DOCKET NO. 72-1004
TAC NO. L23343

TN INC. RESPONSES TO NRC REQUEST FOR ADDITIONAL INFORMATION

Chapter 1 General Information

Question 1.1

Quantify the risks (i.e., describe and assess the probabilities and consequences), in terms of the total dose received by the worker and the public, associated with the storage of high burnup fuel. Specifically, assess the risks for each condition or event, as analyzed in the FSAR, under normal, off-normal and accident conditions. For example, assess the risks associated with:

(a) the ability of the high burnup fuel to be handled and retrieved from the canister in accordance with 10 CFR 72.122(h) (5), 72.122(l) and 72.236(m);

(b) the storage system to perform its intended criticality, shielding, and confinement functions under normal conditions of storage in accordance with Subpart F, "General Design Criteria" of 10 CFR 72;

(c) the storage system to perform its intended criticality, shielding, and confinement functions under off-normal conditions of storage in accordance with Subpart F, "General Design Criteria" of 10 CFR Part 72; and

(d) the storage system to perform its intended criticality, shielding, and confinement functions under hypothetical accident conditions of storage in accordance with Subpart F, "General Design Criteria" of 10 CFR Part 72.

An acceptable analysis could be conducted to address the risks by considering a leak-before-break failure mechanism or potential fuel reconfiguration and their impacts on the integrity of the confinement boundary, shielding features, and criticality. This analysis could be used to demonstrate that any degradation of high burnup fuel during storage would not result in increased risk to the public.

The information in each of the previous Request For Additional Information (RAI) questions is necessary for the staff to verify compliance with the requirements of 10 CFR 72.122(h)(1), and 10 CFR 72.236, subparts (a), (c), (h), and (m).

Response – Question 1.1

The analysis documented in the Duke Topical Report, DPC-NE-2014P (TAC-L23369) [1] evaluated high burnup fuel cladding for a combination of conditions that include vacuum drying, transport to the HSM, blocked vent accident and 100 years of storage.

The results show that integrity of the cladding is maintained and cladding does not fail. Therefore, leak-before-break or potential fuel reconfiguration is highly unlikely during all storage conditions.

Normal operational design conditions consist of a set of events that are expected to occur routinely (FSAR Section 8.1). The probability of occurrence for off-normal operational events and conditions of storage is once per year (FSAR Section 8.1) as these are expected to occur with moderate frequency. The probability of occurrence for accident events, which include design basis accidents and design basis natural phenomena events, is once in the lifetime of ISFSI as these events are expected to occur very infrequently.

The following sections (a), (b), (c), and (d) of this response address Question 1-1 sections (a), (b), (c), and (d), separately

(a) In this section of the response to Question 1-1 (a), the cited regulations are addressed separately, as follows:

<u>10 CFR 72.122 (h) (5)</u>-

This regulation requires packaging of the spent fuel "in a manner that allows handling and retrievability without the release of radioactive materials to the environment or exposures in excess of part 20 limits" and "confinement of the high-level radioactive waste for the duration of the license." These requirements are addressed below.

Handling and retrievability: Section N.10 of the amendment application evaluates the dose consequences of normal conditions for the NUHOMS®-24PHB, including handling operations. While not explicitly evaluated, the expected dose consequences from fuel retrieval would be comparable to the dose consequences during fuel handling. Radioactive materials are not released to the environment during initial loading since known or suspected gross cladding breaches are prohibited. Radioactive materials will not be released to the environment during retrieval since the cladding remains intact (see following discussion of "Confinement of spent fuel for duration of license"). Additionally, operational controls described in Section 5.1.1.9 of the FSAR ensure that even if some of the fuel cladding were to fail during storage, any airborne radioactive material or fission gas inside the DSC would be appropriately managed well within the Part 20 limits.

Confinement of spent fuel for duration of license: Protection from the release of high level radioactive material during the storage term is provided by both the fuel cladding and the DSC. The fuel cladding remains intact at the end of the storage term as demonstrated in Reference [1], and subsequent RAI response submitted on May 24, 2002 (TAC-L23369) [2]. The DSC is designed, analyzed and tested to the "leak tight" criteria of ANSI N14.5-1997. Based on the analyses presented in

Sections N.3.6 and N.3.7, leak tightness is not compromised during normal, off-normal, or accident condition and the DSC maintains its confinement function.

10 CFR 72.122 (l)-

This regulation requires the storage system to provide for ready retrieval of the spent fuel. As discussed in the NRC's Interim Staff Guidance (ISG-2, Fuel Retrievability), this regulation is satisfied when the applicant demonstrates that the storage system is found by the NRC to be in compliance with 10 CFR Part 72 (i.e., designed to allow for decommissioning and has a limited license term). The NUHOMS® storage system has been previously found by the NRC to satisfy these conditions, and the amendment introduces no new considerations that would negate NRC's previous findings. Thus, this regulation is satisfied.

10 CFR 72.236 (m)-

This regulation requires that the storage system design consider removal of the spent fuel from the reactor site, transportation, and disposition by DOE. As discussed above, the NUHOMS®-24PHB provides for removal of the intact fuel at the end of the storage term. Once it is removed, it can be loaded into an NRC-approved transportation cask for transport to DOE and ultimate disposition. Thus, this regulation is satisfied.

(b) In this section of the response to Question 1-1 (b), the criticality, shielding, and confinement functions of the NUHOMS®-24PHB under normal conditions of storage are discussed.

Criticality: Section N.3.6 demonstrates that the NUHOMS®-24PHB DSC shell and basket remain intact (hence, subcritical) during normal conditions of storage. Thus, there would be no dose consequences.

Shielding: Section N.10 of the amendment appliaction quantifies the dose consequences to the workers and the public for storage of high burnup fuel under normal conditions in the NUHOMS®-24PHB system.

Confinement: The NUHOMS®-24PHB DSC is designed, analyzed and tested to the "leak tight" criteria of ANSI N14.5-1997. Based on the analysis presented in Sections N.3.6 and N.3.7, leak tightness of the DSC is not challenged during any normal, off-normal or accident condition and the DSC maintains its confinement function.

(c) In this section of the response to Question 1-1 (c), the criticality, shielding, and confinement functions of the NUHOMS®-24PHB under off-normal conditions of storage are discussed.

Criticality: Section N.3.6 demonstrates that the NUHOMS®-24PHB DSC shell and basket remain intact (hence, subcritical) during off-normal conditions of storage. Thus, there would be no dose consequences.

Shielding: Section N.11.1 of the amendment application quantifies the dose consequences to the workers and the public for storage of high burnup fuel under off-normal conditions in the NUHOMS®-24PHB system.

Confinement: The NUHOMS®-24PHB DSC is designed, analyzed and tested to the "leak tight" criteria of ANSI N14.5-1997. Based on the analysis presented in Sections N.3.6 and N.3.7, leak tightness of the DSC is not challenged during any normal, off-normal or accident condition and the DSC maintains its confinement function.

(d) In this section of the response to Question 1-1 (d), the criticality, shielding, and confinement functions of the NUHOMS®-24PHB under hypothetical accident conditions of storage are discussed.

Criticality: Section N.3.7 demonstrates that the NUHOMS®-24PHB DSC shell and basket remain intact (hence, subcritical) during all postulated accidents and environmental phenomena. Thus, there would be no dose consequences.

Shielding: Section N.11.2 of the amendment application quantifies the dose consequences to the workers and the public for storage of high burnup fuel under accident conditions in the NUHOMS®-24PHB system.

Confinement: The NUHOMS®-24PHB DSC is designed, analyzed and tested to the "leak tight" criteria of ANSI N14.5-1997. Based on the analysis presented in Sections N.3.6 and N.3.7, leak tightness of the DSC is not challenged during any normal, off-normal or accident condition and the DSC maintains its confinement function.

Question 1.2

Quantify the percentage of fuel pin failures, and the associated uncertainty, caused by stresses imparted to the high burnup fuel under hypothetical accident conditions on a per storage cask basis.

Since the applicant is requesting to store high burnup fuel with spalled rods (with a calculated maximum hoop stress of 66% of the yield strength), the effect of any additional stress on the cladding due to the stress state imparted to the cladding under a hypothetical drop accident should be additive to the hoop stress at the temperature of the cladding during the hypothetical accident. The staff considers the approach used in Appendix III of the SAND-90-2406 report, "A Method for determining the Spent Fuel Contribution to Transport Cask Containment Requirements," to be acceptable to quantify the percentage of fuel pin failures under hypothetical accident conditions of storage. Since there is limited data for the fracture toughness and mechanical properties of high burnup fuel, the applicant should consider using a range of properties (e.g., fracture toughness, ductility, etc.) and fuel characteristics (e.g., critical flaw size, reduction in wall thickness due to oxidation and hydride rim formation) consistent with properties and characteristics that are available in the literature including the data that *have measured degraded properties of high burnup fuel. A list of references should be provided along with the analysis.*

Response – Question 1.2

Reference [1] addresses fuel cladding integrity during normal and off-normal conditions of storage. Evaluation of the fuel cladding integrity under accident conditions is not required. Refer to NRC Staff guidance in ISG-3 that excludes the effects of accident on fuel, as the governing criterion is containment integrity. (Note that Reference [1] included the blocked vent accident for conservatism, even though accident events are not required to be analyzed.)

The NUHOMS®-24PHB DSC is designed, analyzed and tested to the "leak tight" criteria of ANSI N14.5-1997. Based on the analysis presented in Sections N.3.6 and N.3.7 of the amendment application, leak tightness of the DSC is not challenged during any normal, off-normal or accident condition and the DSC maintains its containment integrity.

The maximum DSC internal pressure analysis documented in Section N.4.6.4 of the amendment application assumes 100% of the fuel pins from all fuel assemblies rupture during accident conditions. The results documented in Table N.4-7 of the amendment application show that DSC internal pressures are all below the design pressures.

Question 1.3

Provide an attachment which describes any confirmatory testing, mechanical properties, fracture toughness measurements, or high burnup fuel characterizations that may need to be done to support the claims made in the analysis described in 1.2. The attachment should include (as applicable): an overview of the facility, testing apparatus, type and number of samples, testing orientations, pressure, temperature, and any other relevant data, Also, provide a discussion of how the data will be used to support the above analyses based on any theoretical discussions used to respond to questions 1.1 and 1.4 of this RAI.

Response – Question 1.3

Duke Energy's RAI response [2] to Question B.6 addresses this question. Please refer to the Reference [2] for the response to this question.

Question 1.4

In relations to Question 1-1(a), assess whether the phenomenon of high burnup fuel cladding "unzipping" will affect the risks to the worker and public during handling and retrieval operations. Spent fuel cladding unzipping is a phenomenon where the fuel cladding splits in the axial direction of the cladding by excessive spent fuel oxidation. Vacuum drying involves relatively high temperatures and steam conditions that are attributed to this phenomenon.

Response – Question 1.4

Thermodynamic considerations restrict the amount of UO_2 that could be oxidized, which limits the amount of pellet swelling that could occur. Reference [1] uses these thermodynamic considerations to demonstrate that clad unzipping will not occur during dry storage. Please refer to Reference [1] for a complete discussion of the analysis.

References to Chapter 1 RAI Responses

[1] "Fuel Rod Analysis for Dry Storage of Spent Nuclear fuel," Report DPC-NE-2014P, Duke Energy, August 2001, TAC L23369.

[2] Letter from K. S. Canady (Duke Energy) to L. R. Wharton (NRC-SFPO), Responses to Request for Additional Information by NRC Staff, May 24th, 2002, TAC-L23369.

Chapter 3 Structural Analysis

Question 3.1

Table J.4-2 on page J.4-4 (FSAR) and Table J.4.3 on page J.4.5 (FSAR) provide an internal pressure summary for the 24P Dry Shielded Canister (DSC) with burnable poison rod assemblies (BPRAs) for off-normal and accident cases for 40 and 45 GWd/MTU burnup pressures. The results show that the internal burnup pressure slightly exceeds the design basis pressure for the off-normal case, and are very close to design basis pressure with little safety margins for the accident case. Demonstrate that the internal burnup pressure corresponding to 55 GWd/MTU under off-normal and accident cases are within the design basis pressure limits.

This information is required for the staff to assess compliance with 10 CFR 72.24(c)(2).

Response – Question 3.1

In Appendix N.3, the 24PHB DSC is analyzed for higher design pressures during normal, off-normal and accident conditions than the 24P DSC with control components documented in Appendix J. The design pressures for the 24PHB DSC are 15 psig, 20 psig and 68 psig for normal, off-normal and accident conditions, respectively, as shown in Table N.4-7. The calculated maximum internal pressures for 55 GWd/MTU burnups are 6.3 psig, 11.2 psig, and 63.1 psig for normal, off-normal and accident conditions, respectively, as shown in Table N.4-7. These calculated pressures are significantly below the design basis pressures for all normal, off-normal and accident conditions.

Question 3.2

The 24PHB DSCs have been designed for operation at an ambient temperature as low as –40°F. Provide data and analyses to confirm the ductile behavior of the spacer discs, which are made of SA-516 material. Also show that brittle fracture is not a concern for the spacer discs over the range of operating temperatures.

This information is required for the staff to assess compliance with 10 CFR 72.122 which requires that the cold temperature behavior for materials be provided.

Response – Question 3.2

FSAR Table 8.1-3, Note 8, requires that SA-516 spacer disc material be qualified by impact testing as specified in ASME Code, Subsection NF, Article NF-2300. The specified impact test temperature is -20°F. The NRC staff evaluated these brittle fracture criteria and the results of the evaluation are contained in Section 3, page 3-3, of the CoC 72-1004 SER. As described in Section 3 of the SER, the staff imposed certain limiting conditions of operation for handling and transfer of the DSC when basket temperatures are below 0°F. These limiting conditions are incorporated in NUHOMS® CoC 1004 Technical Specification 1.2.13.

Question 3.3

Demonstrate that the gaps between the 24PHB DSC shell and the spacer discs do not close due to different thermal expansion of the materials considering the ambient temperature (100°F) condition together with a side drop accident.

Under the ambient temperature of 100°F, consider the diametrical growth of the spacer discs and the radial shell growth. The cask side drop would further reduce the gap between the spacer disc and the shell.

This information is required for the staff to assess compliance with 10 CFR 72.146 pertaining to design control

Response – Question 3.3

The gap between the 24HB DSC shell and spacer disc under normal thermal and accident side drop condition loads is computed using the following steps:

Step 1: Compute the DSC shell maximum inward displacement under 75g side drop load.

The maximum displacement in the radial direction of the DSC shell for the 75g side drop case is obtained from the ANSYS analysis models of the DSC shell, shown in Figures 8.1-14a and 8.1-14b of the FSAR. The maximum shell displacement occurs at the mid height of the shell (i.e., at the free ends of the half-shell models shown in Figures 8.1-14a and 8.1-14b). The maximum displacements from the analysis (conservatively assumed to be in the inward direction) are 0.16 inches (node 1935 for the top-end half-length model) and 0.12 inches (node 1671 for the bottom-end half-length model)

Step 2: Compute the spacer disc maximum diametrical displacement from the spacer disc under 75g side drop load.

The maximum displacement in the radial direction of the spacer disc for the 75g side drop case is obtained from the ANSYS analysis model of the 24P spacer disc shown in Figure 8.2-4 of the FSAR. The maximum reduction in diameter is 0.225 inches. This is considered the deflection of spacer disc edge inward.

Step 3: Compute spacer disc thermal growth for Heat Load Zoning Configuration #1 and #2 with ambient of 100°F in the Cask.

The spacer disc thermal growth analysis ANSYS model, shown in Figure N.3.6-1 of the amendment application, is used to obtain the maximum diametrical growth of the spacer disc for Heat Load Zoning Configurations 1 and 2. This is 0.25 inch diametrical growth.

Step 4: The resulting gap is computed using following relationship:

$$gap = Gap\text{-}cold - \Delta sdth + \Delta sd75g - \Delta sh75g$$

where:

gap = Gap between DSC shell and spacer disc.

Gap-cold = Cold gap between DSC shell and spacer disc. This is the difference between the ID of the shell and OD of the spacer disc at room temperature, [67.19 - 2(0.625)] - 65.50 = 0.44 inches.

Δsdth = Spacer disc thermal growth. From the ANSYS analysis of the spacer disc, the maximum diametrical thermal growth of the spacer disc for the case of the DSC in the cask and ambient temperature of 100°F is 0.25 inches.

Δsd75g = Spacer disc maximum diametrical displacement under side drop load is obtained from the 75g side drop analysis of the spacer disc and is 0.225 inches.

Δsh75g = DSC Shell maximum displacement under 75g drop load is 0.16 inches.

Therefore, the minimum gap between the spacer disc and DSC shell subjected to normal thermal and 75g side drop load :

$$gap = 0.44 - 0.25 + 0.225 - 0.16 = 0.255 \text{ in.}$$

The gaps between the 24PHBS and 24PHBL (for storing high burn up fuel) DSC shell and spacer discs do not close due to differential thermal expansion of the materials considering the 100°F ambient temperature condition together with a 75g side drop accident.

Question 3.4

Provide Table N.3.7.5, which reports the inner and outer bottom cover plate stresses for the 24PHBS DSC (standard). The table in the current text appears to be missing.

This information is required for the staff to assess compliance with 10 CFR 72.152 which enables the staff to compare cover plate stresses for the 24PHBL DSC (long) and the 24PHBS DSC.

Response – Question 3.4

Table N.3.7.5 is a table of notes only that are common to Tables N.3.7.2, N.3.7.3 and N.3.7.4. No information is missing from this table. The stresses for the 24PHBS DSC inner and outer bottom cover plates are reported in Tables N.3.7.2, N.3.7.3 and N.3.7.4.

Chapter 4 Thermal

Question 4.1

Modify the FSAR to include the proper terminology when discussing spent fuel cladding temperatures.

Section N.4-1 uses the term "maximum fuel cladding temperature," when in fact a maximum homogenized fuel region temperature is reported for all heat transfer calculations performed for this System. This information is needed to assure compliance with 10 CFR 72.11 and 72.236

Response – Question 4.1

Each fuel assembly in the basket is homogenized in its guide sleeve region and effective fuel properties are used for the homogenized fuel assembly region to calculate maximum fuel temperature. This maximum fuel temperature correlates to the maximum fuel cladding temperature based on the validation of fuel effective conductivity values used for the NUHOMS® system design as documented in Appendix B.3 of the FSAR. Section N.4.1 is revised to add this clarification. This method was previously reviewed by the NRC for the NUHOMS-24P Topical Report [4] which was approved by the NRC in its SER of April 1989 [5].

In addition, alternate confirmatory analysis [3] has been performed to benchmark the maximum fuel cladding temperature calculation methodology. The alternate thermal analysis methodology uses different computer codes and analytical modeling techniques to calculate maximum fuel cladding temperatures. This alternate analysis methodology is also validated against the measured test data from systems similar to the NUHOMS®-24PHB system. This alternate confirmatory analysis (NUH-HBU.0403) is submitted to the NRC as a proprietary attachment to this submittal.

Question 4.2

Provide references for all material properties listed in Chapter N.4. Demonstrate that all properties have been obtained from experimental data and that extrapolation of thermal properties has not been performed. This is not provided and is needed to assure compliance with 10 CFR 72.11 and 72.236

Response – Question 4.2

Section N.4.2 is revised to add the requested information.

Question 4.3

State all gap distances used in the DSC model and explain the basis for determining these gap distances.

Section N.4.4.1.1 delineates a bounding heat conductance uncertainty between adjacent components using conservative gaps. This information is needed to assure compliance with 10 CFR 72.11 and 72.236.

Response – Question 4.3

The gaps used in the thermal analysis of the 24PHB DSC are summarized in the table below. When the DSC is in a horizontal orientation in the transfer cask or the HSM, the spacer discs contact the bottom of the DSC shell and the guide sleeves contact the spacer disk ligaments. However, to be conservative, uniform gaps are used in the DSC thermal analysis between these components. The gap sizes used in the DSC thermal analysis are hot condition gaps that account for the thermal expansions of these components. The size and location of gaps are given below:

Gap Number	Gap Location	Gap Size, Inches
1	DSC Shell and spacer discs	0.195
2	Guide sleeves and spacer disc ligaments	0.060
3	Outer guide sleeves and DSC shell	Non-uniform
4	Gaps between adjacent guide sleeves at the center, middle and outer locations in the basket	1.25, 1.00, 0.75

All heat transfer across the gaps is by gaseous conduction. Radiation heat transfer is considered between the DSC shell, spacer discs and guide sleeves. Heat removed by convection heat transfer in all these gaps is conservatively neglected.

Question 4.4

Provide a reference for how homogenized thermal properties used in the analysis are obtained.

For the thermal models described on Page N.4-5, it states that these models simulate the effective thermal properties of the fuel with a homogenized material occupying the volume within the basket. This is not provided and is needed to assure compliance with 10 CFR 72.11 and 72.236.

Response – Question 4.4

As discussed in response to RAI Question 4.2, the effective thermal properties for the homogenized fuel region are obtained from Appendix B.2 and revised Section N.4.2.

Question 4.5

Provide the procedure for determining heat insolation values for all the calculations using solar heat as input.

The discussion provided in Section N.4.4 and elsewhere in Chapter N.4 is insufficient. This is not provided and is needed to assure compliance with 10 CFR 72.11 and 72.236.

Response – Question 4.5

The solar insolation values used are 123 BTU/hr-ft^2 for off-normal and accident conditions and 62 BTU/hr-ft^2 for normal conditions of storage. These values are the same as those used in the FSAR Section 8.1.3.1C, page 8.1-38 for the HSM and Section 8.1.3.3.1A, page 8.1-47 for the Transfer Cask analyses. Note that the DSC basket component and fuel cladding temperatures are not very sensitive to the solar insolation values because of the large thermal mass of the NUHOMS® HSM and transfer cask.

Question 4.6

Modify the SAR to clearly describe what maximum heat load is requested for the DSC, and remove all references to other heat loads.

The heat load requested should be the heat load for which the DSC is analyzed. Analyzing heat loads other than those requested is unnecessary, and can become confusing in the SAR.

Section N.4.4.1.2 states that while a heat load of 26 kW/DSC is analyzed, the payload is limited to a maximum decay heat load of 24 kW/DSC. If 26 kW/DSC is used for this configuration, this invalidates the 24 kW horizontal storage module (HSM) and transfer cask (TC) boundary conditions used for the thermal calculation. This information is needed to assure compliance with 10 CFR 72.24(c)(3).

Response – Question 4.6

The maximum heat load for the NUHOMS® 24PHB is 24kW per DSC for either Configuration 1 or Configuration 2. This statement is added to Section N.4.4.1.2. Reference to 26 kW is deleted from Section N.4.4.1.2.

Question 4.7

Provide a reference for the DSC shell temperatures as used in Section N.4.4.1.2.

Page N.4-6 (Section N.4.4.1.2) states that "the DSC shell temperatures calculated with the 24 Kw HSM and TC models are used as boundary conditions for the Configuration 2 DSC/basket model." This information is needed to assure compliance with 10 CFR 72.11 and 72.236.

Response – Question 4.7

The requested information is included in Table N.4-8 of Section N.4.4.1.2.

Question 4.8

Modify the reference to the TC Thermal Model of Section N.4.4.1.4 to be more specific.

This information is needed to assure compliance with 10 CFR 72.11 and 72.236.

Response – Question 4.8

Section N.4.4.1.4 is revised to reference FSAR Section 8.1.3.3.1 for the DSC temperatures used in the transfer cask thermal model.

Question 4.9

Briefly describe the methodology used to obtain the maximum fuel cladding temperatures during long term storage, short-term loading/unloading and accident conditions for burnups greater than 30 GWD/MTU, and for short-term loading/unloading, transfer and accident conditions.

Section N.4-1 states that the maximum fuel cladding temperatures during long term storage, short-term loading/unloading and accident conditions for burnups greater than 30 GWD/MTU are evaluated using the methodology described in DPC-NE-2013P ("Fuel Rod Analysis for Dry Storage if Spent Nuclear Fuel", Duke Power Company, August 31, 2001). This information is needed to assure compliance with 10 CFR 72.24(d) (1).

Response – Question 4.9

Maximum fuel cladding temperatures are calculated as described in Section N.4 for long-term storage, loading and transfer operations, off-normal conditions and accident conditions. The maximum fuel cladding temperatures are calculated for the maximum decay heat loading for Heat Load Zoning Configurations 1 and 2, as illustrated in Figures N.4.5 and N.4.6. A time-temperature history is developed for the fuel cladding for the following sequences: fuel loading (including vacuum drying); transfer to the HSM; a blocked vent accident occurring immediately after loading into the HSM; and a 100-year storage term. The analysis of the blocked vent accident immediately after HSM loading is conservative since it produces higher fuel cladding temperatures than would result at a later time during storage.

The analysis, documented in Duke Topical Report: DPC-NE-2014P (TAC-L23369)[1], evaluated high burnup fuel cladding for the time-temperature history described above. This topical report establishes that creep strain is the controlling mechanism for cladding degradation in long-term dry storage. It also establishes creep limits and a method to calculate creep strain as a function of fuel cladding temperature history and numerous other parameters more fully described below. It is important to note that the only temperature limit is the short-term limit of 752°F (400°C) to avoid annealing of the fuel cladding. The initial storage temperature of the fuel cladding and its temperature history are not considered limits since variations could produce acceptable creep strain during the storage term. Reference [1] provides the methodology by which other fuel cladding temperature histories may be evaluated for acceptable creep strain.

The methodology used in Reference [1] is summarized below:

Strain is based on the combination of creep strain and thermal strain, which is referred to as total strain. The total strain is calculated by DRYCASK. DRYCASK uses nominal fuel rod design data to determine the parameters for an unirradiated rod at room temperature. DRYCASK uses nominal dimensions for pellet diameter and density, inside and outside cladding diameters, fuel stack and cladding lengths, plenum volume, information on dish and chamfer volumes, pellet resintering density change, and residual gas and helium fill gas pressures. These values are determined from manufacturing specifications, drawings, or design data packages.

DRYCASK needs several additional values to determine the characteristics for an irradiated rod at end-of-life. An in-reactor fuel rod performance code is used to calculate conservative values for the inventory of krypton and xenon fission gases and the cladding oxide thickness. Alternatively, the oxide thickness could be measured directly. Fast fluence and rod-average burnup are obtained using a fuel rod performance code, physics code, or other approved method.

The Duke RAI response, for DPC-NE-2014P, which was submitted to NRC on May 24, 2002 (TAC-L23369)[2] includes a calculation package for the strain analysis. This calculation demonstrates the application of Reference [1] methodology and gives conservatisms applied to achieve a bounding analysis.

Question 4.10

Provide a copy of Reference 4.9.

This information is needed to assure compliance with 10 CFR 72.11 and 72.236.

Response – Question 4.10

A copy of Reference 4.9 is included with these responses.

Question 4.11

Provide the calculation for the amount of fission gas released into the 24PHB DSC cavity from the fuel and all control components for normal, off-normal and accident conditions cases.

A summary of the amount of fission gas released into the DSC is provided in Sections N.4.4.4.1 and N.4.4.4.2. Further detail is needed. This information is needed to assure compliance with 10 CFR 72.11 and 72.236.

Response – Question 4.11

Additional information on calculations of fission gas release is included in revised Sections N.4.4.4.1 and N.4.4.4.2.

Question 4.12

Provide the calculation of the DSC cavity free volume used in different sections of Chapter N.4 for calculating the maximum internal pressure.

This information is needed to assure compliance with 10 CFR 72.11 and 72.236.

Response – Question 4.12

The DSC cavity free volume is calculated as the volume of DSC cavity minus the volume occupied by guide sleeves, oversleeves, spacer discs, support rods and fuel assemblies including control components. The calculations are shown below:

Component	**Volume (in^3)**
DSC Cavity	570,404
Guide Sleeves (24)	-14,750
Oversleeves (24)	-2,203
Spacer Discs (8)	-20,312
Support Rods (4)	-5,525
Fuel Assemblies with Control Components (24)	-142,844
Support Ring (1)	-307
Total	384,463

Table N.4-7 shows that calculated pressures are significantly below the design basis pressures used in the structural analysis documented in Section N.3. Therefore, any minor change to the cavity free volume would have a negligible impact on the calculated pressures and no impact on the design basis pressures.

Question 4.13

Evaluate the difference between using the ideal gas law and a partial pressure relationship for obtaining the maximum normal operating pressure. Clarify how the approach used meets the guidance in the SRP Section 5c, Pressure Analysis (NUREG-1536).

This information is needed to assure compliance with 10 CFR 72.24(d).

Response – Question 4.13

The pressure within the DSC is the sum of partial pressures from each of the gas constituents in the DSC cavity. The gases in the DSC that contribute to the internal pressures are helium, xenon, krypton, and tritium. The atmosphere is dry and inert in the DSC cavity after vacuum drying and backfilling with helium. The maximum temperatures and pressures for gases in the DSC cavity for all normal, off-normal and accident conditions are less than 650°F and 68 psig respectively. In this operating range, the gases in the DSC cavity behave like ideal gases and follow the ideal gas law. Therefore, pressures calculated using ideal gas law or a partial pressure relationship would be the same, thus satisfying the requirements of NUREG-1536.

Question 4.14

Provide an explanation of how 10 CFR 72.212(b)(2) relates to Section N.4.6.

Page N.4-14 (Section N.4.6) references to 10 CFR 72.212(b)(2) (potential fires and explosions). The relationship of this regulation to the section in question is unclear. This information is needed to assure compliance with 10 CFR 72.11 and 72.236.

Response – Question 4.14

Certificate of Compliance, CoC 1004, Technical Specifications, General Requirements and Conditions, Section 1.1.1, Condition 5 has the requirement that potential for fire and explosion should be addressed based on site-specific considerations. Therefore, Section N.4.6 includes the reference to 10 CFR 72.212(b)(2) for fire and explosion events.

Question 4.15

Clarify how the short term events are defined in Section N.4.1 as stated in the last paragraph of Section N.4.6.2.

Section N.4.1 does not appear to define short term events. This information is needed to assure compliance with 10 CFR 72.11 and 72.236.

Response – Question 4.15

Definition of short-term events is added to Section N.4.6 and reference to Section N.4.1 is deleted.

Question 4.16

Justify the maximum calculated homogenized fuel region temperature for normal, off-normal, and accident events given that the maximum calculated temperatures for these conditions are approaching or equal to the maximum permitted temperatures for these conditions.

Given that homogenization of the fuel tends to reduce the temperature peaks that exist across the fuel assemblies, a calculated maximum temperature equal to the temperature limit is unacceptable. A more precise fuel model should be employed, or a complete *error analysis should show that the value obtained, accounting for errors, is below the temperature limit. This information is needed to assure compliance with 10 CFR 72.24(d).*

Response – Question 4.16

As discussed the response to RAI Question 4.1, the methodology provides conservative predictions of actual fuel cladding temperatures, based on experimental validation of the methodology and also by the alternate confirmatory analysis performed. Moreover, as discussed in the response to RAI Question 4.9, fuel cladding temperature is not considered the limiting parameter except for the short-term temperature limit of 752°F (400°C) to preclude annealing. Rather, the limiting parameter is creep strain, which is calculated by the methodology presented in the Reference [1]. With this in mind, it is appropriate to identify not only the conservatisms employed in the calculation of the maximum fuel cladding temperature, but in the calculation of the creep strain, as well.

Conservative assumptions included in the calculation of maximum cladding temperatures are described below:

1) The cladding temperature limit for fuel burned to less than 30 GWD/MTU is calculated based on methodology given in PNL-6189 report. In this report, it is documented (Appendix C, page C.8 of PNL-6189) that there is a margin of 32°C in CSFM model prediction of cladding temperature limit (408°C) when compared to the limit predicted by test data (450°C) at Federal Republic of Germany (FRG) at approximately 40 MPa cladding hoop stress because of assumed faster creep rate in the CSFM model.

2) The PNL-6189 report methodology to calculate cladding temperature limit is based on a fuel rod failure probability of less than 0.5% (5 in 1,000) if the cladding temperature limit is exceeded. The 24PHB DSC design assumes failure of 1%, 10% and 100% of all the rods from all the assemblies during normal, off-normal and accident conditions, respectively. The maximum cladding temperature occurs in the fuel assembly that is closest to the center of the basket. The difference in maximum cladding temperature between the fuel assemblies at the center of the basket and assemblies in the middle and outer periphery of the basket are approximately 25°F to 145°F, respectively. Therefore, there is a significantly greater margin in the calculated cladding temperature for the majority of the fuel assemblies in the 24PHB DSC.

3) Credit for any convection in the DSC basket cavity is not taken.

4) Conservative gaps are assumed between basket components as described in response to RAI Question 4.3.

Conservative assumptions included in the calculation of creep strain are described below:

1) A significant margin is demonstrated between the calculated creep strain for the analyzed fuel rod with the maximum cladding temperatures and the creep strain limit. Please refer to the sample calculation included in Reference [2].

2) The creep strain calculation includes consideration of a blocked vent accident occurring at the worst time in the storage term (at the beginning). This accident is conservatively included to demonstrate margin, even though demonstration of cladding integrity is not required for accident conditions per NRC Staff guidance provided in ISG-3.

3) The creep strain calculation includes consideration of a 100-year storage term, while the licensed storage term is 20 years.

4) A bounding fission gas pressure is used for the analyzed rod.

5) The DRYCASK code conservatively under-predicts fuel rod void volumes, which results in a conservative stress value due to higher pressure.

6) A thin membrane cladding stress formulation is used to evaluate cladding stresses.

7) Fuel rod internal volume calculation assumes that the cladding has no external oxide formation. This minimizes the void volume increase during the cladding creep-out process, which results in a conservative stress value due to higher pressure.

8) Conversely, the cladding stress calculation assumes the maximum oxide thickness over the entire rod length.

Based on these conservatisms, there is higher margin in both the calculated maximum cladding temperatures and the creep strain associated with the maximum temperature rod.

Question 4.17

Correct the apparent discrepancy between Page N.4-18 and Figure N-4-16.

Page N.4-18, Section N.4.7.1.1 states that Figure N-4.16 provides the temperature distribution within the basket at the end of the 36 hour vacuum drying transient but Figure N.4-16 caption states that this distribution is obtained at 35 hours.

This information is needed to assure compliance with 10 CFR 72.11 and 72.236.

Response – Question 4.17

The Section N.4.7.1.1 is corrected to state that Figure N.4-16 is for 35 hours instead of 36 hours. Figure N.4-6 is correct. Note that the values in Table N.4-6 are based on temperatures at the end of 36 hours.

Question 4.18

Explain how the maximum temperature limit of the fuel can be reached without affecting the integrity of the fuel cladding or describe any imposed operational technical specification that is in place to prevent exceeding fuel clad temperature limits.

Response – Question 4.18

As previously discussed in the responses to RAI Questions 4.1, 4.9, and 4.16, the calculated fuel cladding temperature for the high burnup cladding is not considered a limit. The actual limit is the cladding creep strain which is influenced not only by temperature, but also by a number of other fuel rod parameters. The conservatisms in the calculation of the maximum cladding temperature and the creep strain are listed in the response to RAI Question 4.16.

Technical Specification 1.2.1 and corresponding Tables 1-1h, 1-2n, 1-2o, 1-2p and Figures 1-5 and 1-6, limit the decay of the stored assemblies and the location in which they may be placed within the DSC basket. This limits the fuel cladding temperature to the analyzed range to ensure that the cladding does not fail during the storage term.

Technical Specification 1.2.17a imposes a time limit on the duration of vacuum drying operation to ensure that the maximum fuel cladding temperature does not exceed 752°F, which would have an uncertain impact on the cladding integrity. In addition, Technical Specification 1.3 is also modified to impose a maximum duration of 34 hours for blockage of HSM air inlet and outlets when a 24PHB DSC is stored within an HSM.

Question 4.19

Explain the heat transfer mechanisms considered when performing the analysis of the hypothetical fire accident, and discuss how these mechanisms adequately model the fire environment.

Section N.4.6.3 does not provide an adequate description of how different heat transfer mechanisms are included in the hypothetical accident modeling approach. This information is needed to assure compliance with 10 CFR 72.11 and 72.236.

Response – Question 4.19

The fire transient analysis presented in Section N.4.6.3 is based on very conservative assumptions. It is assumed that liquid neutron shield (water) is present throughout the 15-minute fire transient even though it is expected to be lost and replaced with air very early in the fire transient. This assumption maximizes the heat input from the fire to the canister because of the high thermal conductivity of water compared to air. To maximize the canister temperature during the post-fire transient, it is assumed that water in neutron shield cavity is lost at the beginning of post-fire transient and replaced by air as the heat flow is now from canister to the ambient.

The gaps included in the thermal model of the 24PHB DSC are summarized in the response to RAI Question 4.3. These gaps are not removed for calculating the cladding temperatures during accident conditions. The canister shell temperatures increase by a negligibly small amount (<0.3°F) during fire transient. This increase is small during fire transient as the canister is protected due to the large thermal mass of the transfer cask. This shows that heat input from the fire to the canister is not significant. Since the canister shell temperature is almost unchanged, the cladding temperatures during 15-minute fire transient also are almost unchanged. Therefore, the assumption of not removing the gaps during fire transient has negligible impact on the cladding temperatures.

Question 4.20

Provide the mesh sensitivity studies performed that demonstrate for all finite element analysis (FEA) models that the results of the calculated temperatures have converged and are not mesh-dependent (i.e., that temperature distributions do not change when the element mesh is refined).

This information is needed to assure compliance with 10 CFR 72.24(d).

Response – Question 4.20

The mesh sensitivity study is performed by reducing the total number of 3D Solid70 elements in the model to 96% (from 20736 to 19860 solid elements) and to 68% (from 20736 to 14064 solid elements). The 70°F ambient temperature long term storage case is considered for this study. A Jackobi conjugate gradient iterative equation solver is applied for analysis and the solution converged after 5 equilibrium iterations for the 68% mesh case and after 4 equilibrium iterations for the 96% and 100% mesh cases.

All material properties, component dimensions, and boundary conditions are the same as those used in the original model. The component temperature differences between the models are listed in the following table.

Component	**SAR Model (100% Solid Elements)**	**96% Solid Elements Model**	**Maximum Temperature Difference $T_{SAR}-T_{96\%}$**	**68% Solid Elements Model**	**Maximum Temperature Difference $T_{SAR}-T_{68\%}$**
	T_{max}, °F	T_{max}, °F	$\lvert\Delta T\rvert$, °F	T_{max}, °F	$\lvert\Delta T\rvert$, °F
Fuel cladding	644.63	644.97	0.34	643.05	1.58
Guide sleeves	643.95	644.3	0.35	642.46	1.46
Spacer disk	632.4	632.83	0.43	630.93	1.47
Support rod	451.39	452.64	1.25	452.44	1.05

The results above show that as the solid element numbers are increased, the temperature differences are reduced. Since none of the component temperatures changed by more than 1.58°F, the original SAR finite element model is not mesh sensitive.

Question 4.21

Provide the radial (nodal) DSC maximum temperature distribution for the bounding case.

The SAR states that storage blocked vent conditions are bounding for all other events, but an ANSYS radial (nodal) distribution was not provided on the reported results. This information is needed to assure compliance with 10 CFR 72.11 and 72.236.

Response – Question 4.21

The radial (nodal) DSC maximum temperature distribution for the blocked vent transient case is given in Section N.4.6.2 as Figure N.4-17.

Question 4.22

Provide all ANSYS analysis models in .db or .inp format.

When submitting any ANSYS analysis model, verify that all the necessary files needed for running, or if necessary, modifying the ANSYS calculation are included. Include also a summarized description of the ANSYS calculation options (either for batch or interactive calculations) used and the computer platform utilized for the performed calculations. This information is needed to assure compliance with 10 CFR 72.11 and 72.236.

Response – Question 4.22

The attached CDs include proprietary ANSYS files of bounding Heat Load Zoning Configuration 1- .txt (batch input), .db, .rth and .sub for the following cases:

1. For 70°F ambient HSM storage case, use load step 3 from CD#1
2. For 100°F ambient transfer case, use load step 3 from CD#2
3. For blocked vent accident transient analysis case, use CD#3
4. For vacuum drying transient analysis case, use CD#4

A Pentium III 600 workstation is utilized to perform these ANSYS analyses.

Question 4.23

Modify all references by including the amendment letter in front of them (e.g., Reference 4.1 should be changed to N.4.1, etc.) consistent with the SAR nomenclature. The current reference numbering is not clear (e.g., it appears that the reference numbering corresponds to the original standardized NUHOMS® Final Safety Analysis Report). This information is needed to assure compliance with 10 CFR 72.11.

Response – Question 4.23

A note is added to Section N.4.1 to clarify that all references are given in Section N.4.8.

Question 4.24

Modify the SAR to clearly present a logic sequence for Sections N.4.5.1 and N.4.5.2.

The information presented in these Sections of the SAR should follow a logic sequence in terms of model description, assumptions, applied boundary conditions and results.

This information is needed to assure compliance with 10 CFR 72.11.

Response – Question 4.24

Sections N.4.5.1 and N.4.5.2 are revised as requested.

Question 4.25

Clarify the last paragraph of Section N.4.6.2. It currently states that the maximum temperatures of the basket assembly after 34 hours are listed in Table N.4-2; however, these temperatures are listed in Table N.4-1.

This information is needed to assure compliance with 10 CFR 72.11.

Response – Question 4.25

The temperature reported in Table N.4.6-2 is for transfer cask accident steady state case. These temperatures are bounded by the blocked vent accident case temperature results reported in Table N.4.6-1. The last sentence of Section N.4.6.2 is revised to add this clarification.

References to Chapter 4 RAI Responses

[1] "Fuel Rod Analysis for Dry Storage of Spent Nuclear fuel," Report DPC-NE-2014P, Duke Energy, August 2001, TAC L23369.

[2] Letter from K. S. Canady (Duke Energy) to L. R. Wharton (NRC-SFPO), Responses to Request for Additional Information by NRC Staff, May 24th, 2002, TAC-L23369.

[3] Calculation, Alternate Thermal Analysis of the NUH-24PHB DSC, Transnuclear Calculation Number NUH-HBU.0403, Revision 0.

[4] "Topical Report for the NUTECH Horizontal Modular Storage System for Irradiated Nuclear Fuel, NUHOMS-24P," NUH-002, Revision 1A.

[5] "Safety Evaluation Report Related to the Topical Report for the NUTECH Horizontal Modular Storage System for Irradiated Nuclear Fuel NUHOMS-24P Submitted by Nutech Engineers, Inc.," U.S. Nuclear Regulatory Commission, ONMSS, April 1989.

Chapter 5 Shielding

Question 5.1

Justify the acceptability of fuel burned to greater than 45 GWd/MTU, considering the uncertainties with source term determinations associated with high burnup fuel. No benchmark data exists for SAS2H, to provide adequate validation of the isotopic depletion calculations for fuel burned above 45 GWd/MTU for PWR assemblies. This information is needed to assure compliance with 10 CFR 72.104, 72.106, and 72.236.

Response – Question 5.1

The shielding analysis for this amendment application is based on a hybrid source term consisting of the predicted neutron source from an assembly with 55 GWd/MTU burnup, an initial enrichment of 3.4 wt. % U-235 and a cooling time of eight years. The gamma source is from an assembly with 46 GWd/MTU burnup, an initial enrichment of 3.2 wt. % U-235 and a cooling time of 5.5 years and an additional gamma source term from design basis BPRAs. The bounding dose rates on the surface of the Transfer Cask (TC), HSM and bounding occupational exposures are from Heat Load Zoning Configuration 2 (twenty - 1.3 kW assemblies per DSC). Fuel with a burnup of 55 GWd/MTU is chosen because it represents the largest neutron source for any fuel allowed by the Fuel Qualification Table (FQT) to be stored in the canister. Fuel with a burnup of 46 GWd/MTU is chosen because it represents a very large gamma source due to the relatively short cooling time of the fuel.

Therefore, the direct gamma component of the dose rate is based on a source term that is bounded by the available measured data for PWR fuel as documented in Reference [2]. Evaluations of the existing data with SAS2H and the 44-group ENDF/B-V library used in our analysis are documented in References [3] and [4]. These comparisons all show generally good agreement between the calculations and measurements, and show no trend as a function of burnup in the data that would suggest that the isotopic predictions, and therefore neutron and gamma source terms, would not be in good agreement. A similar conclusion is also reached by the results documented in JAERI report [5]. In fact, for the case with 46,460 MWd/MTU burnup, the isotopic predictions are all within 2% of those measured. There are ongoing efforts, some of which are documented in Reference [2], to obtain more data for burnups above 45 GWd/MTU. There is no reason to expect that the ongoing evaluations of the higher burnup fuel will result in less favorable comparisons.

As noted in References [1] and [2], there is no public data for the neutron component currently available that bounds a fuel burnup of up to 55 GWd/MTU. However, as documented in Reference [1] and confirmed in our SAS2H analysis, the total neutron source with increasing burnup is more and more dominated by spontaneous fission neutrons. Reviewing the output from our SAS2H runs, the neutron source term is due almost entirely to the spontaneous fission of Cm-244 (~98% of all neutrons both spontaneous fission and (α,n)). After reviewing the measured Cm-244 content compared to the Cm-244 content predicted by SAS2H and the 44-group ENDF/B-V library documented in References [3] and [4] for burnups up to 46,460 MWd/MTU, it is readily apparent that the calculated values are within ±11 % of the measured values, with most of the predicted values within ±5% of the measured. Finally, there is no observed trend as a function of burnup in the data that would indicate that the predicted Cm-244 content is significantly different at higher burnups.

As documented in Reference [1] and as observed in preparing the FQT, the gamma dose rate increases nearly linearly with burnup relative to the direct gamma component and the neutron dose rate increases with burnup to the fourth power. Therefore, as burnups go beyond 45 GWd/MTU, the contribution from neutron (and associated n,γ) components to the total dose rates measured on the surfaces of the DSC, TC and HSM increase in relative importance to that of the gamma component. However for the NUHOMS® HSM, this increase in the importance of the neutron source term has a relatively minor effect on the area dose rates on and around the HSM as these are dominated by the gamma component. The surface dose rates on the HSM are dominated by the gamma component because the HSM is constructed of thick reinforced concrete, which is an excellent neutron shield. The ratio of the average neutron to the average gamma dose rate on the surfaces of the HSM is from 0.12 to 0.02 (See Table N.5-4). Therefore, even a postulated substantial increase in the neutron source term would have a relatively minor effect on the site dose rate evaluation presented in Section N.10 of the amendment application.

For the TC, the neutron source term has a relatively minor effect on the area dose rates during most of the cask handling operations as the DSC cavity and the annulus between the TC and DSC is filled with water and most of the work is done around the top of the cask. The neutron component is of more importance on and around the TC during transfer operations but, in general, only represents slightly less than half the total dose rate on the sides and top of the TC. While the dose rate on the bottom of the TC is predominately from the neutron source, relatively little occupational dose is received from this area. The dose rates for the design basis fuel on the surfaces of HSM and TC are shown in Table N.5-3 and Table N.5-4. These tables show that gamma dose rates are substantially higher than neutron dose rates. Therefore, the neutron component of the dose is a relatively minor fraction of the total occupational and site boundary dose.

The occupational exposure calculations demonstrate that most of the dose received by workers during cask loading and transfer operations is due to the gammas on and around the cask. The only surface of the TC that is dominated by neutrons is at the bottom of the cask. Less than 5% of the total occupational exposure is due to the doses around the bottom of the cask because very little work is performed on or around the bottom of the cask with fuel in the TC.

The dose rates around the loaded HSM are measured per the proposed Technical Specification 1.2.7a to ensure that the fuel loaded is in compliance with the license. If the measured dose rates are greater than those predicted, the site would have to perform an evaluation to demonstrate that they still fall within the 10 CFR 72.104 evaluations. If not, further evaluation would be performed to demonstrate that the health and safety of the public and workers (ALARA) is maintained.

As discussed above, any impact of uncertainties in source terms is expected to be negligible for the NUHOMS® system. Therefore, isotopic depletion calculations with SAS2H for fuel burned above 45 GWd/MTU are appropriate.

Question 5.2

Clearly define the meaning of "low enriched UO_2 rods," as reference on page N.5-1. Describe how a low enriched rod differs from the B&W 15X15 spent fuel rod and its impact on the shielding analysis. Additionally, explain the rationale for why fuel assemblies were "reconstituted" with low enriched rods, rather than stainless steel rods.

This information is needed to assure compliance with 10 CFR 72.104, 72.106, and 72.236.

Response – Question 5.2

For in-reactor operation, fuel assemblies with damaged or leaking fuel rods can be reconstituted in order to replace damaged rods. A typical replacement is a fuel rod that contains pellets of naturally enriched UO_2 or "low enriched UO_2. This type of rod is used because it is similar in design and behavior to the standard fuel rod and is analyzed using standard approved methods for in-reactor operation or storage. It avoids the negative reactivity associated with stainless steel replacement rods and produces a small amount of power. Bullet 2 on FSAR page N.5-1 has been revised accordingly to clarify the meaning of "low enriched UO_2 rods

If grid damage exists, solid filler rods made of stainless steel or Zircaloy could be used as a replacement. A solid filler rod is used because a low enriched UO_2 replacement rod is more susceptible to a through wall defect caused by the grid damage. A maximum of 10 such filler rods can be substituted into a single fuel assembly. Fuel that is acceptable for continued in-reactor operation will not affect the cladding analysis in Duke Topical Report DPC-NE-2014P, which demonstrates that the cladding will not degrade during storage.

Question 5.3

Provide a detailed description, including examples of output files, for the ANISN evaluation conducted for reconstituted fuel described on page N.5-5.

Response – Question 5.3

Fuel reconstituted with stainless steel rods differs from non-reconstituted fuel rods in two ways with respect to determining a representative source term and for performing the associated shielding evaluation. First, the total metric tons of heavy metal (Uranium) is slightly lower, as fuel rods that contained uranium are replaced with stainless steel rods which displace the same amount of water as the original fuel rods, but contain no fuel. Therefore, a B&W 15x15 fuel assembly that originally contains 208-fueled rods with 2.356 kgU each, has an initial loading of 0.49 MTU. If this assembly is reconstituted with ten (10) stainless steel rods, then the total uranium loading of the reconstituted assembly is reduced to (208-10)*2.356kgU or 0.466 MTU. The second difference is that there is more stainless steel in the assembly that is irradiated as a result of the inclusion of the stainless steel rods into the assembly.

These changes affect the source term calculation performed with SAS2H in the following ways. First the "basis" uranium content is reduced from 0.49 MTU to 0.466 MTU. This results in a reduced neutron and decay heat source term for a given burnup because there is less uranium in the assembly. The second effect is on the gamma source term. One might expect that the gamma source would also be decreased due to the reduced uranium content in the assembly. However, the presence of the additional stainless steel rods, which become irradiated (Co-59), results in a substantial increase in the gamma dose rate, masking the effect of the reduced uranium content, especially at shorter cooling times. SAS2H runs are performed to calculate neutron, gamma and decay heat source terms as a function of burnup, initial enrichment and cooling time for fuel reconstituted with ten stainless steel rods. The shielding evaluation uses the results of these runs to determine the minimum cooling time required for the design basis source terms.

The effect of the differences between assemblies reconstituted with stainless steel rods and non-reconstituted assemblies on the shielding evaluation is discussed below. First, as discussed above, the source terms are different. Second the fueled region of the canister has slightly different material densities because some of the zirconium clad UO_2 fuel pellets are replaced with solid stainless steel rods in each assembly. This difference in the smeared region has a negligible effect on the calculated surface dose rate owing to the shielding capabilities of the thick reinforced concrete walls in the HSM and the heavy steel and lead walls of the TC. Therefore, as stated in Section N.5.2, the reconstituted fuel source terms are evaluated "...using the ANISN models that are used to develop the fuel qualification tables."

The ANISN models referred to in Section N.5.2 are those used to develop the response function described in Section N.5.2.4. Using this response function, the relative strength of the gamma source term due to the reconstituted fuel can be evaluated on the surface of the HSM roof and on the side of the TC. As discussed in the Section N.5.2.4, the response function explicitly accounts for differences in the gamma source spectrum and the magnitude of the source. Therefore it is a direct comparison of all of the source terms against the design basis source term. Therefore, the only difference between fuel reconstituted with stainless steel rods and non-reconstituted fuel in the shielding models that is not explicitly accounted for in this shielding evaluation is the slight difference in the material densities in the fueled region of the canister. As discussed above, this difference is negligible compared to the concrete roof in the ANISN model of the HSM and the heavy steel and lead walls in the ANISN model of the TC. In addition, the source term evaluation conservatively assumes that the stainless steel rods are irradiated during all three cycles of operation, which increases the activation source term from the stainless steel in the reconstituted rod by slightly less than a third, more than accounting for the minor difference in the material densities.

A copy of two example ANISN input and output files used to generate the response function described in Section N.5.2.4 is included on the attached CD. The first case

(HNEUT.ai/ao) determines the neutron and (n,γ) response function for the HSM and the second case (CG23.ai/ao) determines the group 23 gamma response function for the TC.

Question 5.4

Provide an explanation of "assembly average enrichment." Use the explanation to clarify the meaning of the sentence in the third paragraph of page N.5-5, "For reconstituted fuel with lower enriched uranium oxide rods, the assembly average enrichment produced the same total assembly decay heat, neutron and gamma source."

This information is needed to assure compliance with 10 CFR 72.104, 72.106, and 72.236.

Response – Question 5.4

The definition of "assembly average enrichment" is very similar to that used for BWR fuel. A typical BWR fuel assembly contains radial and axial variations in fuel pellet enrichments. In a single fuel rod, fuel pellets of various enrichments are stacked on top of each other in order to control the axial flux profile and resultant burnup profile in the core. Likewise, in the lattice at any give axial position in the assembly the fuel pellets have variable enrichment. The peak pellet enrichment, the maximum lattice average enrichment, and the assembly average enrichment are typically listed to describe the enrichment of a BWR fuel assembly. The peak pellet enrichment is the maximum enrichment found in any single pellet in the assembly before irradiation. The maximum lattice average enrichment is the maximum calculated area average planer enrichment found over the height of the assembly prior to irradiation, and the average assembly enrichment is the average uranium enrichment of the assembly before irradiation (i.e., total grams of U-235 divided by the total grams of uranium). For fuel reconstituted with lower enriched uranium rods, it is conservative to use the assembly average enrichment as defined above for BWR fuel, i.e., the total grams of U-235 in the reconstituted assembly divided by the total grams of uranium in the assembly had it been configured in its reconstituted form prior to irradiation.

For BWR fuel, the assembly average enrichment, as defined above, is used to generate the neutron, gamma and decay heat source terms. In the same way, for fuel reconstituted with lower enriched uranium oxide, the assembly average enrichment can be used to generate the neutron, gamma and decay heat source terms. The reconstituted fuel assembly contains the same heavy metal content as an identical non-reconstituted assembly and is "identical" in all other ways related to source term and shielding properties when compared with non-reconstituted fuel assembly. Therefore, the source terms that SAS2H generates for a non-reconstituted B&W 15x15 fuel assembly with 208 fuel rods with a uniform 4.0 wt. % enrichment are identical to a reconstituted fuel assembly with 15 natural uranium rods (0.72 wt. % U-235) and the remaining rods with an initial enrichment of 4.25 wt. % U-235. The assembly average enrichment for the reconstituted fuel assembly is also 4.0 wt. % U-235, therefore the resulting source term calculated by SAS2H is identical to that of the non-reconstituted fuel assembly with an initial enrichment of 4.0 wt. % U-235.

Page N.5-5 of the FSAR is revised to provide an explanation of assembly average enrichment.

Question 5.5

Provide an explanation with the supporting calculation packages that served as the bases for constructing Table N.5-12. Include in the explanation whether Configuration 1 or 2 was used in determining the volume data and how the empty cells are addressed.

Table N.5-12 as presented introduces doubt regarding how Configuration 2, the assumed bounding condition, is applied in the shielding analysis. This information is needed to assure compliance with 10 CFR 72.104, 72.106 and 72.236.

Response – Question 5.5

The Heat Load Zoning Configuration 1 is the basis for zone volumes shown in Table N.5-12. This table is revised to include the zone volumes for Heat Load Zoning Configuration 2.

The volumes of the zones for Configuration 1 are calculated as follows. Zone 1 encompasses the center four assemblies as shown in Figure N.2-1 of the amendment. The equivalent cross sectional area of this four assembly region is calculated such that the cross sectional area of the four fuel assembly compartments is conserved. The cross section of a fuel assembly compartment is 8.9 inches square. The cross sectional area is therefore $4*(8.9 \text{ inches})^2 = 316.84 \text{ in}^2$ or $2{,}044 \text{ cm}^2$. This forms an equivalent radius of 25.52 cm. The lengths of the various assembly regions are given in Table N.5-5 of the amendment and are reproduced below.

Bottom Nozzle	–	location of FA bottom nozzle and fuel rod end plugs (8.38 in)
In-core	–	location of active fuel (142.29 in)
Plenum	–	location of fuel rod plenum springs and top plugs (8.73 in)
Top nozzle	–	location of top nozzle (extends 6.23 in. above the plenum)

The volumes of the assembly regions in Zone 1 are therefore the product of the cross sectional area of Zone 1 and the length of the assembly region.

Zone 2 encompasses the middle ring of twelve assemblies as shown in Figure N.2-1 of the amendment. The equivalent cross sectional area of this twelve assembly ring is calculated such that the cross sectional area of the twelve fuel assembly compartments is conserved. The cross sectional area of Zone 2 is therefore $12*(8.9 \text{ inches})^2 = 950.52 \text{ in}^2$ or $6{,}132 \text{ cm}^2$. This forms an equivalent annular region with an inner radius of 25.52 cm and an outer radius of 51.02 cm. The volumes of the assembly regions in Zone 2 are therefore the product of the cross sectional area of Zone 2 and the length of the assembly region.

The radius of Zone 3 is calculated by conserving the total area occupied and enclosed by the 24 fuel assemblies, including the guide sleeves, and guide sleeve wrappers, in the loaded DSC. The distance to the outer edge of each outer cutout in the spacer disc is taken from the drawings and the area of this region is calculated. The resultant cross sectional area is 2,523.64 in^2 or 16,281 cm^2. The radius of the equivalent cylinder is 71.99 cm. Therefore the cross sectional area attributed to Zone 3 is $\pi(71.99^2-51.02^2) =$ 8,104 cm^2. The volumes of the assembly regions in Zone 3 are therefore the product of the cross sectional area of Zone 3 and the length of the assembly region.

The volumes of the Zone 3 for Configuration 2 are simply the sum of the volumes of Zone 2 and 3 from Configuration 1.

The outer ring of assemblies (Zone 3 for both configurations) control the dose rates on the surfaces of the HSM and the TC. Therefore, one would expect that Configuration 2 would result in the controlling shielding configuration because it allows 20 of the "hottest" fuel assemblies, and thus the strongest neutron and gamma source terms, to be configured around the edges of the DSC. The ANISN and DORT calculations performed in support of this application demonstrate the same. For models of Configuration 2, the center region (radius=25.51 cm) is modeled as void. Note that the radii given above are modeled exactly as stated in the DORT models, and are rounded to the nearest cm in the ANISN models.

Question 5.6

Clarify how the empty cells in Configuration 2 are homogenized (smeared) to simplify shielding calculations.

Smearing source regions may be non-conservative in situations where empty spaces are smeared in with other material densities. This information is needed to assure compliance with 10 CFR 72.104, 72.106 and 72.236.

Response – Question 5.6

Only Configuration 2 contains four empty cells and these empty cells are all located in the center of the DSC. As discussed in the response to RAI Question 5.5, the center region is modeled as void for models of Configuration 2. This is slightly conservative, as we do not take credit for some of the self-shielding that would be provided by the guide sleeves in the four empty cells.

Question 5.7

Correct page N.5-38 to show the publishing date for Reference no. 5.11 as 1977.

This information is needed to assure compliance with 10 CFR 72.236

Response – Question 5.7

Reference 5.11 is corrected to show 1977 as the publishing date.

References to RAI Responses for Chapter 5

[1] U.S. Nuclear Regulatory Commission, "Nuclide Importance to Criticality Safety, Decay Heating, and Source Terms Related to Transport and Interim Storage of High Burnup LWR Fuel," NUREG/CR-6700, Published January 2001, ORNL/TM-2000/284.

[2] U.S. Nuclear Regulatory Commission, "Review of Technical Issues Related to Predicting Isotopic Compositions and Source Terms for High Burnup LWR Fuel," NUREG/CR-6701, Published January 2001, ORNL/TM-2000/277

[3] "An extension of the Validation of SCALE (SAS2H) Isotopic Predictions for PWR Spent Fuel," ORNL/TM-13317, MD DeHart and OW Hermann, September 1996.

[4] "Validation of the SCALE System for PWR Spent Fuel Isotopic Composition Analyses," ORNL/TM-12667, OW Hermann, SM Bowman, MC Brady, CV Parks, March 1995.

[5] "Technical Development on Burn-up Credit for Spent LWR Fuels," JAERI-Tech 2000-071, Japan Atomic Energy Research Institute, September 21, 2000.

Chapter 7 Confinement

Question 7.1

Submit the helium leak test procedure described in Section M.8.1.4 "DSC Sealing Conditions," Step 5 (page M.8-7) that will be used to verify technical specification 1.2.4a.

Include step by step descriptions of the methodologies used, and applicable drawings which show how appropriate confinement welds will be tested and when. Also include a description of the sensitivity obtainable of each qualified method and the required testing sensitivity level to demonstrate compliance with technical specification 1.2.4a. Additionally, address affects such as ambient helium, testing location, and any other applicable affects.

This information is needed to assure compliance with 10 CFR 72.190, 72.192 and 72.236.

Response – Question 7.1

The proper section number and page number for the helium leak test procedure is Section N.8.1.4 "24PHB DSC sealing Operations," (page N.8-2).

Listed below are generic steps, which provide additional instructions for performing helium leak testing to meet Technical Specification (T.S.) 1.2.4.a requirements. Detailed procedural steps are to be developed by the general licensee to satisfy site-specific considerations.

Step 1: Leak Testing the Pressure Boundary (Refer to Figure N.3-1).

(a) Following the completion of the DSC drying operations, the DSC is backfilled with helium to about 24 psia as described in Section 5.1.1.3 in Step 23, and leak tested in accordance with ANSI N.14.5 to a sensitivity of 1×10^{-5} atm cm^3/sec.

(b) The DSC cavity is then evacuated to T.S. 1.2.2 limits and repressurized to about 17.2 psia with helium in accordance with T.S. 1.2.3a limits.

Step 1 assures integrity of all field installed pressure boundary welds and assures that helium is behind them except the siphon and vent port cover plate welds shown in Figure N.3-1.

Step 2: Injecting helium in the Siphon and Vent Port Volume:

Step 1 of Section N.8.1.4 states that helium is to be injected into the blind space prior to seal welding the prefabricated plugs over the vent and siphon ports. The "blind space" is the volume of the siphon and vent port cavities (See DETAIL 1 of SAR drawing NUH-HBU-1000 for a detail of the vent port cavity). This is accomplished as follows:

(a) Dry the siphon and vent ports.

(b) Run the helium in the supply line long enough to purge the line of air.

(c) Insert the helium needle into the vent port as far as possible, and place the cover plate over the port. The needle rests on the vent/siphon block. One side of the cover sits in its recess, and the other side rests on the needle.

(d) Run the helium purge at a rate slow enough not to lift the cover or to draw in air by turbulence or venturi effect. Purge long enough to turn over the port volume (approximately 4 cubic inches) by at least 50 times, based on the measured flowrate. For example, 2 minutes at 100 cubic inches helium/minute (0.06 CFM, 3.5 CFH).

(e) Withdraw the needle, letting the vent cover drop into place.

(f) Tack two opposite sides of the vent port cover.

(g) After tacking, complete root pass for the vent port cover plate.

(h) Complete weld per normal procedure and perform a dye penetrant test for the vent ports per T. S. 1.2.5 limits.

(i) Repeat steps 2c) through 2h) above for the siphon port cavity.

Step 2 assures helium in the vent and siphon port cavities. Based on the mock up tests performed for the 61BT DSC that demonstrate 95% helium content in the cavities, Transnuclear, Inc. conservatively recommends assuming 50% helium and correspondingly reducing the maximum allowable helium leak measurement from 1E-7 to 5E-8 atm.cc/s (ANSI N14.5, example B15).

Step 3: Testing the DSC Field Installed Closure Weld Pressure Boundary to Leak Tight Criteria:

Step 3 may be performed using bell jar or a seal cap or by using the test port or other appropriate methods. The use of the test port provided in the outer top cover plate (OTCP) for performing the testing is described herein:

(a) Place the OTCP to the DSC shell and complete the OTCP root pass weld.

(b) With helium environment behind all the field installed pressure boundary welds assured by Steps 1 and 2 above, hook up a mass spectrometer to the OTCP test port (Refer to Figure N.3-1) provided in the outer top cover plate. The spectrometer sensitivity must be consistent with the maximum limit discussed in step 2 above. Verify that the DSC pressure boundary meets the leak tight criteria of T.S. 1.2.4.a.

(c) Continue loading operations per step 5 of Section N.8.1.4.

Question 7.2

Clarify the standards and criteria that the leak test personnel will be qualified too. Provide information which shows that personnel performing leak tests have the qualifications necessary (such as SNT-TC-1A) to perform such tests.

This information is needed to assure compliance with 10 CFR 72.190, 72.192, and 72.236.

Response – Question 7.2

The leak test personnel are qualified in accordance with SNT-TC-1A. A new sentence is added to Section N.8.1.4 step 4 as follows: "Verify that the personnel performing the leak test are qualified in accordance with SNT-TC-1A." A similar sentence is also added to the first paragraph of Section N.9.1.3.

Chapter 11 Accident Analysis

Question 11.1

Include an analysis, which analyzes the probability and effects of an inadvertently loaded assembly. Based on recent events, the staff no longer accepts that inadvertent loading of an assembly with a heat generation rate greater than design basis is not credible based on quality assurance programs, particularly when complicated loading schemes exist as in this application.

The staff does accept that analyses do not have to be performed for "misloadings," if the probability and consequences of such an event can be shown to be not significant to safety from a thermal, containment, shielding, and criticality perspective.

Update Section M.11.2 as applicable. This information is required to verify compliance with 10 CFR 72.94 and 72.236(I).

Response – Question 11.1

The correct Section is N.11.2

The risk of misloading of a fuel assembly or BPRA is minimized based on the following additional administrative controls:

1. Additional requirements are added for utility to prepare loading maps prior to DSC loading for fuel assemblies including BPRAs, if applicable, to be loaded in a given canister based on the CoC Technical Specifications.

2. This loading map is required to be independently verified before any fuel load.

3. Additional independent verification is required that the loading map has been followed correctly and accurately after the fuel load, but before the top shield plug is placed.

These additional checks and verification steps are added to Chapter N.8 to assure that the double contingency criteria is applied for misloading of an assembly or BPRA and therefore, the risk of inadvertently misloading of an assembly is negligibly small.