intended to be incorporated by reference in the TS. For example, the footer, "NUHOMS® HD System Technical Specifications," denotes pages in the introduction to the SAR which are incorporated into the TS. The opening page of Section 12 appears to indicate the Operating Controls and Limits are

intended to be part of the TS, but the "Technical Specifications" footer does not appear on any pages in Section 12.

This information is needed to assure compliance with 10 CFR 72.44(c).

Response to 1.2

The only SAR sections incorporated by reference in the Technical Specifications are SAR Chapter 9 Sections 9.1.7.1, 9.1.7.2, 9.1.7.3, 9.5.2, 9.5.3.5, and 9.5.4.3. These portions are clearly marked in the SAR, using bold text, with text boxes explaining this approach. TS 4.3.1, "Neutron Absorber Tests," also makes this clear in its reference to these SAR sections.

The original application for approval of the NUHOMS® HD System provided proposed Technical Specifications (TS) and TS Bases as Chapter 12 of the Safety Analysis Report, consistent with NUREG-1536, "Standard Review Plan for Dry Cask Storage Systems." Upon approval of Certificate of Compliance (CoC) No. 1030, Amendment 0, dated January 10, 2007, the TS were listed as Appendix A of the CoC, and were therefore removed from the SAR with the issuance of Final Safety Analysis Report (FSAR) Revision 0, dated February 23, 2007. The TS Bases are part of FSAR Chapter 12. Page 12-1 of the SAR has been changed to reflect this history and therefore provide clarity.

1.3 **(SAR Sections 1.2.3, 2.1, etc.)** Describe the difference between "fuel types" as specified in the original FSAR, and "fuel classes" as specified in Amendment 1. Explain the reason for the change, and the basis for inclusion.

The original approved FSAR specified the 32PTH Dry Shielded Canister (DSC) contents as "up to 32 intact PWR Westinghouse 15x15... *fuel assemblies*" (emphasis added). The application for Amendment 1 specifies the DSC contents as "up to 32 intact PWR Westinghouse 15x15... *class* fuel assemblies"(emphasis added). This appears to be a more inclusive category, and any additional fuel (including future changes or improvements to existing designs) that could be included under fuel classes should be described and fully qualified. Rather than selecting the potential worst-case fuel design, the design-basis limits used to determine whether fuel can be loaded in the future should be based on the current worst case fuel design.

This information is needed to determine compliance with 10 CFR 72.236(a).

Response to 1.3

The term "fuel type" refers to the individual fuel assembly designs whereas the term "fuel class" includes various fuel assembly types. The "fuel class" was originally included in Revision 0 of NUHOMS® HD Technical Specification 2.1.b, 2nd paragraph and in the original FSAR in Section 2.1 to allow for equivalent reload fuel assemblies that are enveloped by the fuel assembly design characteristics given in the Technical Specifications for a given assembly class to be accepted for loading.

All available fuel designs are evaluated and the bounding designs for each class are utilized in the design basis calculations using fuel assembly and cask models that are inherently conservative. Therefore, equivalent reload fuel assembly designs that are

covered by the evaluations documented in the UFSAR and whose significant parameters are bounded by those described in the UFSAR and Technical Specifications are acceptable for storage in the NUHOMS® HD system.

Technical Specification 1.1 is revised to add a definition of FUEL CLASS.

1.4 (Technical Specifications and SAR Sections 1.2.3, 2.1.1, 5.2, etc.) Include the definition of reconstituted fuel in the TS, Section 1.1. In addition, specify the cooling time requirements as stated in Section 5.2 of the SAR and in the Table 2 of the TS.

The definition for reconstituted fuel is not included in the TS, and is necessary to characterize what may be included in a reconstituted fuel assembly. In addition, the cooling time requirements described in Section 5.2 of the SAR should be included in the TS, if they are necessary to assure acceptable performance.

This information is needed to determine compliance with 10 CFR 72.236(a).

Response to 1.4

Technical Specification 1.1 is revised to add the following definition for RECONSTITUTED FUEL ASSEMBLY.

RECONSTITUTED FUEL ASSEMBLY:

A RECONSTITUTED FUEL ASSEMBLY is an INTACT FUEL ASSEMBLY where one or more fueled rods are replaced by rods containing stainless steel rods, stainless steel clad rods, low enriched uranium rods or natural uranium rods or Zircalloy (including other zirconium based alloy) rods or Zircalloy pellets. The nominal volume of the replacement rods is equivalent to the replaced fueled rods in the active fuel region of the fuel assembly.

The cooling time requirement applicable to A RECONSTITUTED FUEL ASSEMBLY is already specified in a footnote to Table 4 (2) of the proposed Technical Specification consistent with SAR Section 5.2. Also see the response to RAI 2.4.

1.5 (SAR Section 1.2.3, Table 2-1, etc.) Describe how WEV 17x17 assemblies, WEO 17x17 assemblies, and ANP Advanced MK BW 17x17 fuel assemblies fit into the fuel assembly *classes* that the 32 PTH DSC is designed to store, or explain why they are no longer included as acceptable contents.

Table 2-1 of the original FSAR includes WEV 17x17 assemblies and WEO 17x17 assemblies. These assemblies were described in Section 1.2.3 of the original SAR, as included in the Westinghouse 17x17 type fuel assemblies. In addition, the original FSAR includes ANP Advanced MK BW 17x17 fuel assemblies. However, the SAR for Amendment 1 does not specifically include any of these, and they have been removed from the Fuel Qualification Tables.

This information is necessary to determine compliance with 10 CFR 72.236(a).

Response to 1.5

As detailed in the response to RAI 1.3 above, the definition of fuel assembly class is provided in Technical Specification 1.1. Table 6-3 and Table 6-4 in Chapter 6 of the UFSAR contain all the design information of all the fuel assembly designs evaluated for criticality. These tables also categorize these fuel assembly designs into the various FUEL CLASSES. Accordingly, the WEV 17x17 design, WEO 17x17 design, and ANP Advanced MK BW 17x17 design all belong to the 17x17 class of fuel assemblies. Therefore, the individual fuel assembly designs are no longer included (specifically) as authorized contents, however, they are authorized to be stored because they are included as part of the 17x17 FUEL CLASS.

CHAPTER 2 – PRINCIPAL DESIGN CRITERIA

2.1 **(Technical Specifications)** Include clarifying statements in the proposed TS to specify that during loading, drying, or unloading operations the provisions of ISG-22, "Potential Rod Splitting Due to Exposure to an Oxidizing Atmosphere During Short-Term Cask Loading Operations In LWR or Other Uranium Oxide Based Fuel" will be satisfied, or provide justification and associated technical bases for any proposed alternative.

Amendment 1 to the NUHOMS HD Safety Analysis Report appears to indicate that with the exception of 0.25 volume percent of oxidizing gas in the cask during drying (Section B 12.3.1.1, page B12-10), the fuel assemblies will be maintained in an inert environment during loading, drying, and unloading. The staff requests assurance that the fuel assemblies will have limited (or no) exposure to an oxidizing environment during the loading, drying, or unloading operations.

This information is needed to determine compliance with 10 CFR 72.166.

Response to 2.1

Bases Section B 12.3.1 has been modified to provide the assurance requested, as follows:

"Technical Specification 3.1 requires the use of helium during the bulkwater removal process. Therefore, water from the DSC cavity is replaced by helium during the bulkwater removal process. Fuel cladding temperatures are low during this short duration process due to the presence of liquid water and helium.

Therefore use of helium during bulkwater removal, vacuum drying and long term storage operations assures that the fuel assemblies will have limited (or no) exposure to the oxidizing environment."

2.2 **(Technical Specifications)** Incorporate by reference wording from Section 2.1.1 of the Safety Analysis Report into the proposed TS such that:

a) The maximum fuel cladding temperature limit of 400°C (752°F) is set for normal conditions of storage and all short term operations from the spent fuel pool to the Independent Spent Fuel Storage Installation (ISFSI) pad including vacuum drying and helium backfilling of the NUHOMS®-32PTH DSC per the guidance delineated in ISG-11, Rev. 3, "Cladding Considerations for the Transportation and Storage of Spent Fuel."

b) The change in fuel cladding temperature is restricted to less than 65°C (117°F) and is limited to less than 10 cycles during DSC drying, backfilling and transfer operations, per the guidance delineated in ISG- 11, Rev. 3.

c) The maximum fuel cladding temperature limit is set to 570°C (1058°F) for accidents or off normal thermal transients, per the guidance delineated in ISG-11, Rev. 3.

The temperature limits defined in ISG-11, Rev. 3, have been established by the staff to ensure the integrity of the fuel cladding.

This information is needed to assure compliance with 10 CFR 72.166.

Response to 2.2

Technical Specifications 2.1: "Fuel to be Stored in the 32PTH DSC" and 3.1: "Fuel Integrity" assure that the authorized fuel is only loaded and the integrity of the fuel cladding is maintained. Therefore additional requirements in the Technical Specifications for the integrity of the fuel cladding are not necessary. The analyses documented in SAR Chapter 4 demonstrate that the NUHOMS® HD System meets the guidance given in ISG-11, Revision 3.

SAR Section 2.1.1 has been revised to reflect ISG-11, Revision3 and to provide text consistent with the ISG.

2.3 (SAR, Chapter 2, Tables 2-1, 2-3, etc.)

a) Clarify the composition of both the cladding and internal components of the non-fuel hardware that is being proposed as an addition to the contents of the package.

b) Additionally, provide data or discussion to show that no adverse chemical, galvanic, or other reactions between or among the hardware, or canister internals, or with water during loading, drying, unloading, or storage.

This information is needed to determine compliance with 10 CFR 72.120(3)(d).

Response to 2.3

a) The cladding material for the control components (CCs) include, stainless steel, Inconel, Zirconium-based alloys such as Zircaloy, M5 or Zirlo. The internal component material for the CCs includes non-fuel materials like Inconel, B_4C, Ag-In-Cd, Al_2O_3, etc.

SAR Section 5.2 is revised to add this clarification.

b) An evaluation of the NUHOMS® HD System for Chemical, Galvanic and other reactions is provided in Chapter 3, Section 3.4.1 of the UFSAR. Since the materials utilized in the CCs are similar to those utilized in fuel assemblies and/or basket materials, the evaluation presented in UFSAR Section 3.4.1 is also applicable to the CCs. The evaluation concludes that no adverse reactions (chemical, galvanic or other) are likely, due to the presence of CCs.

2.4 (SAR, Section 2.1.1, Table 2-1. TS p. 2-1, etc.) Clarify if the dimensions of the stainless steel or zirconium "dummy rods" placed in reconstituted assemblies are identical to the dimensions of the spent fuel rods, or provide the dimensions of the dummy rods for analysis, if different.

If the dimensions of the "dummy rods" do not match those of the spent fuel rods, the free volume within the cask will be different, which may influence the criticality analysis.

This information is needed to determine compliance with 10 CFR 72.236(a).

Response to 2.4

The criticality analysis documented in SAR Chapter 6 Section 6.2 and Section 6.4.2.3 is carried out assuming that the active fuel length of the fuel assembly extends "infinitely" in the axial direction. Therefore, only the nominal volume of the replacement or "dummy rods" placed in the reconstituted assemblies is required to be equivalent to that of the replaced fueled rods in the active fuel region of the fuel assembly. This requirement is included in Technical Specification 1.1. See the response to RAI 1.4.

2.5 **(Technical Specifications)** Provide justification for the increase in maximum assembly weight in the amended TS.

TS 2.1 lists maximum assembly plus control component (CC) weight as 1585 lbs. The maximum assembly weight of the design basis WE 17x17 is shown as 1575 lbs in Table 2-1 of the FSAR. What is the reason for this change?

This information is necessary to verify compliance with 10 CFR 72.236(a).

Response to 2.5

The structural analysis of the NUHOMS® HD System documented in UFSAR Chapter 3 was performed using a maximum fuel assembly plus control component weight of 1585 lbs. The other disciplines are not impacted by this change. The change in Technical Specification 2.1 is made to be consistent with the analysis documented in the UFSAR.

See Response to RAI 2.7 for additional information.

2.6 **(Technical Specifications)** Is the increase in maximum MTU/assembly taken into account in determining the bounding source-term analyses for all assembly classes? If not, explain how this is conservative or inconsequential, or provide alternate specifications in the Fuel Qualification Table (FQT).

Revised Table 2 lists maximum MTU/Assembly for every fuel class as 0.476 MTU. In the original FSAR, the WE 15x15 and CE 14x14 are listed as 0.467 MTU/assembly, and 0.385 MTU/assembly, respectively. It is not clear if an analysis was performed for any fuel classes other than the design-basis assembly at the maximum uranium loading. Explain how the design-basis assembly at maximum uranium loading analysis bounds the fuel classes, or provide further analysis.

This information is necessary to verify compliance with 10 CFR 72.104.

Response to 2.6

The increase in the maximum MTU per fuel assembly for all the fuel assembly classes is based on a calculation that determines a bounding set of source terms for all assembly classes. The analyses performed in SAR Section 5.2 demonstrate that the 0.476 MTU bounds the source terms for all assembly classes.

See response to RAI 5.4 for additional information.

2.7 **(SAR Table 2.1)** Explain how the increased maximum weight to 1,585 lbs per fuel assembly is considered in evaluating structural adequacy of the system components, including the fuel basket, under the design basis transfer cask side-and corner-drop handling accidents.

The maximum assembly plus CC weight of 1,585 lbs, as listed in the reformatted and expanded Table 2-1, is higher than the previously approved weights ranging from 1,450 to 1,575 lbs.

This information is necessary to determine compliance with 10 CFR 72.236(a).

Response to 2.7

Even though the fuel assembly weights were listed as 1450 lbs to 1575 lbs in Chapter 2, all of the structural analyses documented in Chapter 3 were performed with a bounding weight of 1585 lbs.

The structural evaluations of the system components are all based on the fuel assembly weight of 1585 lbs as described in the following UFSAR sections:

Description	*UFSAR Section/Pages*	*Remark*
32PTH DSC Weight	*Chapter 3, Section 3.2.1 (page 3-14)*	*1585 lbs is used in the weight calculation*
32PTH Type 1 DSC Weight	*Chapter A.3, Section A.3.2.1 (page A.3-7)*	*1585 lbs is used in the weight calculation*
Basket Evaluation	*Appendix 3.9.1, Section 3.9.1.2.3 (page 3.9.1-10, handling loads) and (page 3.9.1-15, accident loads)*	*1585 lbs is used in the basket structural calculation*
OS187H Transfer Cask Evaluation	*Appendix 3.9.2, Section 3.9.2.1.2 (page 3.9.2-3)*	*1585 lbs is used in the DSC weight calculation (32x1585 lbs = 50.72 kips), total loaded DSC weight is 108.76 kips. 115 kips is conservatively used in the cask structural evaluation*
OS187H Type 1 Transfer Cask Evaluation	*Appendix A.3.9.2, Section A.3.9.2.1.2 (page A.3.9.2-2)*	*1585 lbs is used in the Type 1 DSC weight calculation (32x1585 lbs = 50.72 kips), total loaded Type 1 DSC weight is 109.5 kips. 115 kips is conservatively used in the cask structural evaluation*

2.8 **(SAR Section 2.1.1)** Specify in the SAR that utilities which choose to load fuel with burnups greater than 45 GWD/MTU may be unable to transport those fuel assemblies at a later time.

Section 2.1.1 of the Safety Analysis Report states the maximum assembly burnup is 60 GWD/MTU. Currently, fuel assemblies with burnups above 45 GWD/MTU may be licensed for storage under 10 CFR Part 72, but may not be transported under 10 CFR Part 71. Utilities which chose to load fuel with burnups above 45 GWD/MTU do so at their own risk, as these utilities may not be able to transport these fuel assemblies at a later time.

This information is necessary to assure compliance with 10 CFR 72.236(a).

Response to 2.8

The Amendment 1 Application to CoC 1030 for the NUHOMS® HD System is for storage only under the provisions of 10CFR72. Transnuclear is in the process of preparing a revision to CoC 9302 for transportation of the 32PTH DSC with fuel characteristics including fuel assemblies with burnups above 45 GWD/MTU which are authorized in CoC 1030 in a suitable transportation package. This transportation CoC revision will demonstrate that all the fuel assemblies which are currently authorized for storage in CoC 1030, including fuel assemblies with burnups greater than 45 GWD/MTU, are also authorized for transportation in a suitable transportation package under the provisions of 10CFR71.

Based on previous discussion with NRC and the directives provided at that time, it is Transnuclear's understanding that the storage applications under 10CFR72 shall not include any evaluation for 10CFR71 conditions to demonstrate compliance with 10CFR71. These evaluations shall only be included in a CoC revision or new application under the provisions of 10CFR71.

CHAPTER 3 - STRUCTURAL EVALUATION

3.1 **(SAR Section 3.5.3.2)** Technical Specification 5.3.2 (Cask Drop), states in the Background, "...the potential exists to drop the cask 15 inches or more." Revise the SAR by performing an evaluation of structural integrity of the fuel cladding associated with corner- or end-dropping, as appropriate, a loaded 32PTH DSC within the OS187 transfer cask (TC) en route from the fuel handling building to the ISFSI.

NUREG-1536, page 3-6, in the section on "Structural Design Criteria and Design Features," specifies that "the [cask] design must ensure that the spent fuel will not experience accelerations that would damage its structural integrity or jeopardize its subcritical condition or retrievability." Section 3.5.3.2 of the (original) SAR states that the structural integrity of the fuel cladding due to the end drop loading condition will be evaluated by the user under the 10 CFR 50 site license. Recognizing that fuel clad ductility capability may be limited for the high burnup fuel certified for the NUHOMS HD system, the staff notes that, for a 10 CFR 72 general license, the structural integrity of fuel clad must be evaluated for the cask drop conditions described in the Technical Specifications for administrative lifting controls.

This information is necessary to assure compliance with 10 CFR 72.122(b).

Response to 3.1

Technical Specification Section 5.3 "Lifting Controls" contains two subsections:

- *Subsection 5.3.1 (Transfer Cask Lifting Heights)*

  This subsection describes the transfer cask lifting height and lifting orientations (handling requirements). The cask drop scenarios are based on the requirements of this subsection, which specifies that during the transfer operations the TC/32PTH DSC is in the horizontal position on the trailer and shall not exceed a maximum height of 80 inches from the bottom of the transfer cask.

  The SAR fuel drop analysis (side drop) is consistent with these Technical Specification handling requirements. As described in UFSAR Section 3.1.1.4, the transfer cask is transported to the ISFSI in a horizontal configuration. Therefore the only credible accident during storage or transfer operations is a side drop. A vertical or corner drop accident may be credible under 10CFR50 during loading onto the trailer (for example, use of a non-single failure lifting device) or during transportation operations governed under 10CFR71.

  These drop scenarios have been reviewed and found acceptable by the Staff, as stated in the Safety Evaluation Report (SER) for CoC 1030, Section 3.4.1.1(b) (page 3-11) "The end and corner drops are generally not considered credible during storage and transfer operations because the cask will always be in the horizontal orientation. The Staff finds this assumption meets the requirements of 10CFR72; however, an additional safety review by the user of the casks is necessary to demonstrate fuel cladding integrity under 10CFR Part 50 or to demonstrate that the drop accidents are not credible".

  Based on the discussion above, structural integrity of the fuel rod is analyzed for loads due to an 80 inch side drop impact as described in the UFSAR, Section

3.5.3 (page 3-29).

- *Subsection 5.3.2 (Cask Drop-Inspection Requirement)*

  As described in this Technical Specification, "The 32PTH DSC will be inspected for damage after any transfer cask drop of fifteen inches or greater." This subsection is an additional requirement established by the NRC Staff for inspection of NUHOMS® components following a cask drop during the transfer operations. The following paragraph describes the background:

  The NUHOMS® HD System is based on the Standardized NUHOMS® System described in CoC 1004. In the SER for CoC 1004 [1], page 3-20, the Staff performed independent calculations for the drop and "concluded that a drop of the loaded DSC from a height greater than 38 cm (15 inches) may cause damage to the DSC and the stored fuel. Because the ASME Code, Section III, for Service Level D permits plastic deformation, portions of the DSC shell and basket may sustain damage, without compromising the confinement boundary or geometry of the spent fuel array. However, such potential damage is the cause for limiting conditions of operation and surveillance." Therefore the Inspection Requirement was included in the Technical Specification.

  This same requirement from CoC 1004 was also included in Technical Specification 5.3.2 of CoC 1030 for the NUHOMS® HD System which specifies that the 32PTH DSC will be inspected for damage after any transfer cask side drop of fifteen inches or greater. The cask drop scenario (side drop) is defined in Section 5.3.1 of CoC 1030, therefore the inspection is limited to only the drop scenario (side drop) defined in Technical Specification 5.3.1.

  Technical Specification 5.3.2 of CoC 1030 (Cask Drop) is revised to clarify the drop scenarios (side drop only).

  References for RAI Response 3.1

[1] "SAFETY EVALUATION REPORT OF VECTRA TECHNOLOGIES, INC. a.k.a. PACIFIC NUCLEAR FUEL SERVICES, INC. SAFETY ANALYSIS REPORT FOR THE STANDARDIZED NUHOMS HORIZONTAL MODULAR STORAGE SYSTEM FOR IRRADIATED NUCLEAR FUEL," U.S. NUCLEAR REGULATORY COMMISSION OFFICE OF NUCLEAR MATERIALS SAFETY AND SAFEGUARDS, December 1994

3.2 **(SAR Appendix 3.9.10)** With respect to the mode shape displayed in Figure 3.9.10-18 of the original SAR, for the dominant axial vibration frequency of 141.07 hz, post-process the corner drop results by considering only the lid center for nodal averaging acceleration responses to ensure that a bounding forcing function input is used for the fuel clad evaluation as described in Question 3.1 above.

SAR Pages 3-9.10-11 through -13 describe post-processing of the LS-DYNA results, which suggest that the Figure 3.9.10-22 time-history response was a nodal average over the entire transfer cask lid. The staff notes that, per the Figure 3.9.10-18 mode shape, the vibratory component of the cask lid axial response may contribute significantly to the fuel clad impact response. As such, the nodal averaging associated with the entire lid or

for the nodes in the immediate vicinity of the point of impact at the corner of the cask may be inadequate, and it should only be performed for the lid center with the highest modal coefficients to capture the maximum fuel clad response.

This information is necessary to assure compliance with 10 CFR 72.122(b).

Response to 3.2

As documented in the NUHOMS® HD Technical Specifications and UFSAR, end drop and corner drop are not credible drop scenarios for the NUHOMS® HD System. Also, the response to RAI 3.1 further clarifies the credible drop scenarios.

CHAPTER 4 – THERMAL EVALUATION

4.1 **(SAR Section 4.8.2)** Provide a justification for use of the UO_2 material properties in Section 4.8.2 of the SAR. Update any analyses that rely on these values if any changes in the UO_2 material properties are warranted

The values provided for UO_2 material properties in Section 4.8.2 are from a dated source and are for unirradiated UO_2. Recent data has indicated that there may be potential changes in the characteristics and properties of irradiated UO_2 that could affect heat transfer in the fuel. This could potentially have an impact on the effective thermal conductivity for the spent fuel assemblies.

References:

"Thermal conductivities of irradiated UO_2 and $(U,Gd)O_2$," K. Minato et al. Journal of Nuclear Materials 300 (2002) 57–64.

"Effect of burn-up on the thermal conductivity of uranium dioxide up to 100.000 MWdt," C. Ronchi et al. Journal of Nuclear Materials 327 (2004) 58-76.

This information is needed to satisfy the provisions of 10 CFR 72.24(c)(3) and 10 CFR 72.236(f).

Response to 4.1

TN reviewed the above references for the thermal conductivity of UO_2. Based on the Ronchi study, UO_2 thermal conductivity of irradiated UO_2 with ~62 GWd/t and irradiation temperature $T_{irr} \geq 1300K$ (average T_{irr} for fuel pellet during irradiation according to M. Amaya et al.) drops significantly (more that 50%) compared to un-irradiated UO_2. The thermal conductivity values of UO_2 in Section 4.8.2 of the SAR [NUREG/CR-0200, Revision 6, SAR Reference 4.30] are compared to the values obtained from the Ronchi study as shown in Figure 4.1. The comparison shows that the SAR values in the fuel assembly temperature of interest are higher by approximately a factor of two compared to values obtained from the Ronchi study.

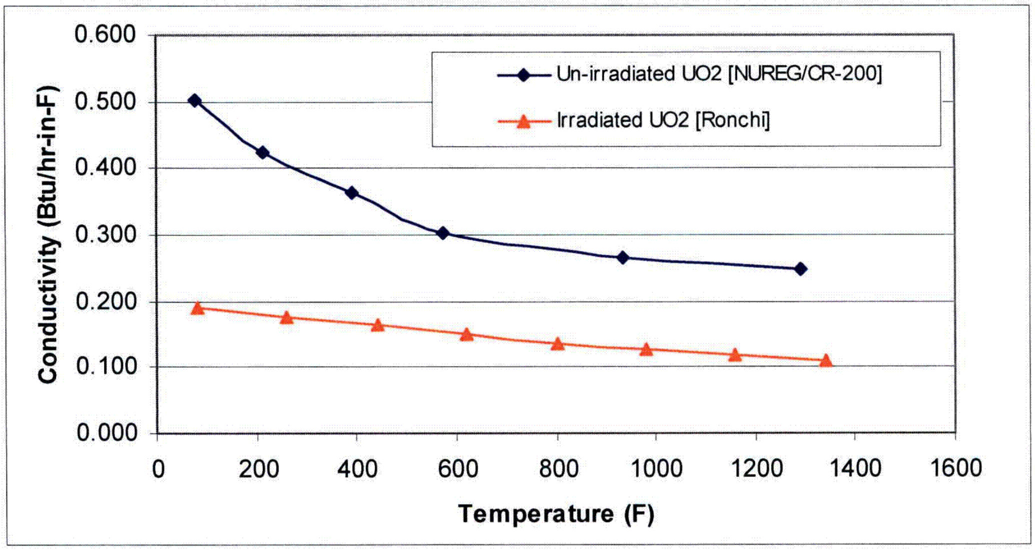

Figure 4.1 UO_2 Thermal Conductivity

Using irradiated UO_2 conductivity decreases the effective conductivity of the fuel assembly in the transverse direction. Note that as discussed in SAR Section 4.8.3.2, axial effective thermal conductivity of the fuel assembly is calculated based on the fuel cladding material only and does not include the UO_2 fuel pellet thermal conductivity. Therefore, the axial effective conductivity of the fuel assembly is not impacted.

A sensitivity analysis is performed to determine the impact of the irradiated UO_2 conductivity on the fuel cladding temperatures for the NUHOMS® HD System.

In the first step, the transverse effective conductivity for fuel assemblies with irradiated and un-irradiated UO_2 conductivities are calculated based on the methodology described in SAR Section 4.8.3.1.

In the second step, the calculated fuel assembly effective thermal conductivities from the first step are used in the 32PTH DSC model from Section 4.3.1.3 of the SAR to determine the maximum fuel cladding temperatures. Normal transfer conditions for the 32PTH DSC in the OS187H transfer cask with heat load zoning configuration 1 at 115° F ambient is selected for this analysis.

The transverse effective conductivity for fuel assemblies calculated based on irradiated [Ronchi et al] and un-irradiated [NUREG/CR-200] UO_2 conductivities are compared in Figure 4.2. The transverse effective conductivity for fuel assemblies used in the SAR evaluation based on UO_2 conductivities applied in the ANSYS model for fuel assembly effective conductivity calculation documented in SAR Section 4.2 (1) is added to Figure 4.2 for reference.

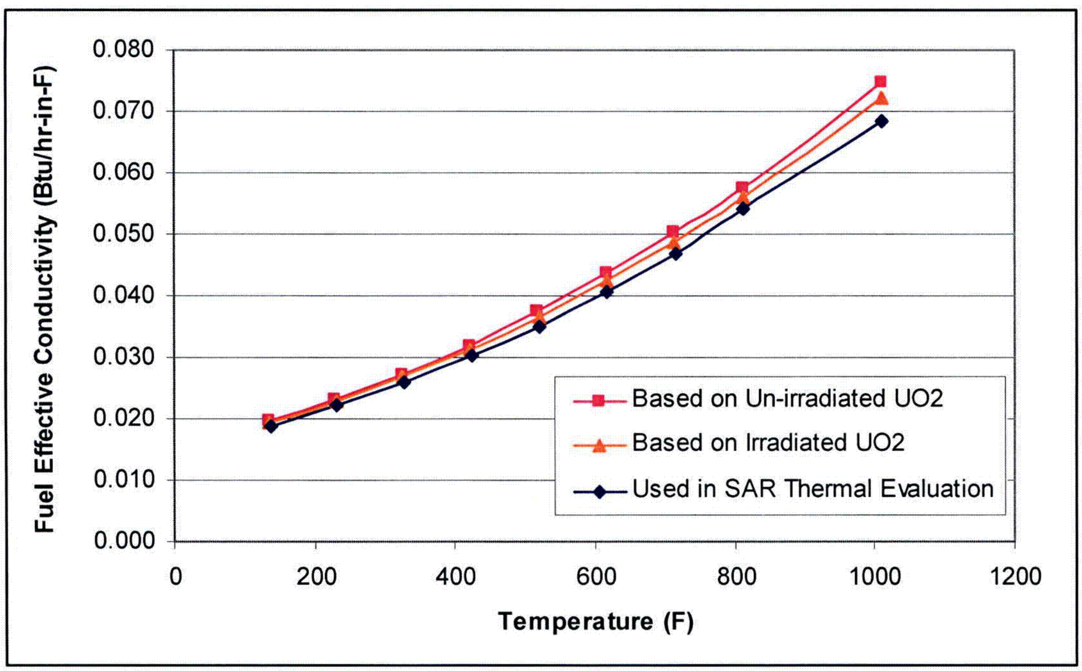

Figure 4.2 Transverse Effective Conductivity for Fuel Assembly

As seen in Figure 4.2, the fuel assembly effective thermal conductivity with irradiated UO_2 conductivity is approximately 3% lower than the one with un-irradiated UO_2 conductivity at the fuel cladding temperature of 700° F. The results of the sensitivity runs for the maximum fuel cladding temperature are summarized in Table 4.1 below.

Table 4.1 Maximum Component Temperatures from Sensitivity Analysis

	Maximum Temperature (°F) 32PTH DSC in OS187H, HLZC #1, 115°F Ambient (1)	
Component	(2)	(3)
Fuel Cladding	716	717
Fuel Compartment	691	692
Basket Al Plates	691	691
Basket Rails	560	560
DSC Shell	475	475

Notes:
1. *These values are comparable to those shown in SAR Table 4-1, 2nd part for Config. # 1.*
2. *K_{eff} for the fuel assembly is based on un-irradiated UO_2 conductivity as shown in SAR Section 4.2, subsection 1 [NUREG/CR-0200].*
3. *K_{eff} for the fuel assembly is based on irradiated UO_2 conductivity values from the Ronchi Study.*

The sensitivity analysis results show that the fuel cladding temperature changes by approximately 1 ° F (0.14%) which is negligible. These results show that the fuel cladding temperatures are not sensitive to changes in the conductivity of UO_2 due to irradiation. Therefore, use of UO_2 fuel pellet conductivity from NUREG/CR-0200 (SAR Reference 4.30) is reasonable for irradiated UO_2.

However, it should be noted that the ANSYS model described in SAR Section 4.8.3.1 erroneously but conservatively used UO_2 conductivity values which are lower than those shown in SAR Section 4.8.2 (1). These values are shown in Table 4.2 below.

Table 4.2 UO_2 Thermal Conductivity

SAR, Section 4.8 (1)		Used in ANSYS Model described in Section 4.8.3.1	
Temperature	Conductivity	Temperature	Conductivity
(°F)	(Btu/hr-in-°F)	(°F)	(Btu/hr-in-°F)
77	0.503	32	0.056
212	0.423	212	0.063
392	0.362	392	0.068
572	0.302	752	0.072
932	0.266		
1292	0.248		
1472	0.250		

The UO_2 conductivity values used in the ANSYS model are at least 30% lower than the values obtained from the Ronchi study. Use of lower UO_2 thermal conductivity values in the ANSYS model of the fuel assembly results in conservatively lower values of effective thermal conductivity for the fuel assembly. This in turn results in higher calculated fuel

cladding and DSC component temperatures which are conservative. The transverse effective conductivity for the fuel assembly used in the SAR thermal analysis is compared to the corresponding values from the sensitivity analysis in Figure 4.2.

Since the effective thermal conductivity for the fuel assembly used in the SAR thermal analysis is lower than the effective thermal conductivity for the fuel assembly with irradiated UO_2, the calculated maximum component temperatures and fuel cladding temperatures are conservative and the difference in irradiated and un-irradiated UO_2 fuel pellet thermal conductivity values does not affect thermal analysis results reported in the Amendment 1 SAR.

SAR Section 4.8.2 has been revised for clarification. Also a new Section 4.8.6 is added to document the sensitivity analysis.

Sample input and output files for the sensitivity analysis described in this response are included with this submittal on the enclosed proprietary disk.

4.2 **(SAR Chapter 4)** Provide the thermal analysis models (input and output files) for the NUHOMS HD system that were revised for Amendment 1.

The staff needs to review this information as part of its review to make a reasonable assurance safety finding for the amendment application.

This information is needed to satisfy the provisions of 10 CFR 72.24(d) and 10 CFR 72.236(f).

Response to 4.2

Due to the addition of the Combustion Engineering (CE) 16x16 class fuel assembly to the authorized contents of the NUHOMS® HD system, the model of this fuel assembly was developed for calculating its effective thermal conductivity. The input and output files for this model (A001-kHe_CE16x16.inp and A001-kHe_CE16x16.out, respectively) were provided with the application for Amendment 1 for NUHOMS® HD CoC No. 1030 (TN letter E-25747, Enclosure 4, Nov. 1, 2007).

There are no other thermal analysis models revised for NUHOMS® HD system Amendment 1.

Sample input and output files for the sensitivity analysis described in the response to RAI 4.1 are included with this submittal on the enclosed proprietary disk.

4.3 **(SAR Chapter 4)** Review the RAIs provided as part of the Standardized NUHOMS (CoC 1004) Amendment 10 (Accession numbers ML072410348 and ML081150596) and Amendment 11 (ML072980876) reviews that also apply to the NUHOMS HD design, and provide responses specific to the NUHOMS-HD Amendment 1 (CoC 1030), as appropriate.

Given the similarities between the Standardized NUHOMS and NUHOMS HD designs, changes made to the analyses or operating procedures of the Standardized NUHOMS designs, due to responses to previously issued RAIs, could potentially impact the NUHOMS HD design. The staff needs reasonable assurance that issues addressed for

the Standardized NUHOMS design are reviewed and applied to the NUHOMS HD design, as appropriate.

This information is needed to satisfy the provisions of 10 CFR 72.24(d) and 10 CFR 72.236(f).

Response to 4.3

TN has evaluated all the thermal RAI questions received for the Standardized NUHOMS® (CoC 1004) Amendment 10 (Accession numbers ML072410348 and ML081150596) and Amendment 11 (ML072980876).

The RAIs were categorized into requests for justification, providing detailed description, providing clarification, performing evaluations, sensitivity analysis, additional technical specification, and system specific questions for NUHOMS® (CoC 1004) Amendment 10 and Amendment 11.

The NUHOMS® HD Technical Specification 3.1.1 is consistent with changes made to Amendments 10 and 11 for the use of helium for removal of bulkwater from the DSC cavity. Additionally, clarifications made to Chapter T.8 and U.8 of the CoC 1004 SAR in response to RAI 4-1 as part of Standardized NUHOMS® (CoC 1004) Amendment 10 (Accession numbers ML072410348) are considered to be applicable to CoC 1030 Amendment 1. NUHOMS® HD SAR Section 8.1.1.2, Step 13 is revised to add a note identical to Amendment 10 SAR Chapter U.8, Section U.8.1.2 step 17 to assure that air will not enter the DSC cavity and helium will be present in the DSC cavity during movement of the transfer cask from the fuel pool to the decon area. The note is described below:

"Provisions shall be made to assure that air will not enter the DSC cavity. One way to achieve this is by replenishing the helium in the DSC cavity during cask movement from the fuel pool to the decon area in case of malfunction of equipment used for cask movement."

None of the other RAIs from Amendment 10 or 11 to CoC 1004 resulted in changes to the thermal analysis methodology used for these amendments or applicable portions of the NUHOMS® HD System.

4.4 **(SAR, Section 4.2)** Provide thermal conductivity data for aluminum clad metal matrix composites (MMCs) both parallel to, and perpendicular to the plate surface. Describe how any anisotropic thermal conductivity of aluminum clad MMCs is considered in the thermal analysis of the package.

The thermal analysis of the package assumes that Boral® has anisotropic thermal properties (Section 4.2, subsection 9 of the Safety Analysis Report), but does not take into account that aluminum clad MMCs may also have anisotropic thermal properties.

This information is required for compliance with 10 CFR 72.124(b), 10 CFR 72.236(c), and 10 CFR 72.236(f).

Response to 4.4

The aluminum clad metal matrix composite (MMC) is considered as an isotropic material with the same conductivity of 145 W/m-K (6.98 Btu/hr-in-°F) as the unclad MMC in this SAR. The composition of the core in the aluminum clad MMC is identical to the composition of the regular MMC with no clad. A conservative conductivity of 145 W/m-K (6.98 Btu/hr-in-°F) is selected for the regular (unclad) MMC as stated in SAR Section 4.2.(9). ASME code [ASME Code Section II, Part D - Properties, 2004 with 2006 Addenda] shows the thermal conductivity of aluminum clad at 400° F is in the same range as 145 W/m-K (6.98 Btu/hr-in-°F) for Aluminum series 5000 or series 6000. Therefore, assuming isotropic conductivity for the aluminum clad MMC and the core is appropriate and thermal analyses results provided for the Amendment 1 application remain unaffected.

CHAPTER 5 – SHIELDING EVALUATION

5.1 **(SAR Section 5.1)** Show that the source-term for fuel assemblies with less than 1.5 wt% U-235 is bounded by the design-basis assembly.

The third paragraph on page 5-2 of the SAR states that "fuel assemblies with enrichment between 0.2 wt % U-235 and 1.5 wt % U-235 are qualified by limiting their burnup ... (ensuring) that the shielding analysis is also bounding for these fuel assemblies." Furthermore, the discussion at the top of page 5-4 of the SAR describes burnup limits to maintain gamma and neutron source terms within the limiting assembly design. No reference or analysis is provided as a basis for this assumption.

Present a source-term comparison of the design-basis assembly and the low-enriched fuel assembly in question at the burnup indicated. A complete description of the analysis will include, with justification, assumed bounding dimensions, burnup, power history, cross section libraries used, code versions, and material information.

This information is necessary to verify compliance with 10 CFR 72.104.

Response to 5.1

The fuel qualification methodology employed in the calculation of the design basis source terms is based on the use of SAS2H to rank the source terms due to the various allowable combinations of burnup, enrichment and cooling time (FQT parameters).

A new Section 5.2.3 is added to the SAR to provide a detailed description of the fuel qualification calculations and methodology employed to determine the design basis source term. The results of this section include the various evaluations performed to rank the candidate FQT parameters for their effect on the HD System dose rates. These results also include the evaluations performed for the low enriched fuel assemblies (enrichment between 0.2 wt % U-235 and 1.5 wt % U-235) as part of the scope for this amendment.

5.2 **(SAR Section 5.2)** Provide detailed information on the SAS2H analysis of assemblies with an average enrichment below 1.5 wt%.

The Fuel Qualification Table appears to apply to all classes of fuel assemblies. The depleted uranium assembly (0.2 wt% U-235) was intended for a specific fuel design burned approximately to 5,000 MWd/MTU, yet the allowable burnup has been greatly extended to 20,000 MWd/MTU. Using the methods explicitly described in the amendment and the sample input deck in the original SAR, staff confirmatory analysis of such an assembly cooled for the minimum five year period allowed by the FQT did not support the applicant's statement at the top of page 5-4 of the SAR, that a 0.2 wt% enriched assembly at 20 GWd/MTU and cooled for the minimum five years would be bounded by the design-basis assembly. Further, information methods and assumptions are not included in the amendment and the FQT limits are potentially non-conservative.

An analysis at the boundary of enrichment categories is necessary to demonstrate that the allowable enrichment and burnup levels, when permitted to all fuel classes, are still bounded by the design basis assembly as stated in the application.

This information is necessary to verify compliance with 10 CFR 72.126.

Response to 5.2

A detailed description of the fuel qualification calculations and methodology employed to determine the design basis source terms is provided in new Section 5.2.3 of the SAR. This section contains all the details required to model the fuel assemblies with low enrichment and the results of the evaluation are also included.

5.3 **(SAR Section 5.2)** Justify the enrichment chosen for the bounding design-basis assembly.

The source term analysis on page 5-3 is based on a design bounding assembly determination by initial heavy metal alone. In the example given, an initial enrichment of 4 wt% U-235 is not conservative according to NUREG-1536, "Standard Review Plan for Dry Cask Storage," which states: "the shielding source term ... should be based on the lowest enrichment (for a given burnup)." With the changes to the FQT to more generalized bounding parameters, it is appropriate for all analyses in the application to follow the same thought. Since the cask contents are no longer restricted, specific designs with set heavy metal content and enrichment that can be analyzed, bounding conditions need to be taken into account at all steps of the source-term analysis. For this specific assembly class and enrichment, the design basis analysis should utilize 2.5 wt% U-235. Include such an analysis for each enrichment group in the FQT, or limit the burnup below the initial enrichment analyzed.

This information is necessary to verify compliance with 10 CFR 72.126.

Response to 5.3

A detailed description of the fuel qualification calculations and methodology employed to determine the design basis source terms is provided in new Section 5.2.3 of the SAR. This section contains all the details required to ensure that the source terms utilized in the design basis shielding calculations is bounding.

5.4 **(SAR Section 5-3 and Table 2-1)** Show that MK BW 17x17 is still bounding with the new limits allowed in the FQT.

The reason given for using the MK BW 17x17 as the design basis assembly was the higher heavy metal content. Given that all assemblies have the same maximum heavy metal limit, justify the continued use of the MK BW 17x17 as a design basis assembly.

This information is necessary to verify compliance with 10 CFR 72.126.

Response to 5.4

The design basis source terms utilized for the shielding calculations are calculated with the MK BW 17x17 as the design basis assembly since it has the highest heavy metal loading of 0.476 MTU. However, the fuel assembly hardware used in the source term calculations are based on the Westinghouse 17x17 standard fuel assembly because it has the highest material mass (e.g., steel, inconel) which contributes to the gamma

source terms. Therefore, the design basis fuel assembly is a hybrid MK BW 17x17 fuel assembly with a heavy metal loading of 0.476 MTU containing the hardware of the Westinghouse 17x17 standard fuel assembly. The MTU loading is the most important parameter in the calculation of source terms. Other small differences due to the differences in the fuel pin geometry have no significant effect on source terms. Therefore, the design basis source terms will remain unchanged as long as the maximum MTU loading remains unchanged.

5.5 **(SAR Section 5.2)** Show that the enrichments used for the Thimble Rod Assemblies (TPAs) and the Burnable Poison Rod Assemblies (BPRAs) source term analyses are conservative.

The TPA and BPRA source term analyses are conducted with different initial enrichments of the fuel source term. Describe the basis for the different initial enrichments. Justify the choice of 3.5 wt% U-235 as the bounding calculation for these control components.

This information is necessary to verify compliance with 10 CFR 72.126.

Response to 5.5

The control component (CC) source terms are calculated in SAS2H by irradiating the CC materials in a host fuel assembly using the light element composition of the CC. The "light element" component of the SAS2H calculation results are the CC source terms for use in the shielding calculations. The use of an enrichment and burnup is only for the purpose of providing a representative "spectrum" and duration for irradiation. The bounding or conservative nature of the CC source term calculation is in the choice of cooling time of 4 days. This is highly conservative since a typical cooling time is 5 years.

Therefore, the choice of the host assembly enrichment utilized in the CC source term calculations is appropriate and the calculated source terms (with a cooling time of 4 days) bounding.

5.6 **(Editorial - SAR Page 5-4)** This discussion on reconstituted fuel assemblies could be more clearly articulated. Please rewrite to clarify the cooling requirements.

The third paragraph on page 5-4 of the SAR, states: "There is no limit on the number of rods that are reconstituted with unirradiated stainless steel or Zircalloy or low enriched UO2." Does this mean that both Zircalloy and low-enriched UO2 may be irradiated or is the adjective 'unirradiated' intended to apply to all materials in that list? Please rewrite to clarify.

The same paragraph also states "that for cooling times less than 10 years, 1 year of cooling time is added." This could be interpreted as permitting an assembly to be treated as if it were cooled an additional year in the pool. The discussion is revisited in the fifth paragraph with improved clarity. Please rewrite the third paragraph on page 5-4 of the SAR to clarify. One possibility would be to move the discussion in the fifth paragraph to the third paragraph.

Response to 5.6

The discussion on reconstituted fuel assemblies on Page 5-4 of the SAR is modified for enhanced clarity. Further, a discussion on the methodology for the qualification of reconstituted fuel assemblies is provided in new Section 5.2.3 of the SAR.

5.7 **(Editorial)** The discussion on possible streaming paths could be more clearly articulated.

The **sixth paragraph on page 5-2** states: "Locations where streaming could occur are discussed in Chapter 10." The staff could find no specific mention of streaming paths in Chapter 10 of the FSAR. Compliance with Regulatory Position 2b of Regulatory Guide 8.8 is stated in Chapter 10 as simply: "Regulatory Position 2b on radiation shielding is met by the heavy shielding of the NUHOMS System which minimizes personnel exposures." The presence of heavy shielding alone does not necessarily preclude streaming paths through joints, vents, gaps, etc. There is a later discussion on locations of the maximum dose rates, which sufficiently addresses the issue; however it is not immediately obvious to look there. It is recommended that this level of detail be provided on page 5-2 from Chapter 10 of the SAR.

Response to 5.7

Chapter 5, Section 5.1 of the SAR is revised as recommended by the staff.

5.8 **(Editorial)** Since this amendment is specifically intended to add the CE 16x16 fuel, it should be included in the discussion in a manner consistent with other authorized cask contents.

Revise **final paragraph on page 5-3** to include the 16x16 fuel in bounding assembly comparison.

Material information on the CE 16x16 class, similar to that found in Tables 5-7 through 5-9 would be very useful for the confirmatory analysis.

Response to 5.8

Chapter 5 of the SAR is modified to include CE 16x16 fuel in the discussions in Section 5.0 (authorized contents) and Section 5.2 (design basis fuel assembly).

Material information on the CE 16x16 class is provided in revised SAR Table 5-7 and Table 5-9.

CHAPTER 8 – OPERATING PROCEDURES

8.1 **(Technical Specifications)** Incorporate wording, by reference, from Section 8.1.1.3 of the SAR into the proposed TS such that:

a) Hydrogen gas monitoring or mitigation measures will be conducted when performing any lid welding or cutting operations.

b) All welding operations will be stopped and the DSC cavity will be purged with helium if the hydrogen concentration exceeds 2.4%.

Hydrogen gas may be evolved during wet loading (or unloading) operations and must be monitored or controlled to preclude the possibility of creating a flammable mixture inside the canister during welding or cutting operations.

This information is required for compliance with 10 CFR 72.166.

Response to 8.1

Technical Specification 5.6, "Hydrogen Gas Monitoring," has been created to meet the intent of this RAI. SAR operating procedure Sections 8.1.1.3 and 8.2.2 are also revised to include references to this new technical specification.

CHAPTER 9 – MATERIALS EVALUATION (ACCEPTANCE TESTS AND MAINTENANCE PROGRAM)

9.1 **(Technical Specifications)** Incorporate Sections 9.5.3.2, 9.5.3.4, 9.5.3.6, 9.5.3.7, 9.5.4.1, and 9.5.4.2 of the SAR into the proposed Technical Specifications by reference. Assuming that clarifications (see RAI 9.3) are made, Section 9.5.3.3 should also be incorporated in the proposed TS by reference.

The acceptance testing of all neutron poisons used in a spent fuel canister is important to the safety of the NUHOMS HD package and should not be amendable under 10 CFR Part 72.48 without approval by NRC staff.

This information is required for compliance with 10 CFR Part 72.44(c)(4) and 10 CFR 72.236(b).

Response to 9.1

The present extent of the incorporating Chapter 9 into the Technical Specifications was established at the request of the NRC starting with the Amendment 8 to CoC 1004 and the original application for CoC 1030, and has been maintained consistently through Amendment 1 to CoC 1027 and Amendment 11 of CoC 1004. The intent of including part of this chapter into the Technical Specifications was to provide some fixed bounding description of the types of neutron poison materials that Transnuclear could qualify and use, lacking any consensus standards on such materials, and recognizing that new neutron poison materials of this kind are entering as well as leaving the market.

One of the reasons of aligning all NUHOMS® products in various CoCs on the subject of neutron absorbers was to reduce the likelihood of errors that could result from lack of standardization. Significant changes, other than correction of errors, in these sections will result in specifications for neutron absorbing materials on different products and CoCs diverging from one another. The fact that an activity is important to safety means that it is controlled under TN's QA program – not that it goes into the Technical Specifications.

To bring this SAR into alignment with the others referenced, Transnuclear proposes the following revisions in Section 9.5.2:

> The B10 areal density is measured using a collimated thermal neutron beam of up to 1.2 centimeter diameter. A beam size greater than 1.2 centimeter diameter but no larger than 1.7 centimeter diameter may be used if computations are performed to demonstrate that the calculated $k_{effective}$ of the system is still below the calculated Upper Subcritical Limit (USL) of the system assuming defect areas the same area as the beam. *[delete the subsequent sentence beginning "Alternatively…."]*
>
> …The lower tolerance limit of B10 volume density is then determined, defined as the mean value of B10 volume density for the sample, less K times the standard deviation, where K is the one-sided tolerance limit factor ~~for a normal distribution~~ with 95% probability and 95% confidence. *If a goodness-of-fit test demonstrates that the sample comes from a normal population, the value of K for a normal*

distribution may be used. Otherwise, use a non-parametric (distribution-free) method of determining the one-sided tolerance limit.

TN also requests correction of an error at Section 9.5.2, paragraph 8, second sentence. Delete the word "minimum" twice: " The ~~minimum~~ B10 areal densities determined by neutron transmission are converted to volume density, i.e., the ~~minimum~~ B10 areal density is divided by the thickness at the location of the neutron transmission measurement or the maximum thickness of the coupon. *"Minimum" in this context does not refer to the minimum SPECIFIED areal density, and therefore its meaning is undefined.*

9.2 **(SAR Section 9.5.3.4)** Delete the statement in Section 9.5.3.4(a) that scanning electron microscopy (SEM) may be used as an alternative method to evaluate the ductility of MMCs.

Due to the heterogeneous nature of MMCs, microscopic strain observed using SEM may not be used to reliably estimate macroscopic ductility. As such, room temperature tensile tests measuring the elongation to failure of the MMCs (both clad and unclad), as stated in Section 9.5.3.4(a), should be conducted.

The Staff considers qualitative acceptance tests, such as visual inspection by SEM or American Society for Testing and Materials (ASTM) E290, in general, not acceptable for verifying the mechanical properties of neutron absorbing materials.

This information is required for compliance with 10 CFR 72.124(b) and 10 CFR 72.236(c).

Response to 9.2

Transnuclear agrees with the deletion of SEM, but disagrees with the claim regarding the use of the ASTM E290. Such qualitative testing might not be appropriate if Transnuclear were using the MMC's mechanical properties in the structural analysis of the basket. This is not the case. The neutron absorbing materials are considered in the structural analysis only as transmitters of bearing loads across their thickness, and as a dead weight. No structural credit is taken by Transnuclear from these neutron absorber materials. Therefore, the purpose of mechanical testing is not to demonstrate that the material has the properties assigned to it for analysis, but only to demonstrate that the material has reasonable physical integrity, and will not shatter under the design conditions.

We therefore propose to replace the existing wording with the wording that has been accepted by the Staff on previous applications. Delete "(Alternatively show that the material fails in a ductile manner, e.g., by scanning electron microscopy of the fracture surface or by bend testing.)" and replace with "As an alternative to the elongation requirement, ductility may be demonstrated by bend testing per ASTM E290[10]. The radius of the pin or mandrel shall be no greater than three times the material thickness, and the material shall be bent at least 90 degrees without complete fracture."

9.3 **(SAR Section 9.5.3.3, and Technical Specifications)** Clarify in Section 9.5.3.3 (and

incorporate into the Technical Specifications by reference) the conditions which necessitate that neutron absorbing plate materials will undergo corrosion and thermal damage testing if there has been a key process change, as defined in Section 9.5.4.3.

a) It is unclear if corrosion testing is required for the neutron absorbing plate material if the neutron absorbing material used is an alloy other than 1100 aluminum.

b) If the neutron absorbing plate material consists only of boron carbide and 1100 aluminum, clarify if an increase of porosity will require qualifying corrosion and thermal damage tests. If applicable, justify why an increase of porosity for a boron carbide and 1100 aluminum-based neutron absorber would not require qualification testing.

The corrosive effects of a spent fuel pool environment and the subsequent drying step may lead to cracking and/or delaminating of the aluminum clad metal matrix composites As such, appropriate qualifying corrosion and thermal damage tests should be required for neutron absorbing materials when there has been a change in the aluminum alloy and/or an increase in porosity.

This information is required for compliance with 10 CFR 72.120(d), 10 CFR 72.124(b) and 10 CFR 72.236(c).

Response to 9.3

Instead of incorporating entire sections into the Technical Specifications, Transnuclear proposes to relocate the density requirements, demonstrated by qualification testing, from SAR Section 9.5.3.1 to Section 9.1.7.2, in order to incorporate them into the Technical Specifications. This is consistent with the original intent to provide a bounding material description in lieu of ASTM or other consensus material specifications. See the response to RAI question 9.1 above

Transnuclear agrees with a), and proposes the following clarification:

> *Delete*
>
> *The need for thermal damage and corrosion (hydrogen generation) testing shall be evaluated case-by-case based on comparison of the material composition and environmental conditions with previous thermal or corrosion testing of MMCs.*
>
> *And replace with*
>
> *Thermal damage and corrosion (hydrogen generation) testing shall be performed unless such tests on materials of the same chemical composition have already been performed and found acceptable. The following paragraphs illustrate two cases where such testing is not required.*

Transnuclear disagrees with suggested change b). Paragraph 9.5.3.4(b) limits the total porosity to 3% maximum, and paragraph 9.5.3.1 limits the interconnected porosity to essentially zero. Isolated porosity will not lead to the problems of water intrusion, cracking, and delamination suggested. These are problems that have occurred with a cermet compacted to less than 95% of full density, and using a much larger boron carbide particle size distribution, and much higher boron carbide content than the

present FSAR would allow for clad MMCs. Clad MMCs are different than Boral© and behave differently.

9.4 **(SAR Section 9.5.4.3)** Specify that for aluminum clad MMCs, significant changes (which should be quantified by the applicant) of the internal core thickness relative to the thickness of the aluminum cladding qualify as a major process change according to Section 9.5.4.3 of the Safety Analysis Report.

The aluminum cladding is expected to be significantly more ductile than the boron carbide reinforced inner core of the aluminum clad MMCs. The relative thickness of the aluminum cladding to the internal core thickness will greatly influence the mechanical properties of the final material.

This information is required for compliance with 10 CFR 72.124(b) and 10 CFR 72.236(c).

Response to 9.4

Transnuclear Agrees and proposes the following addition:

(g) For MMCs with an integral aluminum cladding, a change greater than 25% in the ratio of the nominal aluminum cladding thickness (sum of two sides of cladding) and the nominal matrix thickness could result in changes in the mechanical properties of the final product.

9.5 **(SAR Sections 9.1.7.3 and 9.5.3.4)** Describe how the edges of the neutron poison Plates of aluminum clad neutron absorbers will be adequately sampled for flaws to ensure that edge cracks will be detected.

Cracks may form at the edges of aluminum clad neutron absorbers or on plates with high boron carbide volume fractions (> 40%) during rolling, which may result in localized regions deficient in boron carbide. This is of particular concern in the case of aluminum clad MMCs, where edge cracking may not be detectable by visual inspection.

This information is required for compliance with 10 CFR 72.124(b) and 10 CFR 72.236(c).

Response to 9.5

Neutron absorber edges are in practice not sampled, but are 100% visually inspected.

Clad MMCs are manufactured by extruding or rolling material that is compacted in a can. The result is that the edges of the master sheet are solid aluminum, and are not particularly susceptible to cracking. The greater concern is that the solid portions are fully removed in cutting the finished absorber pieces from the master sheet. Therefore, Transnuclear proposes the following revisions to SAR Sections 9.1.7.1, 9.1.7.2 and 9.1.7.3.

9.1.7.3: Boral. Replace '"Visual inspections shall be performed verify that the Boral® core is not exposed through the face of the sheet at any location" with "Each finished

Boral piece shall be visually inspected on faces and edges. Any piece with cracks through the cladding, exposed core on the face of the sheet, or solid aluminum at the edge of the sheet will be treated as non-conforming."

The purpose of NRC's reference to SAR Section 9.5.3.4 is not clear. Sections 9.1.7.1 and 9.1.7.2 cover visual inspections for borated aluminum and metal matrix composites respectively. These paragraphs are revised to enhance by specifying 100% visual inspection, which is the current practice. The paragraphs on MMCs are similarly expanded to require that the matrix does not show through the face, and that the edges don't show solid aluminum. Transnuclear has revised the following Sections:

9.1.7.1: Borated Aluminum. Add "Each finished borated aluminum piece shall be visually inspected on both faces"

9.1.7.2: MMC. Add "Each finished borated aluminum piece shall be visually inspected on both faces," and "MMCs with an integral aluminum cladding shall also be inspected on the edges, and visual inspection shall include verification that the matrix is not exposed through the faces of the aluminum cladding and that solid aluminum is not present at the edges"

9.6 **(SAR Section 9.5.3.1)** Clarify if the 3% maximum permissible porosity for the aluminum clad metal matrix composites (MMCs) pertains to the open, closed, or total porosity of the MMCs. Quantitatively describe the pore size distribution in these MMCs.

The porosity of the MMCs influences the mechanical strength and corrosion resistance of the MMCs. Pores greater than the average boron carbide particle size, or smaller, interconnected pores (even if enclosed), can affect the homogeneity of the boron carbide distribution in the neutron absorbing material. This, in turn, could lead to neutron streaming, and a reduction in the neutron absorbing effectiveness of the MMCs.

This information is required for compliance with 10 CFR 72.124(b) and 10 CFR 72.236(c).

Response to 9.6

The 3% refers to total porosity. Paragraph 9.5.3.4(b) limits the interconnected porosity to essentially zero. Pore size distribution is not known. Pores of sufficient size and quantity to effect neutron transmission sufficiently to increase k_{eff}, are highly unlikely in a material that is at or above 97% of theoretical density with the limitations on boron carbide particle size and percentage that Transnuclear imposes. In any event, such a phenomenon, if it was significant, would be picked up by the neutron attenuation testing and statistical analysis. Because neutron attenuation testing uses a collimated neutron beam, if there is any streaming, it results in higher transmission, and correspondingly lower areal density measurement. That is, neutron attenuation testing by its very nature accounts for any streaming in its results – it measures the "effective" B10 areal density.

Note that no structural credit is being taken for this material's mechanical properties in the stress and buckling analysis of the basket. Any effect of pore size and distribution on mechanical properties is accounted for in the results of tensile or bending testing. As discussed above, the effect of pores on corrosion is insignificant when they are isolated pores.

9.7 **(SAR Section 9.5.3.1)** Specify how the aluminum clad neutron absorbers will be visually inspected and the qualifications of the individuals performing the inspection to ensure that the aluminum cladding is intact and that the internal core is not exposed to the spent fuel pool environment.

According to Section 9.5.3.4, "testing or examination for exposed interconnected porosity shall be performed by a means to be approved by the Certificate Holder." The Staff requests that the applicant specify how the aforementioned testing or examinations will be conducted.

This information is required for compliance with 10 CFR 72.124(b) and 10 CFR 72.236(c).

Response to 9.7

It is not clear how this question applies to SAR Section 9.5.3.1, which is applicability and scope of MMC qualification testing, and is not connected to visual inspection for acceptance.

Because clad MMCs will not have interconnected porosity, and will be at least 97% of theoretical density, there is no particular concern if their matrix is exposed, but Transnuclear has nonetheless proposed to add the "no exposed matrix through the face of the aluminum cladding" requirement to the visual inspections in response to RAI 9.5 above.

Visual examinations are performed according to SAR Sections 9.1.7.1, .2, and .3. The inspector must, of course, understand the acceptance criteria, but because the visual inspections do not involve characteristics of the material critical for design performance, no special qualifications are required. This is consistent with the surface inspection of any steel or aluminum mill plate under ASTM specifications.

Regarding Section 9.5.3.4 testing for interconnected porosity, to date, Transnuclear has accepted results by ASTM B328-96. This test method is widely used in developing self-lubricating powder metallurgy parts, which requires a quantified measure of a material's interconnected porosity, and commonly referenced in standard specifications for powder metallurgy structural parts, e.g., ASTM B595. Transnuclear finds this test method suitable for quantifying the interconnected porosity of MMCs.

9.8 **(SAR Section 9.5.3.3)** Demonstrate that the aluminum clad metal matrix composites (MMCs) are adequately resistant to the combined affects of corrosion and heating which the MMCs are expected to see during loading and drying of the spent fuel canister.

The aluminum clad metal matrix composites proposed in the amendment are different enough from the unclad material, that qualifying corrosion and thermal damage testing should be considered. The corrosive affects of a spent fuel pool environment in combination with subsequent drying may produce a synergistic effect leading to cracking and/or delaminating of the aluminum clad metal matrix composites.

This information is required for compliance with 10 CFR 72.120(d), 10 CFR 72.124(b) and 10 CFR 72.236(c).

Response to 9.8

The phenomena described are due not so much to corrosion as the intrusion of water into the porous structure at the edges of a coarse, low compaction cermet under conditions of hydrostatic testing, and the subsequent inability of the water to escape rapidly enough when subjected to rapid heatup characteristic of thermal testing.

By specifying a material with much finer boron carbide particles, much higher density, and no interconnected porosity, Transnuclear is specifying a material that is essentially different than Boral®, and would not be subject to water intrusion and the associated phenomena.

9.9 **(SAR Section 9.5.3.4)** Provide statistically significant qualifying data demonstrating that the aluminum clad metal matrix composites meet the minimum mechanical properties specified in Section 9.5.3.4 and the porosity requirements specified in Section 9.5.3.1 (see RAI 9-6).

The aluminum clad metal matrix composites proposed in the amendment are different enough from the unclad material, that qualifying mechanical testing and porosity measurements should be considered.

This information is required for compliance with 10 CFR 72.124(b) and 10 CFR 72.236(c).

Response to 9.9

The purpose of the qualification tests specified in SAR Section 9.5.3.4 is to allow Transnuclear to evaluate candidate materials in accordance with an approved plan with clear acceptance criteria, and to accept such materials independently. Note that the qualification test data from one MMC would be irrelevant for the other MMC, or even for a later version of the same MMC with a higher percentage of boron carbide.

Because we are not taking credit for the mechanical properties of the material in our structural analysis, there is no need to perform a statistically significant number of tensile tests. The testing specified in SAR Section 9.5.3.4 is intended only to show that the process for making the material is capable of providing a product that has some minimal strength and ductility.

9.10 **(Technical Specifications)** In the Technical Specifications, incorporate by reference controlling documents which link the fabrication details of the neutron absorbing materials outlined in Section 9 of the Safety Analysis Report to the procedures which were used to produce the originally qualified neutron absorbing materials.

The fabrication procedures for the neutron absorbers intended for service in spent nuclear fuel storage casks should be the same (unless otherwise described) as those used to produce the neutron absorbers for qualification testing.

This information is needed to assure compliance with 10 CFR 72.44(c)(4).

Response to 9.10

It would be impossible for Transnuclear to include the proprietary process documents of suppliers in the Technical Specifications. Furthermore, this would defeat the entire purpose of the qualification and key process controls program, which is to allow Transnuclear to evaluate and accept neutron absorber materials for use without going through the amendment process for each specific material.

The key process controls outlined in the SAR Section 9.5.4.3 provide the necessary technical and quality control over any process changes, while providing flexibility to improve the product. Placing supplier process controls in the Technical Specifications would eliminate practical opportunities for improvements in the product, and make this key process controls section essentially useless. Note that the SAR Section 9.5.4.3 is already incorporated in the Technical Specification by reference and therefore not subjected to change without the Staff approval.

Transnuclear, along with the NRC and other industry representatives, participated in the development of ASTM C 1671. SAR Chapter 9 was developed in parallel with, and is largely consistent with that standard, including the section on key process controls.

9.11 **(SAR Section 9.5.3.4)** Justify the use of ASTM B311 as a testing method for measuring porosity of aluminum clad MMCs.

ASTM B311 is a test method used for density determination of powder metallurgy materials containing less than two percent porosity, yet the maximum permissible porosity of the aluminum clad MMCs exceeds two percent.

This information is required for compliance with 10 CFR 72.124(b) and 10 CFR 72.236(c).

Response to 9.11

The specified 2% porosity limit ensures that during water immersion the specimen does not gain mass from absorbed water and cause density values higher than the true value. For this reason, ASTM B311 states that specimens with high porosity levels must have their surfaces sealed. Transnuclear procurement specifications do not allow interconnected surface porosity at the surface of the material. In addition to testing for interconnected porosity by ASTM B328-96 described in response to RAI 9.7, Transnuclear or the MMC supplier verify by microscopic examination that the small presence of internal voids are not interconnected to one another nor connected to the surface, essentially confirming that there is no interconnected porosity. Products without interconnected porosity can be tested by ASTM B311 because water has no path inward to occupy the internal voids, including the case of a 97% dense MMC with an integral aluminum cladding. Therefore, density test by ASTM B311 is acceptable.

9.12 **(Licensing Drawings)** Clarify or remove the term, “or equivalent” from the licensing drawing 10494-72-1 when referring to “SA240, Type 304 Steel.”

Any material equivalent to SA240, Type 304 steel should have identical or superior mechanical properties, and an accompanying level of quality assurance identical or superior to materials meeting ASME Code requirements.

This information is needed to determine compliance with 10 CFR 72.236(b).

Response to 9.12

"or equivalent" is removed from the drawings (10494-72-1 and 10494-72-2003-SAR Sheet 1 of 5) and a note is added to these drawings. It states that "Altemate material specifications (plate, bars, or forging) to those specified may be used provided mechanical properties are equal to or greater than material specified and chemical composition is the same."

9.13 **(Licensing Drawings)** Detail or remove the reference to the "alternate weld configuration" in Note 4 on licensing drawing 10494-72-5 and the "alternate equivalent weld detail" on Note 1 on licensing drawing 10494-72-17. Alternatively, demonstrate that these welds have no safety significance.

Any alternate welding configuration must be described adequately, such that a structural evaluation of the weld can be conducted.

This information is needed to determine compliance with 10 CFR 72.236(b).

Response to 9.13

Note 4 on drawing 10494-72-5 and Note 1 on drawing 10494-72-17 are removed from both drawings.

9.14 **(Editorial)** Edit Section 9.5.3.4(b) of the Safety Analysis Report (SAR) so that the maximum permissible porosity of the aluminum clad metal matrix composites is consistent throughout the SAR.

The applicant should consider amending 9.5.3.4(b) of the SAR so that the porosity (see RAI 9-6) of aluminum clad metal matrix composites may not exceed 3% (rather than 2%) to be consistent with the proposed changes in Amendment 1 to the application.

Response to 9.14

SAR Section 9.5.3.4(b) is revised to clarify the requirements for non-clad and clad-MMCs.

9.15 **(Editorial)** Clarify the term "full density" as referred to in Section 9.5.3.3 of the SAR.

The applicant may be referring to those MMCs which pass the density acceptance criteria in Section 9.5.3.4(b), but the term "full density" is usually applied to materials which are ≥ 99.9% of theoretical density.

Response to 9.15

In the SAR Section 9.5.3.3, the words "full density" is deleted. In context, it is clear that only MMCs which meet the other specifications in the section are intended.

9.16 (Editorial) The title of Section 9.5.3.4 should be clarified.

Section 9.5.3.4 does not refer to a qualification test, but rather an acceptance test of samples manufactured during a production run. It is suggested that the applicant rename the title of Section 9.5.3.4 to a title that reflects this more clearly, (e.g., "Required Acceptance Tests and Examinations to Insure Mechanical Properties").

Response to 9.16

Tensile tests are not required for acceptance during production runs. They are only performed once during qualification to verify that the production process results in a durable, non-brittle product. Note that these materials are not being used with structural credit as described in response to RAIs 9.2 and 9.6.

CHAPTER 10 – RADIATION PROTECTION

10.1 **(Editorial)** The first complete bullet point on page 10-6 incorrectly lists the neutron and gamma ray spectra to be shown in "Table 10-."

Include the correct table number on page 10-6.

Response to 10.1

SAR Chapter 10 is modified to correct the following typographical errors:

Line 1, 3rd paragraph, Section 10.2.2, page 10-4, "Table 10-" changed to "Table 10-2"
Line 1, 1st bullet, Section 10.2.2, page 10-6, "Table 10-" changed to "Table 10-3"
Line 1, 6th paragraph, Section 10.2.2.2, page 10-6, "Table 10-" changed to "Table 10-5"
Line 2, 6th paragraph, Section 10.2.2.2, page 10-6, "Table 10-" changed to "Table 10-6"

List of Changed Pages for CoC 1030 Amendment 1 Application Revision 1

Note: In both the TS and the UFSAR, both Amendment 1 Revision 0 changes and Amendment 1 Revision 1 changes are shown, using italicized text and revision bars; however, Revision 1 changes are shaded to distinguish them from Revision 0 changes.

Page	Associated RAI
Tech Specs Page ii	RAI 8.1
Tech Specs Page 1-1	RAI 1.3
Tech Specs Page 1-2	RAI 1.4
Tech Specs Page 2-1	RAIs 1.3 and 1.4
Tech Specs Page 5-7	RAI 3.1
Tech Specs Page 5-8	RAI 3.1
Tech Specs Page 5-11 (new)	RAI 8.1
SAR Drawing 10494-72-1	RAI 9.12
SAR Drawing 10494-72-5	RAI 9.13
SAR Drawing 10494-72-17	RAI 9.13
SAR Drawing 10494-72-2003-SAR (5 sheets)	RAI 9.12
SAR Page 2-2	RAI 2.2
SAR Page 2-3	RAI 2.2
SAR Page 2-21	RAI 2.2
SAR Page 4-i	Updated table of contents for new Section 4.8.6
SAR Page 4-43	RAI 4.1
SAR Page 4-47A (new)	RAI 4.1
SAR Page 4-47B (new)	RAI 4.1
SAR Page 4-47C (new)	RAI 4.1
SAR Page 4-74	RAI 4.1
SAR Page 5-i	Updated table of contents for new Section 5.2.3
SAR Page 5-iii	Updated table of contents for new Tables 5-24 and 5-25
SAR Page 5-1	RAI 5.8
SAR Page 5-2	RAIs 5.1, 5.2, 5.3, 5.7
SAR Page 5-3	RAI 5.8
SAR Page 5-4	RAIs 5.6, 5.8
SAR Page 5-5	RAI 2.3
SAR Page 5-6	RAIs 5.4, 5.8
SAR Page 5-6A (new)	RAIs 5.1, 5.2, 5.3, 5.6
SAR Page 5-6B (new)	RAIs 5.1, 5.2, 5.3, 5.6
SAR Page 5-6C (new)	RAIs 5.1, 5.2, 5.3, 5.6
SAR Page 5-6D (new)	RAIs 5.1, 5.2, 5.3, 5.6
SAR Page 5-6E (new)	RAIs 5.1, 5.2, 5.3, 5.6
SAR Page 5-57	RAIs 5.1, 5.2, 5.3
SAR Page 5-58 (new)	RAIs 5.1, 5.2, 5.3
SAR Table 5-6 and 5-7 (Page 1 of 2)	RAIs 5.4, 5.8
SAR Table 5-7 (Page 2 of 2) (new)	RAIs 5.4, 5.8
SAR Table 5-9	RAIs 5.4, 5.8
SAR Table 5-24 (Page 1 of 4) (new)	RAIs 5.1, 5.2, 5.3
SAR Table 5-24 (Page 2 of 4) (new)	RAIs 5.1, 5.2, 5.3
SAR Table 5-24 (Page 3 of 4) (new)	RAIs 5.1, 5.2, 5.3
SAR Table 5-24 (Page 4 of 4) (new)	RAIs 5.1, 5.2, 5.3

List of Changed Pages for CoC 1030 Amendment 1 Application Revision 1

Note: In both the TS and the UFSAR, both Amendment 1 Revision 0 changes and Amendment 1 Revision 1 changes are shown, using italicized text and revision bars; however, Revision 1 changes are shaded to distinguish them from Revision 0 changes.

Page	Associated RAI
SAR Table 5-25 (new)	RAIs 5.1, 5.2, 5.3
SAR Page 8-3	RAI 4.3
SAR Page 8-4	RAI 8.1
SAR Page 8-12	RAI 8.1
SAR Page 9-3	RAI 9.5
SAR Page 9-4	RAIs 9.3 and 9.5
SAR Page 9-5	RAI 9.5
SAR Page 9-6	No changes, but information shifted onto this page
SAR Page 9-8	RAI 9.1
SAR Page 9-9	RAI 9.1 and 9.3
SAR Page 9-10	RAIs 9.2 and 9.15
SAR Page 9-11	RAIs 9.2 and 9.14
SAR Page 9-13	RAI 9.4
SAR Page 10-4	RAI 10.1
SAR Page 10-6	RAI 10.1
SAR Page 12-1	RAI 1.2
SAR Page B12-10	RAI 2.1

Enclosure 5 to TN E-27377

NUHOMS® HD Amendment 1 Application Revision 1, Changed and New Proposed Technical Specifications and Proposed Updated Final Safety Analysis Report Pages (Public Version)

List of Tables

List of Figures

1.0 USE AND APPLICATION

1.1 Definitions

-- NOTE --

The defined terms of this section appear in capitalized type and are applicable throughout these Technical Specifications and Bases.

Term	Definition
ACTIONS	ACTIONS shall be that part of a Specification that prescribes Required Actions to be taken under designated Conditions within specified Completion Times.
HORIZONTAL STORAGE MODULE (HSM-H)	The HSM-H is a reinforced concrete structure for storage of a loaded 32PTH DSC at a spent fuel storage installation.
DAMAGED FUEL ASSEMBLY	A DAMAGED FUEL ASSEMBLY is a fuel assembly with known or suspected cladding defects greater than pinhole leaks or hairline cracks and which can be handled by normal means. *Fuel assemblies with damage greater than this can not be stored as damaged fuel assemblies.*
DRY SHIELDED CANISTER (32PTH DSC)	A 32PTH DSC is a welded pressure vessel that provides confinement of INTACT or DAMAGED FUEL ASSEMBLIES in an inert atmosphere.
FUEL CLASS	*A FUEL CLASS includes fuel assemblies of a particular type of fuel design. For example, WEV 17x17, WEO 17x17, and ANP Advanced MK BW 17x17 fuel assemblies of a 17x17 type fuel assembly design are part of a 17x17 fuel class.*
INDEPENDENT SPENT FUEL STORAGE INSTALLATION (ISFSI)	The facility within a perimeter fence licensed for storage of spent fuel within HSM-Hs.
INTACT FUEL ASSEMBLY	Spent Nuclear Fuel Assemblies without known or suspected cladding defects greater than pinhole leaks or hairline cracks and which can be handled by normal means.
LOADING OPERATIONS	LOADING OPERATIONS include all licensed activities on a 32PTH DSC while it is being loaded with INTACT or DAMAGED FUEL ASSEMBLIES, and in a TRANSFER CASK while it is being loaded with a 32PTH DSC containing INTACT or DAMAGED FUEL ASSEMBLIES. LOADING OPERATIONS begin when the first INTACT or DAMAGED FUEL ASSEMBLY is placed in the 32PTH DSC and end when the TRANSFER CASK is ready for TRANSFER OPERATIONS.

1.1 Definitions (continued)

RECONSTITUTED FUEL ASSEMBLY	*A RECONSTITUTED FUEL ASSEMBLY is an INTACT FUEL ASSEMBLY where one or more fueled rods are replaced by rods containing stainless steel rods, stainless steel clad rods, low enriched uranium rods or natural uranium rods or Zircalloy (including other zirconium based alloy) rods or Zircalloy pellets. The nominal volume of the replacement rods is equivalent to the replaced fueled rods in the active fuel region of the fuel assembly.*
STORAGE OPERATIONS	STORAGE OPERATIONS include all licensed activities that are performed at the ISFSI while a 32PTH DSC containing INTACT or DAMAGED FUEL ASSEMBLIES is located in an HSM-H on the storage pad within the ISFSI perimeter.
TRANSFER CASK (TC)	The TRANSFER CASK consists of a licensed NUHOMS® OS187H onsite transfer cask. The TRANSFER CASK will be placed on a transfer trailer for movement of a 32PTH DSC to the HSM-H.
TRANSFER OPERATIONS	TRANSFER OPERATIONS include all licensed activities involving the movement of a TRANSFER CASK loaded with a 32PTH DSC containing INTACT or DAMAGED FUEL ASSEMBLIES. TRANSFER OPERATIONS begin when the TRANSFER CASK is placed *horizontal* on the transfer trailer *ready for TRANSFER* OPERATIONS and end when the 32PTH DSC is located in an HSM-H on the storage pad within the ISFSI perimeter.
UNLOADING OPERATIONS	UNLOADING OPERATIONS include all licensed activities on a 32PTH DSC to unload INTACT or DAMAGED FUEL ASSEMBLIES. UNLOADING OPERATIONS begin when the 32PTH DSC is removed from the HSM-H and end when the last INTACT or DAMAGED FUEL ASSEMBLY has been removed from the 32PTH DSC.

2.0 FUNCTIONAL AND OPERATING LIMITS

2.1 Fuel to be Stored in the 32PTH DSC

PHYSICAL PARAMETERS:	
FUEL CLASS	*Intact or damaged Westinghouse 17x17 (WE 17x17), Westinghouse 15x15 (WE 15x15), Combustion Engineering 16x16 (CE 16x16) and Combustion Engineering 14x14 (CE 14x14) class PWR assemblies (with or without control components) that are enveloped by the fuel assembly design characteristics listed in Table 2. Reload fuel manufactured by the same or other vendors but bounded by the design characteristics listed in Table 2 is also acceptable.*
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 32PTH DSC.* • *Authorized CCs include Burnable Poison Rod Assemblies (BPRAs), Thimble Plug Assemblies (TPAs), Control Rod Assemblies (CRAs), Control Element Assemblies (CEAs), 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.* • *Design basis thermal and radiological characteristics for the CCs are listed in Table 3.*
No. of Intact Assemblies	*≤ 32*

5.3 Lifting Controls

5.3.1 Transfer Cask Lifting Heights

The lifting height of a loaded transfer cask/32PTH DSC, is limited as a function of location, as follows:

a) The maximum lift height and handling height for all TRANSFER OPERATIONS where the TC/32PTH is in the horizontal position on the trailer shall be 80 inches.

b) The maximum lift height of the transfer cask/32PTH DSC shall be restricted by site (10CFR50) limits for all handling operations except those listed in 5.3.1a above. An evaluation of the fuel cladding structural integrity shall be performed for all credible drops under the user's 10CFR50 heavy loads program.

These restrictions ensure that any 32PTH DSC drop as a function of location is within the bounds of the accident analysis.

5.3.2 Cask Drop

Inspection Requirement

The 32PTH DSC will be inspected for damage after any transfer cask *side* drop of fifteen inches or greater.

Background

TC/32PTH DSC handling and loading activities are controlled under the 10CFR 50 license until a loaded TC/32PTH DSC is placed on the transporter, at which time fuel handling activities are controlled under the 10CFR 72 license.

(continued)

5.3 Lifting Controls (*concluded*)

5.3.2 Cask Drop (*concluded*)

Safety Analysis

The analysis of bounding drop scenarios shows that the transfer cask will maintain the structural integrity of the 32PTH DSC confinement boundary from an analyzed side drop height of 80 inches. The 80-inch drop height envelopes the maximum height from the bottom of the transfer cask when secured to the transfer trailer while en route to the ISFSI.

Although analyses performed for cask drop accidents at various orientations indicate much greater resistance to damage, requiring the inspection of the DSC after a *side* drop of 15 inches or greater ensures that:

1. The DSC will continue to provide confinement.

2. The transfer cask can continue to perform its design function regarding DSC transfer and shielding.

5.6 Hydrogen Gas Monitoring

For DSCs, while welding the inner top cover/shield plug during loading operations, and while cutting the outer or inner top cover/shield plug during unloading operations, hydrogen monitoring of the space under the inner top cover/shield plug in the DSC cavity is required, to ensure that the combustible mixture concentration remains below the flammability limit. If this limit is exceeded, all welding operations shall be stopped and the DSC cavity purged with helium to reduce hydrogen concentration safely below the limit before welding or cutting operations can be resumed.

DRAWING NO.	SHEET
10494-72-1	1 OF 1

PARTS LIST

ITEM No.	No. REQ'D	NOMENCLATURE OR DESCRIPTION	MATERIAL SPECIFICATION OR PART NUMBER	SAFETY DESIGNATION	CODE JURISDICTION

SECURITY RELATED INFORMATION
WITHHELD UNDER 10 CFR 2.390

REVISION	DESCRIPTION	DATE
3A	C.O.C. 1030 AMENDMENT 1 APPLICATION	12/11/08
2	REVISED PER FCN 721030-058, 081; UPDATED TITLE BLOCK	09/27/07
1	SEE DCN 10494-39	02/09/05

	NAME / INITIALS	DATE
P.E.	PETER SHIH PS	12/11/08
NUCLEAR	Prakash Narayanan PAN	12/11/2008
MECHANICAL THERMAL	PETER SHIH PS	12/11/08
STRUCTURAL	Raheel Haroon RH	12/11/08
CHECKED	D. GANDOV / DG	12/10/08
DRAWN	TOMMY DUFFY TD	12/10/08

UNLESS OTHERWISE SPECIFIED
ALL DIMENSIONS ARE IN INCHES & DEGREES.

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

TRANSNUCLEAR
AN AREVA COMPANY

SAFETY ANALYSIS REPORT
NUHOMS® 32PTH
TRANSPORTABLE CANISTER FOR PWR FUEL
PARTS LIST

DRAWING NO.	SCALE	SHEET	REVISION
10494-72-1	NONE	1 OF 1	3A

PROPRIETARY AND SECURITY RELATED INFORMATION WITHHELD UNDER 10 CFR 2.390

REVISION	DESCRIPTION	DATE
3A	C.O.C. 1030 AMENDMENT 1 APPLICATION	12/11/08
2	REVISED PER FCN 721030-081, 082; UPDATED TITLE BLOCK	09/27/07
1	SEE DCN 10494-44	02/09/05

	NAME / INITIALS	DATE
P.E.	PETER SHIH / PS	12/11/08
NUCLEAR	Prakash Narayanan / PAN	12/11/2008
MECHANICAL THERMAL	PETER SHIH / PS	12/11/08
STRUCTURAL	Raheel Haroon / RH	12/11/08
CHECKED	D. GANDON / DG	12/11/08
DRAWN	TOMMY DUFFY / TD	12/10/08

UNLESS OTHERWISE SPECIFIED
ALL DIMENSIONS ARE IN INCHES & DEGREES.

TOLERANCES
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SAFETY ANALYSIS REPORT
NUHOMS® 32PTH
TRANSPORTABLE CANISTER FOR PWR FUEL
SHELL ASSEMBLY

DRAWING NO.	SCALE	SHEET	REVISION
10494-72-5	NONE	1 OF 1	3A

DRAWING NO. 10494-72-5 SHEET 1 OF 1

DRAWING NO. 10494-72-17 SHEET 1 OF 1

PROPRIETARY AND SECURITY RELATED INFORMATION WITHHELD UNDER 10 CFR 2.390

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2	REVISED PER FCN 721030-081; UPDATED TITLE BLOCK	09/27/07
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P.E.	PETER SHIH PS	12/11/08
NUCLEAR	Prakash Narayanan PAN	12/11/2008
MECHANICAL THERMAL	PETER SHIH PS	12/11/08
STRUCTURAL	Raheel Haroon / RH	12/11/08
CHECKED	O. GANDOV / OG	12/10/08
DRAWN	TOMMY DUFFY/ TD	12/10/08

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SAFETY ANALYSIS REPORT
NUHOMS® OS187H
ONSITE TRANSFER CASK
TOP COVER ASSEMBLY

DRAWING NO.	SCALE	SHEET	REVISION
10494-72-17	NONE	1 OF 1	3A

SECURITY RELATED INFORMATION

PARTS LIST						
ITEM	QTY	PART OR IDENTIFYING NO.	NOMENCLATURE OR DESCRIPTION	MATERIAL SPECIFICATION	QUALITY CATEGORY	CODE CRITERIA

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MECHANICAL THERMAL	PETER SHIH PS	12/11/08
STRUCTURAL	Raheel Haroon/RH	12/11/08
CHECKED	O. GANDOU/OG	12/10/08
DRAWN	TOMMY DUFFY/ TD	12/10/08

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INTERPRETATION OF WELD SYMBOLS PER ANSI / AWS 2.4

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SAFETY ANALYSIS REPORT
NUHOMS® 32PTH TYPE 1
TRANSPORTABLE CANISTER
PWR FUEL BASKET ASSEMBLY

DRAWING NO.	SCALE	SHEET	REVISION
10494-72-2003-SAR	NONE	1 OF 5	1A

DRAWING NO. 10494-72-2003-SAR SHEET 1 OF 5

PROPRIETARY AND SECURITY RELATED INFORMATION WITHHELD UNDER 10 CFR 2.390

DRAWING NO.	SHEET	REVISION
10494-72-2003-SAR	2 OF 5	1A

8 7 6 5 4 3 2 1

DRAWING NO. 10494-72-2003-SAR SHEET 3 OF 5

H G F E D C B A

PROPRIETARY AND
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DRAWING NO.	SHEET	REVISION
10494-72-2003-SAR	3 OF 5	1A

8 7 6 5 4 3 2 1

PROPRIETARY AND SECURITY RELATED INFORMATION WITHHELD UNDER 10 CFR 2.390

DRAWING NO.	SHEET	REVISION
10494-72-2003-SAR	4 OF 5	1A

PROPRIETARY AND
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DRAWING NO.	SHEET	REVISION
10494-72-2003-SAR	5 OF 5	1A

in Figure 2-2, which contain top and bottom end caps that confine any loose material and gross fuel particles to a known, sub-critical volume during normal, off-normal and accident conditions and to facilitate handling and retrievability. Reactor records, visual/videotape records, fuel sipping, ultrasonic examination, and radio chemistry are examples of techniques utilized by utilities to identify damaged fuel.

The end caps are sized to fit inside the fuel compartment (see drawing 10494-72-30). The bottom end cap is slid into the fuel compartment before loading the fuel, utilizing a special tool.

After fuel loading, a top end cap is placed into the fuel compartment. The end caps are not "attached" to the basket, but are a slip/friction fit into the basket compartment. The fuel assembly is thus enclosed/confined by the fuel compartment walls and the end caps. The DSC inner top cover prevents any significant movement of the top end cap. The damaged fuel assemblies can be retrieved simply by removing the top end cap and grappling the fuel assembly by normal means.

The NUHOMS®-32TH DSC basket is designed with three alternate poison materials: Borated Aluminum alloy, Boron Carbide/Aluminum Metal Matrix Composite (MMC) and Boral®.

The NUHOMS®-32PTH DSC basket is analyzed for seven alternate basket configurations, depending on the boron loadings and poison materials.

A summary of the alternate poison loadings considered for each poison material as a function of basket types is presented below:

NUHOMS®-32PTH DSC Basket Type	**Minimum B10 Areal Density, g/cm^2**	
	Natural or Enriched Boron Aluminum Alloy / Metal Matrix Composite (MMC) (Type I)	Boral® (Type II)
A	0.007	0.009
B	0.015	0.019
C	0.020	0.025
D	0.032	N/A
E	0.050	N/A

Table 2-2 shows a parametric equation that can be utilized to qualify spent fuel assemblies for the defined decay heat load zones. The decay heat load can be calculated based on a fuel assembly's burnup, cool time, and initial enrichment parameters. This table ensures that the fuel assembly decay heat load is within the appropriate zone. The development of this equation is provided in Appendix 4.16.2.

The maximum fuel cladding temperature limit of 400°C (752°F) is *set for* normal conditions of storage and all short term operations from *the* spent fuel pool to *the* ISFSI pad including vacuum drying and helium backfilling of the NUHOMS®-32PTH DSC per Interim Staff Guidance (ISG)

No. 11, Revision *3* [15]. In addition, the change in fuel cladding temperature *is restricted* to less than 65°C (117°F) and *is limited* to less than 10 *cycles* during DSC drying, backfilling and transfer operations *[15]*.

The maximum fuel cladding temperature limit *is set to* 570°C (1058°F) *for* accidents or off-normal thermal transients [15].

Calculations were performed to determine the fuel assembly type which was most limiting for each of the analyses including shielding, criticality, thermal and confinement. These evaluations are performed in Chapters 5 and 6. The fuel assembly classes considered are listed in Table 2-1. It was determined that the *Framatome ANP Advanced* MK BW 17x17 *(a WE 17x17 Class Assembly)* is the enveloping fuel design for the shielding, thermal and confinement source term calculation because of its total assembly weight and highest initial heavy metal loading. The bounding source term for shielding analysis is *described* in Table 2-3. Table 2-4 presents the thermal and radiological source terms for the *CCs*.

These values are consistent with the cumulative exposures and cooling times of the fuel assemblies. The gamma spectra for the bounding fuel assembly and *CCs* are presented in Chapter 5.

The shielding evaluation is performed assuming 32 fuel assemblies with the parameters *corresponding to a decay heat of* 1.5kW *per fuel assembly*. Any fuel assembly that is thermally qualified by Table 2-2 is also acceptable from a shielding perspective since the maximum decay heat load is 1.5 kW and only eight (8) are allowed in the 32PTH DSC. The shielding analysis assumes 32, 1.5 kW assemblies are in the 32PTH DSC. Minimum initial enrichments are defined for each of the zones to assure the shielding evaluation is bounding.

For criticality safety, the WE 17x17 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 the soluble boron concentration and fixed poison plate loading. *These results are shown in Table 2-6 and* the analyses results are presented in Chapter 6.

For calculating the maximum internal pressure in the NUHOMS®-32PTH DSC, it is assumed that 1% of the fuel rods are damaged for normal conditions, up to 10% of the fuel rods are damaged for off normal conditions, and 100% of the fuel rods will be damaged following a design basis accident event. A minimum of 100% of the fill gas and 30% of the fission gases within the ruptured fuel rods are assumed to be available for release into the DSC cavity, consistent with NUREG-1536 [17].

The maximum internal pressures used in the structural analysis for the NUHOMS®-32PTH DSC are 15 and 20 psig for normal and off-normal storage and transfer conditions respectively and 120 and 70 psig during transfer and storage accident conditions respectively.

The structural integrity of the fuel cladding due to the side drop is analyzed in Section 3.5.3. The end and corner drops are not considered credible during storage and transfer. The structural integrity of the fuel cladding due to these loads will be addressed by the users under their site license (10CFR50).

14. American Concrete Institute, "Building Code Requirements for Structural Concrete," ACI 318-95.

15. US NRC, Interim Staff Guidance -11, Rev *3*, "Cladding Considerations for the Transportation and Storage of Spent Fuel," dated *November 17, 2003*.

16. U.S. Government, "Domestic Licensing of Production and Utilization Facilities," Title 10 Code of Federal Regulations, Part 50, Office of the Federal Register, Washington, D.C.

17. NUREG-1536, "Standard Review Plan for Dry Cask Storage Systems," 1997.

CHAPTER 4
THERMAL EVALUATION

Table of Contents

4.8 Effective Fuel Properties

4.8.1 Discussion

The NUHOMS®-32PTH DSC finite element models simulate the effective thermal properties of the fuel with a homogenized material occupying the volume within the basket where the fuel assemblies are stored. Effective values for density, specific heat, and conductivity are determined for this homogenized material for use in the finite element models.

The 32PTH DSC is capable of handling a variety of spent PWR fuel assemblies. In order to determine conservative thermal properties of the homogenized fuel assembly, all of the PWR fuel assemblyies types to be stored in the 32PTH DSC are studied. WE and MK BW fuel assemblies are considered in one category with active fuel length of 144". The lowest effective thermal conductivity, density, and specific heat of these studied fuel assembly groupies are selected to apply in the finite element model. Use of these properties would conservatively predict bounding maximum temperatures for the components of the NUHOMS®-32PTH DSC. The effective fuel properties for CE 14x14 assembly are considered separately since CE 14x14 assembly has a shorter active fuel length.

The characteristics of the fuel assemblies to be stored in the 32PTH DSC are listed in Table 4-12.

4.8.2 Summary of Material Properties

1. UO2, Fuel Pellets

Conductivity and specific heat for fuel pellets are taken from [30] and listed below.

Temperature (°C)	k (cal/s-cm-°C) [30]	Temperature (°F)	k* (Btu/hr-in-°F)
25	0.025	77	0.503
100	0.021	212	0.423
200	0.018	392	0.362
300	0.015	572	0.302
500	0.0132	932	0.266
700	0.0123	1292	0.248
800	0.0124	1472	0.250

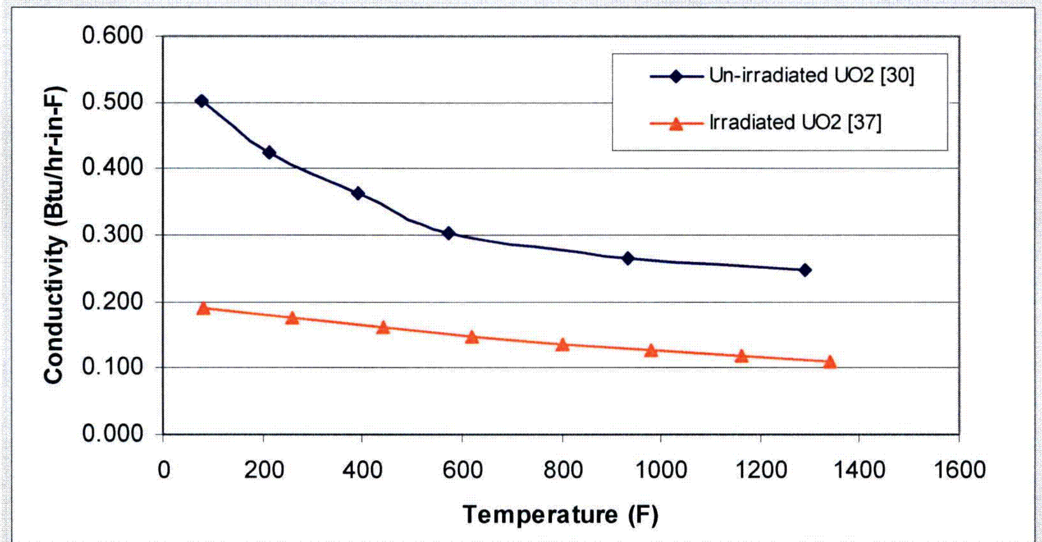
Temperature (°C)	C_p (cal/g-°C) [30]	Temperature (°F)	C_p (Btu/lbm-°F)
0	0.056	32	0.056
100	0.063	212	0.063
200	0.0675	392	0.068
400	0.0722	752	0.072
1200	0.079	2192	0.079

** See Section 4.8.6 for effect of irradiation on thermal conductivity of UO_2*

The density of fuel pellets (UO_2) is 10.96 g/cc = 0.396 lbm/in^3 [30].

4.8.6 Effect of Irradiation on UO_2 Thermal Conductivity

Based on Ronchi study [37], UO_2 thermal conductivity of irradiated UO_2 with ~62 GWd/t and irradiation temperature T_{irr} ≥1300K drops significantly (more that 50%) compared to un-irradiated UO_2. The thermal conductivity values of UO_2 in Section 4.8.2 [30] are compared to the values obtained from [37] study in the figure below.



Irradiated and Un-irradiated UO_2 Thermal Conductivity

The comparison shows that the [30] values in the fuel assembly temperature range of interest are higher by approximately a factor of two compared to values obtained from Ronchi study [37].

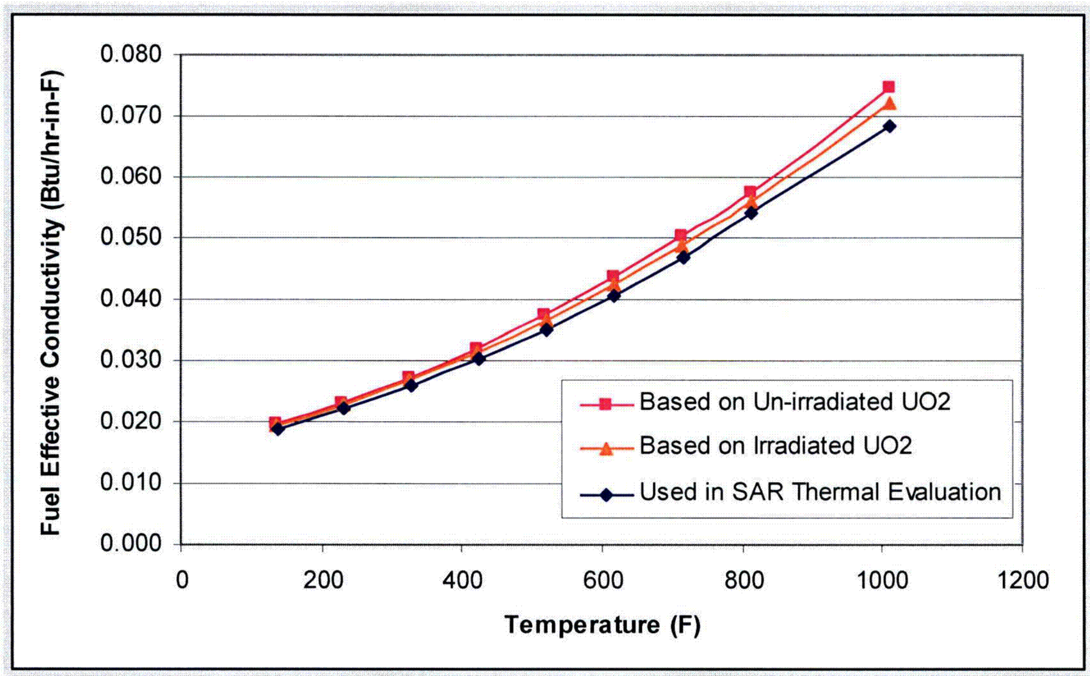
Using irradiated UO_2 conductivity decreases the effective conductivity of fuel assembly in transverse direction. Note that as discussed in Section 4.8.3.2, axial effective thermal conductivity of fuel assembly is calculated based on the fuel cladding material only and does not include the UO_2 fuel pellet thermal conductivity. Therefore, the axial effective conductivity of fuel assembly is not impacted.

A sensitivity analysis is performed to determine the impact of the irradiated UO_2 conductivity on the maximum fuel cladding temperatures. A sensitivity analysis includes two steps. In the first step, the transverse effective conductivity for fuel assemblies with irradiated and un-irradiated UO_2 conductivities are calculated based on the methodology described in Section 4.8.3.1.

In the second step, the calculated fuel assembly effective conductivities from the first step are used in the 32PTH DSC model from Section 4.3.1.3 to determine the maximum fuel cladding temperature. Normal transfer conditions for 32PTH DSC in OS187H transfer cask with heat load zoning configuration 1 at 115°F ambient is selected for this analysis.

The transverse effective conductivity for fuel assemblies calculated based on irradiated [37] and un-irradiated [30] UO_2 thermal conductivities are compared in the figure below. The transverse

effective conductivity for fuel assemblies used in the evaluation based on UO_2 properties used in the ANSYS model for fuel assembly effective conductivity calculation documented in the Section 4.2 (1) is also added to the figure below for reference.



Transverse Effective Conductivity for Fuel Assembly

As seen in the figure above, the fuel assembly effective thermal conductivity calculated with irradiated UO_2 conductivity is approximately 3% lower than the one calculated with un-irradiated UO_2 conductivity at the fuel cladding temperature of 700°F. The results of the sensitivity runs for the maximum fuel cladding temperature calculation using the DSC model from Section 4.3.1.3 are summarized in the table below.

Maximum Component Temperatures - Sensitivity Analysis
(32PTH DSC in OS187H, HLZC #1, 115 °F Ambient)

Component	*(1)*	*(2)*
Fuel Cladding	*716*	*717*
Fuel Compartment	*691*	*692*
Basket Al Plates	*691*	*691*
Basket Rails	*560*	*560*
DSC Shell	*475*	*475*

Notes:
(1) Effective conductivity for fuel assembly is based on un-irradiated UO_2 conductivity as shown in Section 4.2, subsection 1.

(2) Effective conductivity for fuel assembly is based on irradiated UO_2 conductivity values from Ronchi study [37].

The sensitivity analysis results show that values for both cases are comparable to those shown in Table 4-1, 2nd Part for Config. # 1. It also shows that the maximum fuel cladding temperature changes by approximately 1°F (0.14%) which is negligible. These results show that the fuel cladding temperatures are not sensitive to change in UO_2 thermal conductivity due to irradiation. Therefore, use of UO_2 fuel pellets conductivity from [30] is reasonable for irradiated UO_2.

4.8.6.1 UO_2 Thermal Conductivity used in ANSYS Fuel Assembly Model

The ANSYS model described in Section 4.8.3.1 erroneously but conservatively used UO_2 conductivity values which are lower than those shown in Section 4.8.2 (1). A comparison of these values is shown in the table below.

UO_2 Thermal Conductivity

Section 4.8 (1)		*Used in ANSYS model described in Section 4.8.3.1*	
Temperature (°F)	*k (Btu/hr-in-°F)*	*Temperature (°F)*	*k (Btu/hr-in-°F)*
77	*0.503*	*32*	*0.056*
212	*0.423*	*212*	*0.063*
392	*0.362*	*392*	*0.068*
572	*0.302*	*752*	*0.072*
932	*0.266*		
1292	*0.248*		
1472	*0.250*		

As seen from the table above, the UO_2 conductivity values used in the ANSYS model are at least 30% lower than the values obtained from Ronchi study [37]. Use of lower UO_2 thermal conductivity values in the ANSYS model of the fuel assembly results in conservatively lower values of effective thermal conductivity for fuel assembly. This in turn results in higher calculated fuel cladding and DSC component temperatures which are conservative. The transverse effective thermal conductivity for fuel assembly used in the thermal analysis is compared to the corresponding values from sensitivity analysis in the second figure in Section 4.8.6.

Since the effective thermal conductivity for fuel assembly used in thermal analyses of this SAR is lower than the effective thermal conductivity for fuel assembly with irradiated UO_2, the calculated maximum component temperatures are conservative and the difference in irradiated and un-irradiated UO_2 fuel pellet thermal conductivity values does not affect thermal analysis results reported in this SAR.

25. Transnuclear, Inc., "Thermal Testing of the NUHOMS® Horizontal Storage Module, Model HSM-H," Doc. No. E-21625, Rev. 1.
26. Chun, R., Witte, M., Schwartz, M., "Dynamic Impact Effects on Spent Fuel Assemblies," Lawrence Livermore National Laboratory, Report UCID-21246, 1987.
27. Young, W. C., "Roark's Formulas for Stress and Strain," 6th Edition, 1989.
28. Plannel, et al., "Extended Fuel Burnup Demonstration Program – Topical Report – Transport Considerations for Transnuclear Casks," DOE/ET 34014-11, TN-E4226, Transnuclear, Inc. 1983.
29. Brookmire, et al., "Storage of Burnable Poison Rod Assemblies and Thimble Plug Devices in Dry Storage Casks Surry ISFSI," NE-1162, Rev. 0, 1998.
30. Oak Ridge National Laboratory, RSIC Computer Code Collection, "SCALE, A Modular Code System for Performing Standardized Computer Analysis for Licensing Evaluation for Workstations and Personal Computers," NUREG/CR-0200, Rev. 6, ORNL/NUREG/CSD-2/V3/R6.
31. USNRC, SFPO, NUREG/CR-0497, "A Handbook of Materials Properties for Use in the Analysis of Light Water Reactor Fuel Rod Behavior," MATPRO - Version 11, EG&G Idaho, Inc., TREE-1280, 1979.
32. SANDIA Report, SAND90-2406, "A Method for Determining the Spent Fuel Contribution to Transport Cask Containment Requirements," 1992.
33. Diament, R.M.E., "Thermal and Acoustic Insulation," 1986.
34. Kreith, Frank, "The CRC Handbook of Thermal Engineering," 2000.
35. "ASHRAE Handbook, Fundamentals," – SI Edition, 1997.
36. I.E. Idelchik, "Handbook of Hydraulic Resistance," 3rd Edition, 1994.
37. *C. Ronchi, M. Sheindlin, D. Staicu, M. Kinoshita, Effect of burn-up on the thermal conductivity of uranium dioxide up to 1000 000 MWd t^{-1}, Journal of Nuclear Materials, 327 (2004), 58 – 76.*

CHAPTER 5
SHIELDING EVALUATION

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5. SHIELDING EVALUATION

The shielding evaluation presented for the NUHOMS® 32PTH System demonstrates adequacy of the shielding design for the payload described in Chapter 2. The geometry of the NUHOMS® System is described in Chapter 1. The heavy concrete walls and roof of the Horizontal Storage Module (HSM-H) provide the bulk of the shielding for the payload in the storage condition. During fuel loading and transfer operations, the combination of thick steel shield plugs at the ends of the 32PTH-DSC and heavy steel/lead/neutron shield material of the OS187H transfer cask provide shielding for personnel loading and transferring the 32PTH-DSC to the HSM-H. Figure 5-1 through Figure 5-4 and Table 5-1 provide the general configuration and material thicknesses of the important components of the NUHOMS® 32PTH System.

The NUHOMS® HD System is capable of storing CE 14x14 class, WE 15x15 class, CE 16x16 class and WE 17x17 class of PWR fuel assemblies.

For this shielding evaluation, source terms are calculated for the bounding Framatome ANP Advanced MK BW 17x17 (MK BW 17x17) fuel assembly, *a WE 17x17 class fuel assembly.* This fuel assembly is bounding because it contains the greatest mass of fuel.

The 32PTH DSC is also designed to store up to 32 intact standard PWR fuel assemblies with or without Control Components (CCs) such as burnable poison rod assemblies (BPRAs), Control Rod Assemblies (CRAs), Control Element Assemblies (CEAs), 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). The design basis CC for shielding evaluation is the BPRA.

Several burnup/enrichment combinations with minimum 5 year cooling times are addressed for the fuel to provide more flexibility in qualifying fuel for storage. These combinations form the basis for the NUHOMS® 32PTH System fuel specifications in Chapter 12. Bounding operating histories are assumed for the *BPRA* with a minimum cooling time of 4 days. The methodology, assumptions, and criteria used in this evaluation are summarized in the following subsections.

Section 5.4 provides a three dimensional (3-D) shielding analysis for the NUHOMS® 32PTH System using MCNP [2,6]

The shielding evaluation described in this chapter 5.0 is applicable to the 32PTH DSC in the OS187H TC and HSM-H. See Appendix A, Chapter A.5 for discussion of applicability of these analyses for the 32PTH Type 1 DSC in the OS187H Type 1 TC and HSM-H.

5.1 Discussion and Results

The maximum and average dose rates due to 32 design basis PWR fuel assemblies stored with 32 design basis *CCs* (BPRAs) in the NUHOMS® 32PTH System are summarized in Table 5-2 through Table 5-5. Table 5-2 provides the dose rates on the surface of the HSM-H while Table 5-3 through Table 5-5 provide the dose rates on and around the Transfer Cask (top, bottom and sides) during fuel loading, and transfer operations.

As previously stated, the NUHOMS® HD System is capable of storing PWR spent fuel, and *CCs*. Based on the source term calculations presented in Section 5.2, the design basis fuel source term is the Framatome MK BW 17x17 fuel assembly with 60 GWD/MTU burnup, a minimum initial enrichment of 4.0 weight % U-235 and a cooling time of 7 years. The design basis *CC* source term is a BPRA assembly irradiated to 30 GWD/MTU and a cooled for 4 days.

Fuel qualification tables are developed (based on a decay heat equation) that determine the eligibility of fuel assemblies to be stored in the 32PTH DSC. Since bounding parameters are utilized in all the 32 fuel assembly locations in the shielding evaluation, fuel qualification is limited only by the heat capacity of the DSC. This qualification covers fuel assemblies with a minimum enrichment of 0.2 wt. % U-235 and a minimum cooling time of 5 years. Fuel assemblies with enrichment between 0.2 wt. % U-235 and 1.5 wt. % U-235 are qualified by limiting their burnup. This ensures that the shielding analysis is also bounding for these fuel assemblies.

Reconstituted fuel assemblies where fuel pins that are replaced by lower-enriched pins or non-fuel pins are also authorized for storage. A discussion on the fuel qualification methodology is provided in Section 5.2.3.

A discussion of the method used to determine the design basis fuel and *CC* source terms is included in Section 5.2,. The model specification and shielding material densities are given in Section 5.3. The method used to determine the dose rates due to 32 design basis fuel assemblies with 32 design basis *CC* in the NUHOMS® 32PTH System is provided in Section 5.4.

Normal and off-normal conditions are modeled with the NUHOMS® 32PTH System intact, including the filled neutron shield in the transfer cask. The shielding calculations are performed using the MCNP Monte Carlo transport code [2]. Average and peak dose rates on the front, side, top and back of the HSM-H and the OS187H Transfer Cask System are calculated. Occupational doses during loading, transfer to the ISFSI, and maintenance and surveillance operations are provided in Chapter 10. *The areas of highest operational dose are the front of a loaded HSM-H at the air inlet vent, at the cask side or DSC top with a partially or completely drained DSC (cover welding, transfer operations) and at the cask/DSC annulus. Operating procedures, temporary shielding, and personnel training should minimize personnel exposure in these areas.*

For accident conditions (e.g., cask drop, fire), the transfer cask neutron shield water (shown in Figure 5-4 is assumed to be removed and a 1 inch void in the lead due to "lead slump' is also assumed at the top and/or bottom. Site dose and occupational dose analyses are addressed in Chapter 10 (including requirements for site specific 72.104 and 72.106 analyses).

5.2 Source Specification

Source terms are calculated with the SAS2H (ORIGEN-S) module of SCALE 4.4 [1]. The following sub-sections provide a discussion of the fuel assembly and *CC* material weights and composition, gamma and neutron source terms and energy spectrum. The SAS2H results are used to develop source terms suitable for use in the shielding calculations.

There are five principal sources of radiation associated with the NUHOMS® 32PTH System that are of concern for radiation protection. These are:

1. Primary gamma radiation from the spent fuel
2. Primary gamma radiation from activation products in the structural materials found in the spent fuel assembly and the *CC*
3. Primary neutron radiation from the spent fuel
4. Neutrons produced from sub-critical multiplication in the fuel
5. Capture gammas from (n,γ) reactions in the NUHOMS® 32PTH System materials

The first three sources of radiation are evaluated using SAS2H. The capture gamma radiation and sub-critical multiplication are handled as part of the shielding analysis which is performed with MCNP.

The neutron flux during reactor operation is peaked in the active fuel (in-core) region of the fuel assembly and drops off rapidly outside the in-core region. Much of the fuel assembly hardware is outside of the in-core region of the fuel assembly. To account for this reduction in neutron flux, each fuel assembly type is divided into four exposure zones. A neutron flux (fluence) correction is applied to each region to account for this reduction in neutron flux outside the in-core region. The correction factors are given in Table 5-6. The four exposure zones, or regions are [4]:

Bottom—location of fuel assembly bottom nozzle and fuel rod end plugs
In-core—location of active fuel
Plenum—location of fuel rod plenum spring and top plug
Top—location of top nozzle

The Framatome MK BW 17x17 assembly is the bounding fuel assembly design for shielding purposes because it has the highest initial heavy metal loading as compared to the 14x14, 15x15, *16x16* and other 17x17 fuel assemblies which are also authorized contents of the NUHOMS®-32PTH DSC and described in Chapter 2. The SAS2H/ORIGEN-S modules of the SCALE code with the 44 group ENDF/B-V library are used to generate the gamma and neutron source terms. For the bounding MK BW 17x17 fuel assembly, an initial enrichment of 4.0 wt% U-235 is assumed. The fuel assembly is irradiated with a constant specific power of 25 MW/assy to a total burnup of 60 GWD/MTU. A conservative three-cycle operating history is utilized with a 20 day down time between each cycle. The fuel assembly masses for each irradiation region are listed in Table 5-7

Data for the WE 17x17 assembly is from Reference [7]. Some values for the WE 15x15 were assumed to be the same as the WE 17x17. *The fuel assembly masses for each irradiation region of the CE 16x16 fuel assembly are also shown in Table 5-7.* The design-basis heavy metal weight is 0.476 MTU. These masses are irradiated in the appropriate fuel assembly region in the SAS2H/ORIGEN-S models. The mass of hardware for the MK BW 17x17 assembly is the greatest; however, the source term from the irradiated hardware for the WE 17x17 is bounding.

The maximum burnup of fuel assemblies with enrichments between 0.7 wt% U-235 and 1.5 wt % U-235 is limited to 32 GWD/MTU to ensure that their gamma and neutron source terms are bounded by those of the design basis fuel assembly. Similarly, the maximum burnup of fuel assemblies with enrichments between 0.3 wt.% U-235 and 0.7 wt.% U-235 is limited to 25 GWD/MTU. The maximum burnup of fuel assemblies with enrichments between 0.2 wt.% U-235 and 0.3 wt.% U-235 is limited to 20 GWD/MTU.

RECONSTITUTED FUEL ASSEMBLIES

A detailed discussion on the definition, qualification methodology, and evaluations for reconstituted fuel assemblies is provided in Section 5.2.3.

If reconstituted fuel assemblies (considered as intact fuel in the criticality analyses) with stainless steel rods undergo further irradiation, their gamma source term on a per DSC basis shall be bounded by the total design basis gamma source terms shown (on an assembly basis) in Table 5-10 for the design basis fuel assembly.

As explained above, reconstituted fuel assemblies may contain up to 10 irradiated stainless steel rods that replace damaged fuel rods. Because steel rods replace fuel rods, the decay heat of a reconstituted assembly is typically less than the decay heat of an equivalent standard assembly. Conversely, because steel contains Co-59 which activates to form Co-60, for low cooling times a reconstituted assembly typically generates higher dose rates than an equivalent standard assembly. As the half-life of Co-60 is 5.27 years, after 10 years the Co-60 activity has reduced by almost a factor of four and a reconstituted assembly no longer generates higher dose rates than an equivalent standard assembly. To bound this effect, the fuel qualification tables require that for fuel assembly with irradiated reconstituted steel rods with cooling times less than 10 years, additional one year of cooling time is required. For cooling times of 10 years or greater, no additional cooling time is required to bound the reconstituted fuel with steel rods.

TPA

The TPA materials and masses for each irradiation zone are listed in Table 5-8. These materials are irradiated in the appropriate zone for fourteen cycles of operation. The TPA is irradiated to an equivalent assembly life burnup of 210 GWd/MTU over 14 cycles. The model assumes that the TPA is irradiated in an assembly each with an initial enrichment of 3.50 weight % U-235.The fuel assembly, containing the TPA, is burned for three cycles with a burnup of 15 GWd/MTU per cycle. This is equivalent to an assembly life burnup of 45 GWd/MTU over the three cycles. The results for a cooling time of 20 years are increased by the ratio of 14/3 to achieve the equivalent 210 GWD/MTU source.

BPRA

The BPRA materials and masses for each irradiation zone are also listed in Table 5-8. These materials are irradiated in the appropriate zone for three cycles of operation. The model assumes that the BPRA is irradiated in an assembly each with an initial enrichment of 3.50 weight % U-235. The fuel assembly containing the BPRA is burned for three cycles with a burnup of 10 GWd/MTU per cycle. This is equivalent to an assembly life burnup of 30 GWd/MTU over the three cycles. The source term for the BPRA is taken at 4 days cooling time.

VSI

VSIs are very similar in design to burnable poison rod assemblies: the stainless steel baseplate and hold-down spring assembly designs are identical to those used on older Westinghouse BPRAs. Each VSI contains 24 solid Zircalloy-4 damper rods that are attached to the hold-down assembly using a crimp nut top connector. The damper rods are the same diameter and length as BPRA rodlets. The VSIs are assumed to be equivalent in source strength to BPRAs.

Neutron Sources

Neutron sources usually consist of a single pin containing the source material. They are typically irradiated for several cycles prior to final discharge. The neutron source term from these series is several orders of magnitude lower than that of the spent fuel. The gamma source term is bounded by that of a BPRA.

Other CCs

All other CCs listed in Section 5 are not evaluated explicitly. The cladding material for the CCs include, stainless steel, inconel, zirconium based alloys such as zircaloy, M5 or zirlo. The internal component materials include non fuel materials like inconel, B_4C, Ag-In-Cd, Al_2O_3 etc. All these CCs consist of one or several rodlets similar to BPRAs. However, the resulting source terms from these CCs are required to be bounded by that of the design basis BPRA as described above.

Elemental Compositions of Structural Materials

To account for the source terms due to the elemental composition of the fuel assembly and *CC* structural materials the following methodology is used:

1) The material composition for each irradiation region is determined for the assembly and *CC* type.

2) The elemental compositions for each of the structural materials present in each region is determined by multiplying the total weight of each material in a specific irradiation zone (Table 5-7) by the elemental compositions. The fuel assembly and NFAH elemental composition, including impurities, for each material are taken from Reference [7].

3) The results of each material are summed to determine the total elemental composition for each irradiation zone.

4) The elemental composition is multiplied by the appropriate flux factor given in Table 5-6.

5) Finally, the elemental composition is entered in the light element card of the SAS2H input. The elemental composition for the fuel assembly is shown in Table 5-9.

A comparison of the fuel assembly hardware (scaled elemental composition from Table 5-9) data for the WE 15x15, WE 17x17 and CE 16x16 indicates that the CE 16x16 fuel assembly has the lowest cobalt content. The Cobalt content in the CE 16x16 fuel assembly is 2.5 times lower than that of both the WE 15x15, WE 17x17 designs. Therefore, for the purpose of calculation of source terms, the fuel assembly with the highest material mass (e.g., steel, inconel) – the WE 17x17 is selected to be the design basis fuel assembly for modeling the hardware.

The MkBW 17x17 fuel assembly with the highest MTU loading is the design basis fuel assembly for irradiation purposes.

The SAS2H model of the design basis fuel assembly will be based on a geometry and material design of the MkBW 17x17 fuel assembly and the hardware design of the WE 17x17 fuel assembly.

The SAS2H calculation applies the total flux to the light elements; therefore, the total composition must be adjusted by the appropriate flux factor in the input. A SAS2H input is created for each irradiation zone of each fuel assembly and *CC* type. An example input file for the active fuel zone is shown in Section 5.5.2.

5.2.1 Gamma Sources

Source terms for the fuel bounding Framatome MK BW 17x17 fuel assembly and associated burnup/initial enrichment/cooling times and *CCs* are calculated with SAS2H module and the 44 group ENDF/B-V library. The SAS2H calculated contributions from actinides, fission products, and activation products, as applicable, are included for each irradiation region. The 7-year post irradiation cooling time results for the MK BW 17x17 fuel with 60 GWD/MTU burnup, and 4.0 wt % U-235 initial enrichment are shown in Table 5-10. The post irradiation cooling time results for the TPA, and BPRA are shown in Table 5-11, and Table 5-12, respectively.

Based on the results presented in Table 5-11 and Table 5-12 (maximum gamma source term) the design basis *CC* is the BPRA. The spectrum is dominated by Co-60 for all *CC*. These design basis fuel assembly sources with the BPRA source are used in the MCNP calculations to determine the bounding dose rates on and around the NUHOMS® 32PTH System, including the Transfer Cask.

5.2.2 Neutron Source

The total neutron source for the NUHOMS® 32PTH System is also calculated with SAS2H. The total neutron sources for the MK BW 17x17 assembly is summarized in Table 5-13. Again, the design basis source term is for 60 GWd/MTU burnup, 4.00 weight % U-235 initial enrichment and 7-year cooling time. The neutron source term consists primarily of spontaneous fission neutrons (largely from Cm-244) with (α,O-18) sources of lesser importance, both causing secondary fission neutrons. The overall spectrum is well represented by the Cm-244 fission spectrum.

5.2.3 Fuel Qualification

This section provides the basis for qualification of the design basis fuel to be loaded in the NUHOMS® HD System from a shielding perspective. The analyses are performed to demonstrate that the fuel assemblies with the parameters corresponding to the design basis fuel result in the highest calculated dose rate so that bounding shielding analysis can be performed by utilizing these design basis source terms. In order to determine the bounding spent fuel parameters (design basis fuel assembly), the candidate assembly parameters are ranked by their relative radiation source strengths. A simple 1-D shielding calculation based on the OS 187H transfer cask is performed and the radiation dose rates at the cask side surface are determined. The spent fuel parameters that yield the highest total dose rate (gamma + neutron) are considered the design basis for shielding calculations.

The SAS2H module of the SCALE 4.4 [1] computer code system is utilized to determine the dose rates for fuel qualification purposes. The mathematical function developed to determine the heat generation from the spent fuel as a function of enrichment, burnup and cooling time (the decay heat equation) is described in Appendix 4.16.2. The representative fuel qualification tables (FQT) are developed for each heat loading zone based on the decay heat equation and are also shown in Appendix 4.16.2 (Table 4.16.2-2 through Table 4.16.2-5).

The FQT data shown in Table 4.16.2-2 through Table 4.16.2-5 of Appendix 4.16.2 are based on limiting thermal sources. It does not necessarily mean that these fuel parameters also result in limiting radiation sources. The SAS2H dose comparison runs with the spent fuel parameters from these tables are utilized to generate additional restrictions (allowable combinations of burnup and enrichment) on the FQT such that these spent fuel parameters are bounded by the design basis assembly parameters. In other words, the FQT for radiation doses are developed such that the doses due to the fuel assemblies considered as acceptable for loading into the NUHOMS®-32PTH DSC are bounded by those determined for the design basis assembly.

The results of the surface dose rate comparison are shown in Table 5-24. In calculating the surface dose rates, it is assumed that the NUHOMS®-32PTH DSC is loaded with 32 fuel assemblies with the same characteristics. This dose rate comparison is only performed at the active fuel region. It is assumed that the contribution from fuel assembly hardware (end fittings and plenum regions) to the surface dose rate is low enough to assure that the fuel region dose rate comparison is adequate.

The SAS2H model is based on a homogenization of the basket and rail material. Further, the water in the DSC/TC annulus is not modeled as the DSC shell and the TC inner shell are merged into a single shell. The calculation of the dose rate with design basis source term with SAS2H essentially creates a "normalized" design basis dose rate which can then be compared to determine limiting parameters. A conservative specific power of 25 MW is utilized for all burnup values including at 20 and 25 GWD/MTU. The SAS2H input file utilized to perform the fuel qualification calculations is listed in Section 5.5.2.

In order to make a complete and thorough determination of the design basis spent fuel assembly, dose rates from the potential entries in the FQTs shown in Appendix 4.16.2 have to be

determined. However, basic shielding analysis principles can be utilized to reduce the number of SAS2H cask analysis calculations. A total of 42 burnup, enrichment, cooling time combinations have been utilized to perform the surface dose rate comparison studies. The selection of these candidate assembly parameters is based on the following principles:

- *The parameters that affect the dose rates in the increasing order of importance are burnup, enrichment and cooling time. This means that for a given heat load, the fuel assembly with the lowest enrichment and cooling time is expected to produce the largest dose rates.*

- *For the 5 years cooling time, the minimum and maximum burnup cases are analyzed.*

- *The maximum credited burnup (60 GWd/MTU) at the minimum possible cooling time is analyzed.*

- *Remaining cases that are analyzed are representative for each cooling time.*

- *Cases are also analyzed at a low enrichment (as low as 0.2 wt % U235) to determine any other restrictions for loading lower enriched fuel.*

These results indicate that the doses for spent fuel loadings with some of the parameters at a thermal power of 1500 watts per assembly have surface dose rates that exceed the design basis dose rate shown as Case 1. The total doses for those cases that exceed the design basis dose rates are shaded in grey in Table 5-24. These results imply that certain restrictions maybe placed when loading spent fuel at higher thermal power, specifically for loading in Zone 3.

A realistic, yet conservative approach to dose evaluation based on the results from Table 5-24 would indicate that the fuel loading in Zone 3 can be unrestricted. This is due to the fact that only 8 fuel assemblies with a decay heat load of 1500 watts per assembly are authorized be loaded in the NUHOMS®-32PTH canister while all the results (and restrictions) are based on a loading of 32 Zone 3 fuel assemblies. Therefore, any restriction to the Zone 3 fuel loading can only be applied by comparing the surface dose rate from a canister based on a multi-zone fuel loading to that of the design basis fuel. Calculations from other NUHOMS® DSCs indicate that the peripheral locations (16 fuel assemblies) contribute to approximately 80% of the total dose rate while the interior locations (16 fuel assemblies) contribute to the remaining 20%. Assuming that the Zone 2 and Zone 3 fuel assembly locations contribute equally to the dose rates due to the 16 outer locations, the contribution from Zone 3 and Zone 2 locations are approximately 40% and 60% respectively, to the total dose rates. Note that this approach is also conservative since it does not include Zone 1 fuel assemblies.

The average total surface dose rate for a conservative loading, based on 55% of the dose from the worst Zone 3 fuel (Case 2, 261.6 mrem/hour) and 45% of the dose from the worst Zone 2 fuel (Case 35, 215.1 mrem/hour), is 240.7 mrem/hour and is bounded by the design basis value of 241.7 mrem/hour. Moreover, the estimated dose rate from the worst Zone 3 fuel (Case 2, 261.6 mrem/hour) is higher than that from the design basis fuel (Case 1, 241.7 mrem/hour) by less than 10%. Therefore, it can be argued that based on a conservative yet realistic zone loading of fuel assemblies, the design basis fuel assembly analyzed for radiation dose rates is bounding.

All the dose rate comparisons for the fuel assemblies with lower enrichment (as low as 1.2 wt. % U-235, Table 5-24, Case 27 through Case 37) are made with Zone 1A and Zone 2 fuel only. These results conservatively encompass all of the Zone 1B results. These results indicate that the doses for spent fuel loadings with low enrichments and high burnups have surface dose rates that exceed the design basis dose rate shown as Case 1 and are shaded in grey in Table 5-24. Note that some of the burnup, enrichment and cooling time combinations evaluated in Table 5-24 have heat loads that exceed the maximum allowable value of 1500 watts. These cases are intended to be illustrative and demonstrate that the evaluation performed is reasonably encompassing. These results imply that certain restrictions maybe placed when loading spent fuel at very low enrichments combined with impractically high burnups.

For the cases with very low enrichment (less than 1.2 wt. % U-235, Table 5-24, Case 38 through Case 42), only bounding cases at the lowest enrichment, highest allowable burnup and lowest allowable cooling time are considered. Because of the low enrichment, small changes need to be made to the SAS2H models to ensure that the source calculations have converged. This is ensured by setting the NLIB/CYC to 2 for cases with enrichments of 0.7 wt. % U-235 and NLIB/CYC to 5 for cases with enrichments of 0.3 and 0.2 wt. % U-235.

The following is a summary of the significant results of this fuel qualification evaluation:

- *The spent fuel parameters utilized to determine the design basis source terms result in the design basis dose rates on and around the HD system, provided some restrictions are placed.*
- *The decay heat equation can be used to determine the decay heat for fuel assemblies with initial enrichments greater than or equal to 1.5 wt. % U-235. The minimum enrichments for loading in the various zones using the decay heat equation are: Zone 1- 1.5 wt. % U-235, Zone 2- 1.6 wt. % U-235 and Zone 3- 2.5 wt. % U-235.*
- *The maximum burnup for fuel assemblies with initial enrichment between 1.5 wt. % U-235 and 2.5 wt. % U-235 is 55 GWD/MTU.*
- *The maximum burnup for fuel assemblies with initial enrichment between 0.7 wt. % U-235 and 1.5 wt. % U-235 is 32 GWD/MTU.*
- *The maximum burnup for fuel assemblies with initial enrichment between 0.3 wt. % U-235 and 0.7 wt. % U-235 is 25 GWD/MTU.*
- *The maximum burnup for fuel assemblies with initial enrichment between 0.2 wt. % U-235 and 0.3 wt. % U-235 is 20 GWD/MTU.*

The evaluation to determine the cooling time requirements for fuel assemblies with reconstituted rods is documented herein. Reconstituted fuel assemblies are those where one or more fuel rods are replaced by rods that displace the same amount of water in the active fuel region. The material for these replacement rods can be solid zirconium alloy rods, low enriched UO_2 rods (with zirconium alloy cladding), solid or hollow stainless steel rods, etc. For fuel assemblies that do not undergo further irradiation in the reactor following the replacement of these rods, no

other cooling time restrictions are needed. Further, no restrictions on the material or the number of replacement rods are placed for such fuel assemblies.

For fuel assemblies that undergo further irradiation in the reactor following the replacement of these rods, restrictions on the cooling time of the fuel assemblies and in the material and number of the replacement rods is required. Replacement rods made up of zirconium alloy cladded low enriched UO_2 or solid zirconium alloys that undergo further irradiation in the reactor do not result in source terms that are greater than that of the original UO_2 rods. Therefore, no restrictions in their number or cooling time are necessary for their qualification – the thermal and source term qualification (from above) is sufficient.

Fuel assemblies with replacement stainless steel rods that undergo subsequent irradiation are restricted to a maximum of 10 replacement rods. Further, such fuel assemblies need to be cooled for an additional amount of time to ensure that the resulting source terms are bounded by those of the design basis fuel. There is no effect on the source terms/shielding due to the position of the reconstituted rods in the fuel rod array. Reconstituted fuel has a rather small effect on the dose rate such that for fuel assemblies with cooling times less than 10 years, an additional year of cooling time is required if reconstituted rods (with irradiated stainless steel) are present in fuel assemblies.

The remainder of this section documents the analysis performed to determine the cooling time restrictions for these fuel assemblies. This analysis is also performed using the SAS2H module and is similar to the analyses performed for the fuel qualification evaluation documented above.

The irradiation history utilized in these calculations is based on three cycles of equal duration. For the fuel assemblies with reconstituted rods, the first cycle assumes that no replacement of rods have taken place. The second and third cycles assume that 10 stainless steel rods have replaced the original UO_2 rods. For these cycles, the mass of Uranium in the fuel is reduced to 0.458 MTU from 0.476 MTU which corresponds to 10 rods. The mass of stainless steel is increased by 20.58 kg to account for the mass of 10 solid stainless steel rods.

The results of these evaluations are shown in Table 5-25. A total of 15 individual dose rate calculations are performed grouped under 6 case numbers. Case #s ending with "A" are based on fuel assembly without reconstituted rods. Case #s ending with "B" and "C" are based on fuel assembly with 10 reconstituted stainless steel rods that undergo additional 2 cycles of irradiation. These results indicate that for cooling times at or below 10 years, the dose rate from a reconstituted fuel assembly with reconstituted rods is greater than that of an un-reconstituted fuel assembly. This indicates that a cooling time of 10 years can represent a possible limit for fuel assemblies containing reconstituted rods. A comparison of the three SAS2H runs that constitute Case #1, indicate that the dose rate for the design basis fuel assembly increases from 241.7 mrem/hour to 253.0 mrem/hour when it is reconstituted. However, this dose rate drops to 226.5 mrem/hour when this fuel assembly is cooled by an additional year which represents a reduction of 10%. This indicates that an additional year of cooling time can also represent a conservative limit for reconstituted fuel assemblies.

The following is a summary of the significant results of the reconstituted fuel evaluation that provides the limits on the qualification of these fuel assemblies. The restrictions are applicable

only to fuel assemblies that contain no more than 10 replacement rods and that undergo further irradiation following reconstitution.

- *An additional cooling time of 1 year is needed to load reconstituted fuel assemblies in Zone 2 and Zone 3 that would have otherwise qualified based purely on thermal loading requirements.*
- *The cooling time restrictions are limited to reconstituted fuel assemblies with cooling times less than or equal to ten years.*

5.5.2 Sample Input Files

PROPRIETARY INFORMATION
on pages 5-17 to 5-58
Withheld pursuant to 10 CFR 2.390

Table 5-6
Flux Factor By Fuel Assembly Region

Fuel Assembly Region	Flux Factor
Bottom	0.20
In-Core	1.00
Plenum	0.20
Top	0.10

Table 5-7
Fuel Assembly Materials and Masses
(Page 1 of 2)

Region	Material	Mass (kg/assembly) WE 15x15	WE 17x17	MK BW 17x17
Top Fitting				
Upper Tie Plate	SS 304	6.8	6.8	7.0
Hold Down Springs	Inconel 718	1.1	1.37	1.1
Plenum				
Cladding & Guide Tubes	Zr-4	6.1	5.5	6.3
Plenum Spring	SS 302	1.5	1.9	4.7
Fuel Zone				
Cladding & Guide Tubes	Zr-4	99.2	102.9	109.9*
Grids	Zr-4			8.2
	Inconel-718	5.9	5.9	0.8
Grid Brazing Material				
	Nicrobraze 50	1.2	1.2	-
Miscellaneous				
	SS 304	4.6	4.6	0.1*
Bottom Fitting				
Bottom Tie Plate	SS 304	5.7	5.7	4.3
Total		132.1	135.6	142.4

* Clad is M5™ which is treated as Zr-4

Table 5-7
Fuel Assembly Materials and Masses
(Page 2 of 2)
CE 16x16 Fuel Assembly Hardware Materials and Masses

Item	Material	Average Weight (lb./assembly)	Average Weight (kg/assembly)
Active Fuel Zone			
Guide Tubes	Zircaloy-4	21	9.53
Spacer Grids	Zircaloy-4	23.4	10.62
Spacer Grid	Inconel 625	2.6	1.18
Cladding	Zircaloy-4	235.2	106.78
Fuel Rods	UO_2	1137	Total U = 455.5 kg
Plenum Zone			
Guide Tubes	Zircaloy-4	1.5	0.68
Spacer Grid	Zircaloy-4	1.8	0.82
Upper End Cap	Zircaloy-4	1.9	0.86
Cladding	Zircaloy-4	15.7	7.13
Plenum Springs	Stainless Steel 302	16.5	7.49
Spacer Discs	Al_2O_3	1.3	0.59
Top Fitting Zone			
Holddown Plate	Stainless Steel 304	24.6	11.17
Flow Plate	Stainless Steel 304		
Outer Posts	Stainless Steel 304		
Center Guide Post	Stainless Steel 304		
Guide Tubes	Zircaloy-4	0.3	0.14
Holddown springs	Inconel X-750	11.4	5.18
Bottom Fitting Zone			
Guide Tubes	Zircaloy-4	0.9	0.41
Locking Discs/Sleeve	Stainless Steel 304	0.2	0.09
Spacer Grid	Inconel 625	2.6	1.18
Spacer Discs	Al_2O_3	1.3	0.59
Cladding	Zircaloy-4	0.4	0.18
Bottom End Cap	Zircaloy-4	20.6	9.35
Lower End Fitting	Stainless Steel 304	13.1	5.95

Fuel Assembly Zone	Zircaloy	Steel	Inconel
	Component Mass (Kg)		
Bottom Fitting	9.94	6.04	1.18
Active Fuel	126.94	--	1.18
Plenum	9.49	7.49	--
Top Fitting	0.14	11.17	5.18

Table 5-9
Fuel Assembly Material Masses

(kg/assembly)

Scaling Factors	0.1	0.2	1	0.2	
	Top Fitting	**Plenum**	**Active Fuel**	**Bottom Fitting**	**Total**
15x15					
Chromium	0.1501	0.0555	2.2972	0.2166	2.7194
Manganese	0.0138	0.0060	0.1059	0.0228	0.1485
Iron	0.4879	0.2121	4.4512	0.7848	5.9360
Cobalt	0.0011	0.0003	0.0328	0.0009	0.0350
Nickel	0.1178	0.0268	4.3714	0.1017	4.6177
Zirconium	0.0000	1.1945	97.128	0.0000	98.322
Aluminum	0.0007	0.0000	0.0380	0.0000	0.0387
Silicon	0.0070	0.0030	0.0124	0.0000	0.0224
Titanium	0.0009	0.0000	0.0473	0.0000	0.0481
Niobium	0.0061	0.0000	0.3272	0.0000	0.3333
Molybdenum	0.0033	0.0000	0.1768	0.0000	0.1801
Tin	0.0000	0.0195	1.6608	0.0182	1.6986
17x17					
Chromium	0.1551	0.0698	2.3018	0.2166	2.7433
Manganese	0.0139	0.0076	0.1060	0.0228	0.1503
Iron	0.4927	0.2676	4.4595	0.7848	6.0047
Cobalt	0.0012	0.0003	0.0329	0.0009	0.0353
Nickel	0.1317	0.0339	4.3715	0.1017	4.6388
Zirconium	0.0000	1.0770	100.75	0.0000	101.83
Aluminum	0.0008	0.0000	0.0381	0.0000	0.0389
Silicon	0.0071	0.0038	0.0124	0.0182	0.0415
Titanium	0.0011	0.0000	0.0473	0.0000	0.0484
Niobium	0.0076	0.0000	0.3272	0.0000	0.3348
Molybdenum	0.0041	0.0000	0.1768	0.0000	0.1809
Tin	0.0000	0.0176	1.7200	0.0182	1.7558
16 x 16					
Chromium	*0.3104*	*0.2720*	*0.3827*	*0.2767*	*1.2419*
Manganese	*0.0234*	*0.0300*	*0.0049*	*0.0247*	*0.0829*
Iron	*0.8619*	*1.0496*	*0.4978*	*0.8783*	*3.2877*
Cobalt	*0.0033*	*0.0012*	*0.0068*	*0.0021*	*0.0134*
Nickel	*0.3686*	*0.1337*	*0.6159*	*0.2304*	*1.3486*
Zirconium	*0.0133*	*1.8581*	*124.2867*	*1.9559*	*128.114*
Aluminum	*0.0031*	*0.0000*	*0.0101*	*0.0015*	*0.0147*
Silicon	*0.0010*	*0.0150*	*0.0024*	*0.0005*	*0.0188*
Titanium	*0.0041*	*0.0000*	*0.0094*	*0.0019*	*0.0155*
Niobium	*0.0287*	*0.0000*	*0.0655*	*0.0131*	*0.1073*
Molybdenum	*0.0155*	*0.0000*	*0.0354*	*0.0071*	*0.0579*
Tin	*0.0181*	*0.0304*	*2.0310*	*0.0513*	*2.1307*

Table 5-24
Dose Rate Comparison for Fuel Qualification
(Page 1 of 4)

Case ID	Description	Surface Dose Rate (mrem/hour)		
		Neutron	Gamma	Total
1	Enrichment 4.00 wt. % U-235 Burnup 60.0 GWD/MTU Cooling Time 7 Years Design Basis	48.2	193.5	241.7
2	Enrichment 2.50 wt. % U-235 Burnup 46.0 GWD/MTU Cooling Time 5 Years Thermal Power ≈ 1500 Watts	36.6	225.0	261.6
3	Enrichment 2.50 wt. % U-235 Burnup 44.0 GWD/MTU Cooling Time 5 Years Thermal Power 1404 Watts	31.2	208.4	239.6
4	Enrichment 2.50 wt. % U-235 Burnup 40.0 GWD/MTU Cooling Time 5 Years Thermal Power 1236 Watts	21.9	177.7	199.6
5	Enrichment 2.50 wt. % U-235 Burnup 50.0 GWD/MTU Cooling Time 7 Years Thermal Power ≈ 1278 Watts	45.3	186.4	231.7
6	Enrichment 2.75 wt. % U-235 Burnup 45.0 GWD/MTU Cooling Time 7 Years Thermal Power ≈ 1088 Watts	27.7	140.5	168.2
7	Enrichment 3.00 wt. % U-235 Burnup 47.0 GWD/MTU Cooling Time 5 Years Thermal Power ~ 1500 Watts	31.4	209.5	240.9
8	Enrichment 3.00 wt. % U-235 Burnup 52.4 GWD/MTU Cooling Time 6 Years Thermal Power ≈ 1500 Watts	45.0	210.5	255.5
9	Enrichment 3.00 wt. % U-235 Burnup 51.2 GWD/MTU Cooling Time 6 Years Thermal Power 1432 Watts	41.2	200.4	241.6
10	Enrichment 3.00 wt. % U-235 Burnup 52.4 GWD/MTU Cooling Time 10 Years Thermal Power 1069 Watts	38.4	130.0	168.4
11	Enrichment 3.00 wt. % U-235 Burnup 55.0 GWD/MTU Cooling Time 8 Years Thermal Power 1299 Watts	49.3	178.1	227.4

Table 5-24
Dose Rate Comparison for Fuel Qualification
(Page 2 of 4)

Case ID	Description	Surface Dose Rate (mrem/hour)		
		Neutron	Gamma	Total
12	Enrichment 3.00 wt. % U-235 Burnup 59.5 GWD/MTU Cooling Time 15 Years Thermal Power 1069 Watts	48.9	122.5	171.4
13	Enrichment 3.50 wt. % U-235 Burnup 53.5 GWD/MTU Cooling Time 6 Years Thermal Power ~ 1500 Watts	39.6	196.3	235.9
14	Enrichment 3.50 wt. % U-235 Burnup 58.0 GWD/MTU Cooling Time 7 Years Thermal Power ~ 1500 Watts	51.1	200.5	251.6
15	Enrichment 3.50 wt. % U-235 Burnup 50.0 GWD/MTU Cooling Time 5 Years Thermal Power 1566 Watts	31.9	211.7	243.6
16	Enrichment 3.50 wt. % U-235 Burnup 56.6 GWD/MTU Cooling Time 7 Years Thermal Power 1433 Watts	46.8	189.8	236.6
17	Enrichment 3.75 wt. % U-235 Burnup 48.4 GWD/MTU Cooling Time 5 Years Thermal Power ~ 1500 Watts	25.3	190.9	216.2
18	Enrichment 3.75 wt. % U-235 Burnup 54.0 GWD/MTU Cooling Time 6 Years Thermal Power 1500 Watts	37.1	189.7	226.8
19	Enrichment 3.75 wt. % U-235 Burnup 58.5 GWD/MTU Cooling Time 7 Years Thermal Power ~ 1500 Watts	48.1	193.1	241.2
20	Enrichment 4.00 wt. % U-235 Burnup 48.8 GWD/MTU Cooling Time 5 Years Thermal Power ~ 1500 Watts	23.6	185.3	208.9
21	Enrichment 4.00 wt. % U-235 Burnup 54.5 GWD/MTU Cooling Time 6 Years Thermal Power ~ 1500 Watts	34.9	183.8	218.7

Table 5-24
Dose Rate Comparison for Fuel Qualification
(Page 3 of 4)

Case ID	Description	Surface Dose Rate (mrem/hour)		
		Neutron	Gamma	Total
22	Enrichment 4.25 wt. % U-235 Burnup 55.0 GWD/MTU Cooling Time 6 Years Thermal Power: 1475 Watts	32.9	178.4	211.3
23	Enrichment 4.50 wt. % U-235 Burnup 55.0 GWD/MTU Cooling Time 5 Years Thermal Power 1689 Watts	31.3	211.4	242.7
24	Enrichment 4.50 wt. % U-235 Burnup 60.0 GWD/MTU Cooling Time 6 Years Thermal Power: 1646 Watts	42.1	204.0	246.1
25	Enrichment 4.75 wt. % U-235 Burnup 55.0 GWD/MTU Cooling Time 5 Years Thermal Power 1668 Watts	28.6	203.3	231.9
26	Enrichment 4.75 wt. % U-235 Burnup 60.0 GWD/MTU Cooling Time 6 Years Thermal Power 1625 Watts	38.7	195.0.9	233.7
27	Enrichment 1.20 wt. % U-235 Burnup 33.4 GWD/MTU Cooling Time 5 Years Thermal Power ~ 1082 Watts	24.9	187.0	211.9
28	Enrichment 1.50 wt. % U-235 Burnup 34.1 GWD/MTU Cooling Time 5 Years Thermal Power 1083 Watts	22.0	177.2	199.2
29	Enrichment 2.00 wt. % U-235 Burnup 35.1 GWD/MTU Cooling Time 5 Years Thermal Power 1080 Watts	17.8	162.5	180.3
31	Enrichment 1.50 wt. % U-235 Burnup 42.2 GWD/MTU Cooling Time 7 Years Thermal Power ~ 1081 Watts	41.5	177.5	219.0
32	Enrichment 2.00 wt. % U-235 Burnup 43.3 GWD/MTU Cooling Time 7 Years Thermal Power ~ 1082 Watts	35.0	160.4	195.4

Table 5-24
Dose Rate Comparison for Fuel Qualification
(Page 4 of 4)

Case ID	Description	Surface Dose Rate (mrem/hour)		
		Neutron	Gamma	Total
33	Enrichment 1.50 wt. % U-235 Burnup 49.8 GWD/MTU Cooling Time 10 Years Thermal Power 1080 Watts	61.5	181.8	243.3
34	Enrichment 1.60 wt. % U-235 Burnup 50.0 GWD/MTU Cooling Time 10 Years Thermal Power 1080 Watts	59.6	177.6	237.2
35	Enrichment 2.00 wt. % U-235 Burnup 50.8 GWD/MTU Cooling Time 10 Years Thermal Power 1081 Watts	52.7	162.4	215.1
36	Enrichment 1.50 wt. % U-235 Burnup 48.1 GWD/MTU Cooling Time 10 Years Thermal Power 1029 Watts	55.4	168.7	224.1
37	Enrichment 1.60 wt. % U-235 Burnup 57.3 GWD/MTU Cooling Time 15 Years Thermal Power ~ 1080 Watts	72.4	169.2	241.6
38	Enrichment 0.70 wt. % U-235 Burnup 26.0 GWD/MTU Cooling Time 5 Years Thermal Power < 1000 Watts	15.9	156	171.9
39	Enrichment 0.70 wt. % U-235 Burnup 32.0 GWD/MTU Cooling Time 5 Years Thermal Power ~ 1100 Watts	29.3	202	231.3
40	Enrichment 0.30 wt. % U-235 Burnup 15.0 GWD/MTU Cooling Time 5 Years Thermal Power < 1000 Watts	3.3	92.0	95.4
41	Enrichment 0.30 wt. % U-235 Burnup 25.0 GWD/MTU Cooling Time 5 Years Thermal Power ~ 1000 Watts	26.9	194	220.9
42	Enrichment 0.20 wt. % U-235 Burnup 20.0 GWD/MTU Cooling Time 5 Years Thermal Power < 800 Watts	10.5	133.1	143.6

Table 5-25
Fuel Qualification with Reconstituted Fuel Assemblies

SAS2H Case #	*Burn-up (GWD/MTU)*	*Enrichment (wt. % U-235)*	*Cooling Time (years)*	*Total Dose Rate (mrem/hour)*	*Decay Heat (watts/FA)*
1A	*60.0*	*4.0*	*7.0*	*241.7*	*1515*
1B	*60.0*	*4.0*	*7.0*	*253.0*	*≤1515*
1C	*60.0*	*4.0*	*8.0*	*226.5*	*≤1392*
2A	*32.0*	*0.7*	*5.0*	*226.8*	*1074*
2B	*32.0*	*0.7*	*5.0*	*241.4*	*≤1074*
3A	*57.25*	*1.6*	*15.0*	*241.6*	*1080*
3B	*57.25*	*1.6*	*15.0*	*239.8*	*≤1080*
4A	*57.25*	*2.0*	*14.5*	*218.2*	*1076*
4B	*57.25*	*2.0*	*14.5*	*217.5*	*≤1076*
5A	*50.8*	*2.0*	*10.0*	*215.1*	*1081*
5B	*50.8*	*2.0*	*10.0*	*213.0*	*≤1081*
6A	*60.0*	*2.5*	*16.5*	*189.1*	*1070*
6B	*60.0*	*2.5*	*16.5*	*186.7*	*≤1070*

Note: Case #s ending with "A" are based on fuel assembly without reconstituted rods. Case #s ending with "B" and "C" are based on fuel assembly with 10 reconstituted stainless steel rods that undergo an additional 2 cycles of irradiation.

exterior surface of the cask with clean water to minimize surface adhesion of contamination.

4. Place the cask in the location of the fuel pool designated as the cask loading area.

5. Disengage the lifting yoke from the transfer cask lifting trunnions and move the yoke clear of the cask. Spray the lifting yoke with clean water if it is raised out of the fuel pool.

6. Load pre-selected spent fuel assemblies into the DSC basket compartments. The licensee shall develop procedures to verify that the boron content of the water conforms to the Technical Specifications, and that fuel identifications are verified and documented. Damaged fuel must be loaded only in designated compartments fitted with a damaged fuel bottom end cap.

7. After all the fuel assemblies have been placed into the DSC and their identities verified, install damaged fuel top end caps into designated compartments containing damaged fuel.

8. Lower the inner top cover/shield plug[1] in the DSC, aligning it with the guide on the DSC wall, and engaging the drain tube, until it seats on its support ring.

9. Visually verify that the inner top cover/shield plug is properly seated in the DSC. Reseat if necessary.

10. Position the lifting yoke and verify that it is properly engaged with the transfer cask trunnions.

11. Lift the transfer cask to the pool surface and spray the exposed portion of the cask with clean water.

12. Drain any water from above the inner top cover/shield plug back to the spent fuel pool. Up to 1300 gallons of water may be removed from the DSC prior to lifting the transfer cask clear of the pool surface. Up to 15 psig of helium may *only* be used to assist the removal of water. The DSC shall be backfilled *only* with helium after drainage of bulk water.

13. Lift the cask from the fuel pool, continuing to spray the cask with clean water. *Provisions shall be made to assure that air will not enter the DSC cavity. One way to achieve this is by replenishing the helium in the DSC cavity during cask movement from the fuel pool to the decon area in case of malfunction of equipment used for cask movement.*

14. Move the cask with loaded DSC to the area designated for DSC draining and closure operations. The set-down area should be level, or if slightly sloped, the transfer cask and DSC should be placed with the slope down toward the DSC drain/siphon tube.

[1] Including option 2 or option 3 inner top cover as described in Chapter 1 drawings.

8.1.1.3 DSC Closing, Drying, and Backfilling

1. Fill the transfer cask liquid neutron shield if it was drained for weight reduction during preceding operations.

2. Decontaminate the transfer cask exterior.

3. Disengage the rigging from the inner top cover/shield plug, and remove the eyebolts. Disengage the lifting yoke from the trunnions.

4. Disconnect the annulus overpressure tank if one was used, decontaminate the exposed surfaces of the DSC shell perimeter, remove any remaining water from the top of the annulus seal, and remove the seal.

5. Open the cask cavity drain port and allow water from the annulus to drain out until the water level is approximately twelve inches below the top of the DSC shell. Take swipes around the outer surface of the DSC shell to verify conformance with Technical Specification limits.

6. Cover the transfer cask / DSC annulus to prevent debris and weld splatter from entering the annulus.

7. If water was not drained from the DSC earlier, connect a pump to the DSC drain port and remove up to 1300 gallons of water. *Only u*se helium to assist the removal of water. This lowers the water sufficiently to allow welding of the inner top cover/shield plug. Up to 15 psig of helium gas may be applied at the vent port to assist the water pump down.

   CAUTION: Radiation dose rates are expected to be high at the vent and siphon port locations. Use proper ALARA practices (e.g., use of temporary shielding, appropriate positioning of personnel, etc.) to minimize personnel exposure.

7a. Monitor TC/DSC annulus water level to be approximately twelve inches below the top of the DSC shell and replenish as necessary until drained.

8. Install the automated welding machine onto the inner top cover/shield plug.

9. Hydrogen monitoring is required prior to commencing and continuously during the welding of the inner top cover / shield plug *per Technical Specification 5.6.* Insert a hydrogen monitor intake line through the vent port such that it terminates just below the inner top cover/shield plug.

10. Verify that the hydrogen concentration does not exceed 2.4% [1]. If this limit is exceeded, stop all welding operations and purge the DSC cavity with helium via the vent port to reduce hydrogen concentration safely below the 2.4% limit *before resuming welding operations.*

CAUTION: Radiation dose rates are expected to be high at the vent and siphon port locations. Use proper ALARA practices (e.g., use of temporary shielding, appropriate positioning of personnel, etc.) to minimize personnel exposure.

12. Obtain a sample of the DSC atmosphere. Confirm acceptable hydrogen concentration and check for presence of fission gas indicative of degraded fuel cladding.

13. If degraded fuel is suspected, additional measures appropriate for the specific conditions are to be planned, reviewed, and implemented to minimize exposures to workers and radiological releases to the environment.

14. Verify that the boron content of the fill water conforms to the Technical Specifications. Fill the DSC with water from the fuel pool or equivalent source through the drain port with the vent port open. The vented cavity gas may include steam, water, and radioactive material, and should be routed accordingly. Monitor the vent pressure and regulate the water fill rate to ensure that the pressure does not exceed 15 psig.

15. Provide for continuous hydrogen monitoring of the DSC cavity atmosphere during all subsequent cutting operations, *per Technical Specification 5.6*, to ensure that hydrogen concentration does not exceed 2.4%. Purge with helium as necessary to maintain the hydrogen concentration below this limit *before resuming cutting operations*.

16. Provide suitable protection for the transfer cask during cutting operations.

17. Using a suitable method, such as mechanical cutting, remove the weld of the outer top cover plate to the DSC shell.

18. Remove the outer top cover plate.

19. Remove the weld of the inner top cover/shield plug to the shell in the same manner as the outer cover plate. Do not remove the inner top cover/shield plug at this time unless the removal is being done remotely in a dry transfer system.

20. Remove any remaining excess material on the inside shell surface by grinding.

21. Clean the transfer cask surface of dirt and any debris which may be on the transfer cask surface as a result of the weld removal operation.

22. Engage the yoke onto the trunnions, install eyebolts or other lifting attachment(s) into the inner top cover/shield plug, and connect the rigging cables to the eyebolts/lifting attachment(s).

23. Verify that the lifting hooks of the yoke are properly positioned on the trunnions.

9.1.7 Neutron Absorber Tests

CAUTION
Sections 9.1.7.1 through 9.1.7.3 below are incorporated by reference into the NUHOMS® CoC 1030 Technical Specifications (paragraph 4.3.1) and shall not be deleted or altered in any way without a CoC amendment approval from the NRC. The text of these sections is shown in bold type to distinguish it from other sections.

The neutron absorber used for criticality control in the DSC basket may consist any of the following types of material:

(a) Boron-aluminum alloy (borated aluminum)

(b) Boron carbide-aluminum metal matrix composite

(c) Boral®

The 32PTH DSC safety analyses do not rely upon the tensile strength of these materials. The radiation and temperature environment in the cask is not sufficiently severe to damage these metallic/ceramic materials. To assure performance of the neutron absorber's design function only the presence of B10 and the uniformity of its distribution need to be verified, with testing requirements specific to each material. The boron content of these materials is given in Table 9-1.

9.1.7.1 Boron Aluminum Alloy (Borated Aluminum)

See the Caution in Section 9.1.7 before deletion or modification to this section.

The material is produced by direct chill (DC) or permanent mold casting with boron precipitating as a uniform fine dispersion of discrete AlB_2 or TiB_2 particles in the matrix of aluminum or aluminum alloy. For extruded products, the TiB_2 form of the alloy shall be used. For rolled products, either the AlB_2, the TiB_2, or a hybrid may be used.

Boron is added to the aluminum in the quantity necessary to provide the specified minimum B10 areal density in the final product, with sufficient margin to minimize rejection, typically 10 % excess. The amount required to achieve the specified minimum B10 areal density will depend on whether boron with the natural isotopic distribution of the isotopes B10 and B11, or boron enriched in B10 is used. In no case shall the boron content in the aluminum or aluminum alloy exceed 5% by weight.

The criticality calculations take credit for 90% of the minimum specified B10 areal density of borated aluminum. The basis for this credit is the B10 areal density acceptance testing, which shall be as specified in Section 9.5.2. The specified acceptance testing assures that at any location in the material, the minimum specified areal density of B10 will be found with 95% probability and 95% confidence.

Visual inspections shall follow the recommendations in Aluminum Standards and Data, Chapter 4 "Quality Control, Visual Inspection of Aluminum Mill Products and Castings"[5]. ***Each finished borated aluminum piece shall be visually inspected on both faces.***

Local or cosmetic conditions such as scratches, nicks, die lines, inclusions, abrasion, isolated pores, or discoloration are acceptable. Widespread blisters, rough surface, or cracking shall be evaluated for acceptance in accordance with the Certificate Holder's QA procedures.

9.1.7.2 Boron Carbide / Aluminum Metal Matrix Composites (MMC)

See the Caution in Section 9.1.7 before deletion or modification to this section.

The material is a composite of fine boron carbide particles in an aluminum or aluminum alloy matrix. The material shall be produced by either direct chill casting, permanent mold casting, powder metallurgy, or thermal spray techniques. It is a low-porosity product, with a metallurgically bonded matrix. The boron carbide content shall not exceed 40% by volume. ***The boron carbide content for MMCs with an integral aluminum cladding shall not exceed 50% by volume.***

The final MMC product shall have density greater than 98% of theoretical density demonstrated by qualification testing. For MMC with an integral cladding, the final density shall be greater than 97% of theoretical density demonstrated by qualification testing.

Prior to use in the 32PTH DSC, MMCs shall pass the qualification testing specified in Section 9.5.3, and shall subsequently be subject to the process controls specified in Section 9.5.4.

The criticality calculations take credit for 90% of the minimum specified B10 areal density of MMCs. The basis for this credit is the B10 areal density acceptance testing, which is specified in Section 9.5.2. The specified acceptance testing assures that at any location in the final product, the minimum specified areal density of B10 will be found with 95% probability and 95% confidence.

Visual inspections shall follow the recommendations in Aluminum Standards and Data, Chapter 4 "Quality Control, Visual Inspection of Aluminum Mill Products and Castings" [5]. ***Each finished MMC piece shall be visually inspected on both faces.*** **Local or cosmetic conditions such as scratches, nicks, die lines, inclusions, abrasion, isolated pores, or discoloration are acceptable. Widespread blisters, rough surfaces, or cracking shall be evaluated for acceptance in accordance with the Certificate Holder's QA procedures.** ***MMCs with an integral aluminum cladding shall also be inspected on the edges, and visual inspection shall include verification that the matrix is not exposed through the faces of the aluminum cladding and that solid aluminum is not present at the edges.***

References to metal matrix composites throughout this chapter are not intended to refer to Boral®, which is described in the following section.

9.1.7.3 Boral®

See the Caution in Section 9.1.7 before deletion or modification to this section.

This material consists of a core of aluminum and boron carbide powders between two outer layers of aluminum, mechanically bonded by hot-rolling an "ingot" consisting of an aluminum box filled with blended boron carbide and aluminum powders. The core, which is exposed at the edges of the sheet, is slightly porous. The average size of the boron carbide

particles in the finished product is approximately 50 microns after rolling. The nominal boron carbide content shall be limited to 65% (+ 2% tolerance limit) of the core by weight.

The criticality calculations take credit for 75% of the minimum specified B10 areal density of Boral®. B10 areal density will be verified by chemical analysis and by certification of the B10 isotopic fraction for the boron carbide powder, or by neutron transmission testing. Areal density testing is performed on an approximately 1 cm² area of a coupon taken near one of the corners of the sheet produced from each ingot. If the measured areal density is below that specified, all the material produced from that ingot will be either rejected, or accepted only on the basis of alternate verification of B10 areal density for each of the final pieces produced from that ingot.

Each finished Boral piece shall be visually inspected on faces and edges. Any piece with cracks through the cladding, exposed core on the face of the sheet, or solid aluminum at the edge of the sheet will be treated as non-conforming.

9.2 Maintenance Program

The NUHOMS® HD System is designed to be totally passive with minimal maintenance requirements. The 32PTH DSC does not require any maintenance once it is loaded into the HSM-H. The HSM-H does not require any maintenance other than that indicated in off-normal operations, Chapter 11, such as clearing of blocked air inlets. Periodic inspection is therefore limited to the Transfer Cask.

9.2.1 Inspection

The following inspections of the transfer cask should be performed prior to each fuel loading or unloading campaign:

A. Visual inspection of the transfer cask trunnions for damaged bearing surfaces

B. Visual or functional inspection of all taps, threaded inserts, and bolts

C. Functional inspection of all quick-connect fittings

D. Visual inspection of the interior surface of the cask for any indications of excessive wear.

E. Visual inspection of the neutron shield jacket for indications of damage

F. Visual inspection of all Transfer Cask o-rings for indications of damage

Within the year prior to any loading or unloading campaign, the top trunnion bearing surfaces and accessible welds shall be examined by dye penetrant. No linear indications shall be acceptable other than surface scratches and wear.

9.2.2 Tests

The Transfer Cask lid and ram access cover o-rings, vent and drain quick connect fittings, and neutron shield fittings shall be leak tested within the year before the start of any fuel loading or unloading campaign. If bubble leak testing is used, no leak indication is allowed. If pressure drop or helium leak testing is used, the maximum allowable leak for each of the components listed is 10^{-3} ref cm^3/s. If any of the listed components is replaced, that component shall be leak tested before use in fuel loading or unloading operations.

No periodic testing of the 32PTH DSC, HSM-H or routine support equipment is required.

Temperature and radiation monitoring is provided in accordance with the Technical Specifications. Periodic calibration of the monitoring equipment shall be as required by the licensee's quality program.

9.2.3 Repair, Replacement, and Maintenance

Any parts which fail inspections listed in 9.1.2 shall be repaired or replaced. Such parts may be also be accepted as-is if determined appropriate by engineering and licensing review.

9.3 Marking

The HSM-H and 32PTH DSC are marked with the model number, unique identification number, and empty weight in accordance with 10 CFR 72.236(k). The 32PTH DCS nameplate is shown in drawing 10494-72-7.

9.4 Pre-Operational Testing and Training Exercise

A dry run training exercise of the loading, closure, handling, unloading, and transfer of the NUHOMS® HD System shall be performed by each licensee prior to their first use of the system to load spent fuel assemblies. The dry run shall be conducted with simulated fuel to match the weight of the actual fuel. The dry run need not be performed in the sequence of operations in Chapter 8. The dry run shall include:

(a) Loading of mock-up fuel

(b) DSC draining, vacuum drying, welding, and backfilling

(c) Loading of the Transfer Cask onto the Transfer Trailer, and transfer to the ISFSI

(d) DSC transfer to the HSM-H

(e) DSC retrieval from the HSM-H

(f) Re-flooding of a sealed 32PTH DSC

(g) Removal of the covers from a sealed 32PTH DSC

The dry run will simulate, as nearly as possible, the detailed written procedures developed by the licensee for NUHOMS® HD System operations. Guidelines for the dry run follow.

A. An actual or a mock-up 32PTH DSC loaded with mock-up fuel is typically utilized. The 32PTH DSC is loaded into the transfer cask; the transfer cask/DSC annulus seal is installed.

B. Functional testing is performed with the transfer cask and lifting equipment. These tests are to ensure that the transfer cask can be safely lifted from the plant's cask receiving area to the cask washdown area. The cask is partially lowered into the spent fuel pool and positioned in the cask loading area to verify clearances and travel path. The inner top cover is installed to verify handling and alignment operations.

C. The transfer cask is placed on the transfer trailer, which is moved to the ISFSI aligned with an HSM-H. Compatibility of the transfer trailer with the transfer cask, verification of the transfer route to the ISFSI, and maneuverability within the confines of the ISFSI are verified.

thermal conductivity acceptance criterion for the neutron absorber will be based on the nominal thickness specified. The minimum thermal conductivity shall be such that the total thermal conductance (sum of conductivity * thickness) of the neutron absorber and the aluminum 1100 plate shall equal the conductance assumed in the analysis, as shown in Table 9-3, where the acceptance criterion is highlighted.

The aluminum 1100 plate does not need to be tested for thermal conductivity; the material may be credited with the values published in the ASME Code Section II part D. The neutron absorber material need not be tested for thermal conductivity if the nominal thickness of the aluminum 1100 plate is 0.425 inch or greater. This case is examined explicitly in chapter 4, where no credit is taken for the thermal conductivity of Boral®.

9.5.2 Specification for Acceptance Testing of Neutron Absorbers by Neutron Transmission

CAUTION

Section 9.5.2 is incorporated by reference into the NUHOMS® CoC 1030 Technical Specifications (paragraph 4.3.1) and shall not be deleted or altered in any way without a CoC amendment approval from the NRC. The text of this section is shown in bold type to distinguish it from other sections.

Neutron Transmission acceptance testing procedures shall be subject to approval by the Certificate Holder. Test coupons shall be removed from the rolled or extruded production material at locations that are systematically or probabilistically distributed throughout the lot. Test coupons shall not exhibit physical defects that would not be acceptable in the finished product, or that would preclude an accurate measurement of the coupon's physical thickness.

A lot is defined as all the pieces produced from a single ingot or heat. If this definition results in lot size too small to provide a meaningful statistical analysis of results, an alternate larger lot definition may be used, so long as it results in accumulating material that is uniform for sampling purposes.

The sampling rate for neutron transmission measurements shall be such that there is at least one neutron transmission measurement for each 2000 square inches of final product in each lot.

The B10 areal density is measured using a collimated thermal neutron beam of up to 1.2 centimeter diameter. A beam size greater than 1.2 centimeter diameter but no larger than 1.7 centimeter diameter may be used if computations are performed to demonstrate that the calculated $k_{effective}$ of the system is still below the calculated Upper Subcritical Limit (USL) of the system assuming defect areas the same area as the beam.

The neutron transmission through the test coupons is converted to B10 areal density by comparison with transmission through calibrated standards. These standards are composed of a homogeneous boron compound without other significant neutron absorbers. For example, boron carbide, zirconium diboride or titanium diboride sheets are acceptable standards. These standards are paired with aluminum shims sized to match the effect of

neutron scattering by aluminum in the test coupons. Uniform but non-homogeneous materials such as metal matrix composites may be used for standards, provided that testing shows them to provide neutron attenuation equivalent to a homogeneous standard.

Alternatively, digital image analysis may be used to compare neutron radioscopic images of the test coupon to images of the standards. The area of image analysis shall be up to 1.1 cm^2.

The minimum areal density specified shall be verified for each lot at the 95% probability, 95% confidence level or better. The following illustrates one acceptable method.

The acceptance criterion for individual plates is determined from a statistical analysis of the test results for their lot. The B10 areal densities determined by neutron transmission are converted to volume density, i.e., the B10 areal density is divided by the thickness at the location of the neutron transmission measurement or the maximum thickness of the coupon. The lower tolerance limit of B10 volume density is then determined, defined as the mean value of B10 volume density for the sample, less K times the standard deviation, where K is the one-sided tolerance limit factor with 95% probability and 95% confidence [7]. ***If a goodness-of-fit test demonstrates that the sample comes from a normal population, the value of K for a normal distribution may be used. Otherwise, use a non-parametric (distribution-free) method of determining one-sided tolerance limit.***

Finally, the minimum specified value of B10 areal density is divided by the lower tolerance limit of B10 volume density to arrive at the minimum plate thickness which provides the specified B10 areal density.

Any plate which is thinner than this minimum or the minimum design thickness, whichever is greater, shall be treated as non-conforming, with the following exception. Local depressions are acceptable, so long as they total no more than 0.5% of the area on any given plate, and the thickness at their location is not less than 90% of the minimum design thickness.

Non-conforming material shall be evaluated for acceptance in accordance with the Certificate Holder's QA procedures.

9.5.3 Specification for Qualification Testing of Metal Matrix Composites

9.5.3.1 Applicability and Scope

Metal matrix composites (MMCs) shall consist of fine boron carbide particles in an aluminum or aluminum alloy matrix. The ingot shall be produced by either powder metallurgy (PM), thermal spray techniques, or by direct chill (DC) or permanent mold casting. In any case, the final MMC product shall have a metallurgically bonded matrix. Boron carbide particles for the products considered here typically have an average size in the range 10-40 microns, although the actual specification may be by mesh size, rather than by average particle size. No more than 10% of the particles shall be over 60 microns. The material shall have negligible interconnected porosity exposed at the surface or edges.

Prior to initial use in a spent fuel dry storage or transport system, such MMCs shall be subjected to qualification testing that will verify that the product satisfies the design function. Key process

controls shall be identified per Section 9.5.4 so that the production material is equivalent to or better than the qualification test material. Changes to key processes shall be subject to qualification before use of such material in a spent fuel dry storage or transport system.

ASTM test methods and practices are referenced below for guidance. Alternative methods may be used with the approval of the certificate holder.

9.5.3.2 Design Requirements

In order to perform its design functions the product must have at a minimum sufficient strength and ductility for manufacturing and for the normal and accident conditions of the storage/ transport system. This is demonstrated by the tests in Section 9.5.3.4. It must have a uniform distribution of boron carbide. This is demonstrated by the tests in Section 9.5.3.5.

9.5.3.3 Durability

There is no need to include accelerated radiation damage testing in the qualification. Such testing has already been performed on MMCs, and the results confirm what would be expected of materials that fall within the limits of applicability cited above. Metals and ceramics do not experience measurable changes in mechanical properties due to fast neutron fluences typical over the lifetime of spent fuel storage, about 10^{15} neutrons/cm^2.

Thermal damage and corrosion (hydrogen generation) testing shall be performed unless such tests on materials of the same chemical composition have already been performed and found acceptable. The following paragraphs illustrate two cases where such testing is not required.

Thermal damage testing is not required for MMCs consisting only of boron carbide in an aluminum 1100 matrix, because there is no reaction between aluminum and boron carbide below 842°F, well above the basket temperature under normal conditions of storage or transport[3].

Corrosion testing is not required for MMCs consisting only of boron carbide in an aluminum 1100 matrix, because testing on one such material has already been performed by Transnuclear[4].

9.5.3.4 Required Qualification Tests and Examinations to Demonstrate Mechanical Integrity

At least three samples, one each from the two ends and middle of the test material production run shall be subject to:

a) room temperature tensile testing (ASTM- B557[5]) demonstrating that the material has the following tensile properties:

[3] Sung, C., "Microstructural Observation of Thermally Aged and Irradiated Aluminum/Boron Carbide (B4C) Metal Matrix Composite by Transmission and Scanning Electron Microscope," 1998

[4] Boralyn testing submitted to the NRC under docket 71-1027, 1998

[5] ASTM B557 Standard Test Methods of Tension Testing Wrought and Cast Aluminum and Magnesium-Alloy Products.

- Minimum yield strength, 0.2% offset: 1.5 ksi
- Minimum ultimate strength: 5 ksi
- Minimum elongation in 2 inches: 0.5%
  As an alternative to the elongation requirement, ductility may be demonstrated by bend testing per ASTM E290[10]. The radius of the pin or mandrel shall be no greater than three times the material thickness, and the material shall be bent at least 90 degrees without complete fracture.

and

b) testing (ASTM-B311[6]) to verify more than 98% of theoretical density *for non-clad MMCs and 97% for the matrix of clad MMCs*. Testing or examination for exposed interconnected porosity shall be performed by a means to be approved by the Certificate Holder.

9.5.3.5 Required Tests and Examinations to Demonstrate B10 Uniformity

CAUTION

Section 9.5.3.5 is incorporated by reference into the NUHOMS® CoC 1030 Technical Specifications (paragraph 4.3.1) and shall not be deleted or altered in any way without a CoC amendment approval from the NRC. The text of this section is shown in bold type to distinguish it from other sections.

Uniformity of the boron distribution shall be verified either by:

(a) Neutron radioscopy or radiography (ASTM E94[7], E142[8], and E545[9]) of material from the ends and middle of the test material production run, verifying no more than 10% difference between the minimum and maximum B10 areal density, or

(b) Quantitative testing for the B10 areal density, B10 density, or the boron carbide weight fraction, on locations distributed over the test material production run, verifying that one standard deviation in the sample is less than 10% of the sample mean. Testing may be performed by a neutron transmission method similar to that specified in Section 9.5.2, or by chemical analysis for boron carbide content in the composite.

9.5.3.6 Approval of Procedures

Qualification procedures shall be subject to approval by the Certificate Holder.

[6] ASTM B311, Test Method for Density Determination for Powder Metallurgy (P/M) Materials Containing Less Than Two Percent Porosity

[7] ASTM E94, Recommended Practice for Radiographic Testing

[8] ASTM E142, Controlling Quality of Radiographic Testing

[9] ASTM E545, Standard Method for Determining Image Quality in Thermal Neutron Radiographic Testing

[10] *ASTM E290, Standard Methods for Bend Testing of Materials for Ductility*

9.5.4.3 Identification and Control of Key Process Changes

CAUTION
Section 9.5.4.3 is incorporated by reference into the NUHOMS® CoC 1030 Technical Specifications (paragraph 4.3.1) and shall not be deleted or altered in any way without a CoC amendment approval from the NRC. The text of this section is shown in bold type to distinguish it from other sections.

The manufacturer shall provide the Certificate Holder with a description of materials and process controls used in producing the MMC. The Certificate Holder and manufacturer shall identify key process changes as defined in Section 9.5.4.2.

An increase in nominal boron carbide content over that previously qualified shall always be regarded as a key process change. The following are examples of other changes that may be established as key process changes, as determined by the Certificate Holder's review of the specific applications and production processes:

(a) Changes in the boron carbide particle size specification that increase the average particle size by more than 5 microns or that increase the amount of particles larger than 60 microns from the previously qualified material by more than 5% of the total distribution but less than the 10% limit,

(b) Change of the billet production process, e.g., from vacuum hot pressing to cold isostatic pressing followed by vacuum sintering,

(c) Change in the nominal matrix alloy,

(d) Changes in mechanical processing that could result in reduced density of the final product, e.g., for PM or thermal spray MMCs that were qualified with extruded material, a change to direct rolling from the billet,

(e) For MMCs using a 6000 series aluminum matrix, changes in the billet formation process that could increase the likelihood of magnesium reaction with the boron carbide, such as an increase in the maximum temperature or time at maximum temperature, and

(f) Changes in powder blending or melt stirring processes that could result in less uniform distribution of boron carbide, e.g., change in duration of powder blending.

(g) For MMCs with an integral aluminum cladding, a change greater than 25% in the ratio of the nominal aluminum cladding thickness (sum of two sides of cladding) and the nominal matrix thickness could result in changes in the mechanical properties of the final product.

In no case shall process changes be accepted if they result in a product outside the limits in Sections 9.5.3.1 and 9.5.3.4.

10.2 Radiation Protection Design Features

10.2.1 NUHOMS® HD System Design Features

The NUHOMS® HD System has design features which ensure a high degree of integrity for the confinement of radioactive materials and reduction of direct radiation exposures during storage. Those features are described in Section 10.1.2.

10.2.2 Offsite Dose Calculations

Calculated dose rates in the immediate vicinity of the NUHOMS® HD System are presented in Chapter 5, which provides a detailed description of source term configuration, analysis models and bounding dose rates. Off-site dose rates and doses are presented in this section. This evaluation determines the neutron and gamma-ray off-site dose rates including skyshine in the vicinity of the two generic ISFSI layouts containing design-basis contents in the DSCs.

The first generic ISFSI evaluated is a 2x10 back-to-back array of HSM-Hs loaded with design-basis fuel and control components (NFAH) in NUHOMS® 32PTH DSCs. The second generic layout evaluated is two 1x10 front-to-front arrays. This evaluation provides results for distances ranging from 6.1 to 600 meters from each face of the two arrays.

The total annual exposure for each ISFSI layout as a function of distance from each face is given in Table 10-2 and plotted in Figure 10-1. The total annual exposure estimates assume 100% occupancy for 365 days.

The Monte Carlo computer code MCNP 2 calculates the dose rates at the specified locations around the arrays of HSMs. The results of this calculation provide an example of how to demonstrate compliance with the relevant radiological requirements of 10 CFR 20 6, 10 CFR 72 5, and 40 CFR 190 8 for a specific site. Each site must perform specific site calculations to account for the actual layout of the HSMs and fuel source.

The assumptions for the MCNP analyses are summarized below.

- The 20 HSMs in the 2x10 back-to-back array are modeled as a box enveloping the 2x10 array of HSMs including the 3-foot shield walls on the two ends of the array. MCNP starts the source particles on the surfaces of the box.
- The 20 HSMs in the two 1x10 front-to-front arrays are modeled as two boxes which envelope each 1x10 array of HSMs including the 3-foot shield walls on the two ends and back of each array. MCNP starts the source particles on the surfaces of one of the boxes.
- The ISFSI approach slab is modeled as concrete. Because the ground composition has, at best, only a secondary impact on the dose rates at the detectors, any differences between this

that surface. The activity of each surface is determined by multiplying the sum of the normalized group fluxes, calculated above, by the average surface dose rate and by the area of the surface. This calculation is performed for the roof, sides, back and front of the HSM-H. The sum of the surface activities is then input as the tally multiplier for each of the MCNP tallies to convert the tally results to fluxes (particles per second per square centimeter).

- Neutron and gamma-ray spectra are shown in Table 10-3. The group fluxes on the roof are taken from the MCNP run. The dose rate contribution from each group is the product of the flux and the flux-to-dose factor. The "Input Current" column in the tables is simply the roof flux in each group, divided by half the total dose rate and represents the roof current normalized to one mrem per hour.

10.2.2.1 Activity Calculations

The surface activities are summarized in Table 10-4.

10.2.2.2 Dose Rates

Dose rates are calculated for distances of 6.1 meters (20 feet) to 600 meters from the edges of the two ISFSI designs.

Neutron and gamma-ray sources are placed on each surface using the spectra and activities determined above. The angular distribution of source particles is modeled as a cosine distribution. The contribution of capture gamma-rays has been neglected, as has the contribution of bremsstrahlung electrons. The inclusion of coherent scattering greatly increases the variance in a problem with point detector tallies without improving the accuracy of the calculation. Thus, coherent scattering of photons is ignored.

For the 2x10 back-to-back array with end shield walls, the "box" dimensions are 1260 cm wide, 3129 cm long, and 564 cm high.

For the two 1x10 front-to-front arrays with end and back shield walls, the "box" dimensions for each array are 721 cm wide, 3129 cm long, and 564 cm high. The two 1x10 arrays are 1026 cm (34 feet) apart.

Point detectors are placed at the following locations as measured from each face of the "box": 6.095 m (20 feet), 10 m, 20 m, 30 m, 40 m, 50 m, 60 m, 70 m, 80 m, 90 m, 100 m, 200 m, 300 m, 400 m, 500 m, and 600 m. Each point detector is placed 91.4 cm (3 feet) above the ground.

The MCNP results for each detector from the front of 2x10 back-to-back array are summarized in Table 10-5. The MCNP results as a function of distance from the back of the two 1x10 front-to-front arrays are summarized in Table 10-6. The MCNP results as a function of distance from the side of the 2x10 back-to-back array and the two 1x10 front-to-front arrays are summarized in Table 10-7.

The preceding analyses and results are intended to provide high estimates of dose rates for generic ISFSI layouts. The written evaluations performed by a licensee for an actual ISFSI must consider

12 OPERATING CONTROLS AND LIMITS

The original application for approval of the NUHOMS® HD System provided proposed Technical Specifications (TS) and TS Bases as Chapter 12 of this Safety Analysis Report (SAR). Upon approval of Certificate of Compliance (CoC) No. 1030, Amendment 0, dated January 10, 2007, the TS were listed as Appendix A of the CoC, and were therefore removed from this SAR with the issuance of Final Safety Analysis Report (FSAR) Revision 0, dated February 23, 2007.

LCO

Utilizing helium as the medium to assist during drainage of bulk water ensures that the fuel cladding remains under the limits during the entire vacuum drying operations.
A stable vacuum pressure of < 3 torr further ensures that all liquid water has evaporated in the 32PTH DSC cavity, and that the resulting inventory of oxidizing gases in the 32PTH DSC is below 0.25 volume %.

Technical Specification 3.1 requires the use of helium during the bulkwater removal process. Therefore, water from the DSC cavity is replaced by helium during the bulkwater removal process. Fuel cladding temperatures are low during this short duration process due to the presence of liquid water and helium.

Therefore use of helium during bulkwater removal, vacuum drying and long term storage operations assures that the fuel assemblies will have limited (or no) exposure to the oxidizing environment.

APPLICABILITY

This is applicable to all 32PTH DSCs *during LOADING OPERATIONS but before TRANSFER OPERATIONS.*

ACTIONS

The actions specified require *checking for any leaks in the vacuum drying system or welds and correcting them or* establishment of a helium pressure of at least 0.5 atmosphere within the time limits specified in the LCO. The timeframe specified applies to the vacuum drying operations and the helium backfill operations. If the required vacuum can not be established within the timeframe specified in the Condition column of the Actions table, a helium atmosphere (with a pressure of at least 0.5 atmosphere) is to be established within *30 days* or perform an assessment and implementation of corrective actions to return the 32PTH DSC to an analyzed condition or reflood the DSC submerging all fuel assemblies. The 15 psig limit in the action section is conservatively below the maximum analyzed blowdown pressure.

SURVEILLANCE REQUIREMENTS

Ensure a minimum oxidizing gas content *and maintain cladding integrity.*

REFERENCES

SAR Chapter*s* 3 and 4

Listing of the Files Contained in Enclosure 7
(All files are Proprietary)

Disk ID No. (size)	Discipline	System	File Series (topics)	Number of Files
DISK 1 (DVD) (3.65 GB)	Thermal	NUHOMS® HD	A001-README.txt (Instruction of Sample Input/Output Files in the Disk)	A001-A001 for a total of 1
			B001-BW17x17_corr.inp to B007-BW17x17_corr.out (Input and Output Files for 17x17 MkBW fuel assembly with un-irradiated UO_2 conductivity from HD SAR Section 4.8.2)	B001 to B007 for a total of 7
			C001-BW17x17_irr.inp to C007-BW17x17_irr.out (Input and Output Files for 17x17 MkBW fuel assembly with *irradiated UO_2 conductivity from the Ronchi's study)*	C001 to C007 for a total of 7
			D001-DSC1_corr.inp to D008-DSC1_corr.out (Input and Output Files for DSC thermal analysis, normal transfer in OS187H, HLZC#1 @ 115°F ambient – Keff for FA is calculated based on un-irradiated UO_2 conductivity from HD SAR Section 4.8.2)	C001 to C008 for a total of 8
			D001-DSC1_irr.inp to D008-DSC1_irr.out (Input and Output Files for DSC thermal analysis, normal transfer in OS187H, HLZC#1 @ 115°F ambient – Keff for FA is calculated based on irradiated UO_2 conductivity from the Ronchi's study)	E001 to E008 for a total of 8