TRANSNUCLEAR, INC.

NUHOMS®-24PHB SYSTEM

PRELIMINARY SAFETY EVALUATION REPORT

AMENDMENT NO. 6

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PRELIMINARY SAFETY EVALUATION REPORT

Docket No. 72-1004
NUHOMS®-24PHB SYSTEM
Certificate of Compliance No. 1004
Amendment No. 6

SUMMARY

By application dated August 31, 2001, as supplemented on June 13, 2002, November 18, 2002, January 10, 2003, and March 7, 2003, Transnuclear, Inc. (TN) requested approval of an amendment, under the provisions of 10 CFR Part 72, Subpart K, to Certificate of Compliance No. 1004 for the Standardized NUHOMS® System.

TN requested a change to the Certificate of Compliance (CoC), including its attachments, and revision of the Final Safety Analysis Report (FSAR). The requested change was to add a modified storage system designated as the NUHOMS®-24PHB System to accommodate high burnup spent fuel assemblies as authorized contents to the Standardized NUHOMS® System.

The NRC staff has reviewed the application using the guidance provided in NUREG-1536, "Standard Review Plan for Dry Cask Storage Systems," January 1997 (SRP). Only those SRP chapters with a corresponding applicant request for revision or changes are addressed in the NRC staff's safety evaluation report. The applicant added a transfer cask, OS-197H, and a horizontal storage module, HSM-102, to be used in the NUHOMS®-24PHB System, as authorized under the provisions of 10 CFR 72.48. The staff did not review any changes associated with the OS-197H or HSM-102 changes, with the exception of the shielding function provided by the HSM-102 design. Based on the statements and representations in the application, as supplemented, the staff concludes that the TN Standardized NUHOMS® system, as amended, meets the requirements of 10 CFR Part 72. The proposed Amendment No. 6 changes to the CoC are indicated by double change bars in the margins. The pending changes for proposed Amendment No. 5 to the TN Standardized NUHOMS® System are indicated by single change bars in the margins.

1.0 GENERAL DESCRIPTION

The TN NUHOMS®-24PHB System consists of the following components: (1) 24 PHB Dry Shielded Canister (DSC), which contains the spent fuel; (2) Horizontal Storage Module (HSM) Model 102, which contains the DSC during storage; and (3) Standard, OS197 or OS197H Transfer Cask (TC), which contains the DSC during loading, unloading and transfer operations. Each DSC stores up to 24 B&W 15x15 spent fuel assemblies, with or without burnable poison rod assemblies (BPRAs), with an average burnup of up to 55,000 MWd/MTU (high burnup fuel). There are two DSC configurations: the 24 PHBS and 24PHBL DSC. The 24 PHBS and 24 PHBL DSC shell and basket assembly designs are essentially the same as the NUHOMS®-24 standard length and long cavity shell assemblies, respectively, except that the 24PHB DSC outer top cover plate has a test port/plug to allow testing of the canister shell for leak tightness.

1.1 Updated FSAR Revision 6

The applicant updated Section 7.2.3 of the FSAR to document the methodology used to determine fuel qualification tables for the NUHOMS® 24P and 52B canisters. The qualification tables are presented in Tables 3.1-8a and 3.1-8b of the FSAR. The NRC has approved these fuel qualification tables, as specified in Tables 1-2a and 1-2b of the Technical Specifications issued in Amendment No. 2 to the CoC, dated August 30, 2000. The applicant did not request any changes to these fuel qualification tables in the Safety Analysis Report (SAR) amendment. The staff reviewed the updated Section 7.2.3. The updated information and associated qualification tables appear to be consistent with information previously submitted in support of Amendment No. 2, “NUH004.0510, Fuel Assembly Qualification,” Revision 4, dated February 10, 1999. Therefore, this update is adequate to reflect the methodology used to generate the fuel qualification tables in Tables 3.1-8a and 3.1-8b of the FSAR.

2.0 PRINCIPAL DESIGN CRITERIA

The objective of evaluating the principal design criteria related to the system, structures, and components (SSC) important to safety is to ensure that they comply with the relevant general criteria established in 10 CFR Part 72 (Ref. 1).

2.1 Structures, Systems, and Components Important to Safety

The SSCs important to safety are described in SAR Section N.2.3.

2.2 Design Basis for SSCs Important to Safety

The NUHOMS®-24PHB DSC design criteria summary includes the range of spent fuel types and configurations to be stored, and design criteria for environmental conditions and natural phenomena.

2.2.1 Spent Fuel Specifications

The applicant revised the allowable contents of the NUHOMS®-24PHB DSC to include intact B&W fuel assemblies meeting the parameters specified in Table 1.1i of Technical Specification 1.2.1, "Fuel Specifications." The specification includes a maximum burnup of 55 GWd/MTU clad in Zircaloy (either Zircaloy-2 or Zircaloy-4) and B&W 15x15 BPRA designs as listed in Appendix J of the FSAR. A detailed description of the allowable fuel and storage configurations is provided in Tables N.2-1 through N.2-5 in the SAR.

2.2.2 External Conditions

Section N.2.2 of the SAR identifies the bounding site environmental conditions and natural phenomena for which the NUHOMS®-24PHB DSC is analyzed. In cases where these did not change, no descriptions were given. External conditions are further evaluated in Sections 3 through 12 of this SER.

2.3 Design Criteria for Safety Protection Systems

A summary of the design criteria for the safety protection systems of the NUHOMS®-24PHB DSC, is presented in Section N.2.3 of the SAR. Details of the design are provided in Sections N.3 though N.11 of the SAR.

The applicant has designed the NUHOMS®-24PHB DSC to provide storage of spent fuel for 40 years. The Standardized NUHOMS® System has been licensed by the NRC staff for 20 years of storage. The fuel cladding integrity is assured by the NUHOMS®-24PHB DSC and basket design which limits fuel cladding temperatures and maintains a nonoxidizing environment in the cask cavity. The NUHOMS®-24PHB DSC is designed to maintain a subcritical configuration during loading, handling, storage, and accident conditions. A combination of soluble boron in the pool, and favorable geometry are employed. The NUHOMS®-24PHB DSC shell, closure, and basket are designed and fabricated in accordance with ASME Boiler and Pressure Vessel Code (BP&V) (Ref. 2), Section III, Subsections NB, NC, NF, NG, and Appendices. There are no changes to the Code exceptions as noted in SAR Section N.3.1.2.2.

2.4 Evaluation Findings

F2.1 The staff concludes that the principal design criteria for the NUHOMS®-24PHB DSC are acceptable with regard to meeting the regulatory requirements of 10 CFR Part 72. This finding is reached on the basis of a review that considered the regulation itself, appropriate regulatory guides, applicable codes and standards, Interim Staff Guidance (ISG), and accepted engineering practices. A more detailed evaluation of design criteria and an assessment of the compliance with those criteria is presented in Sections 3 through 12 of the SER.

2.5 References

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

2. American Society of Mechanical Engineers, ASME Boiler and Pressure Vessel Code, Section III, Subsection NB, NC, NF, NG and Appendices, 1983 Edition with Winter 1985 Addenda.

3.0 STRUCTURAL EVALUATION

This section presents the structural design review of the amendment request for the addition of the NUHOMS®-24PHB DSC to the CoC. The structural review of the amendment was conducted against the appropriate 10 CFR Part 72 regulations (Ref. 1), industry standards and codes to ensure safe nuclear fuel storage. The review was conducted to assess the safety analysis of structural components, the structural criteria, and the methodology to evaluate structural capabilities for the structures, systems, and components (SSCs) important to safety under normal, off-normal and accident conditions. Review procedures were consistent with the Standard Review Plan for Dry Cask Storage Systems, NUREG-1536 (SRP) (Ref. 2). The NUHOMS®-24PHB system allows storage of high burnup fuels, and adds a test port and plug to the top cover plate to allow for leak tight testing in the DSC. The applicant stated that no design or configuration changes were required for the HSM or TC to accomodate the NUHOMS®-24PHB System. The NUHOMS®-24PHB is basically consistent with the NUHOMS®-24P DSC design, which has been previously reviewed and approved. Hence during the review process, applicable structural calculations and analyses for the 24P DSC components have been adopted for qualifying the components in the 24PHB system.

3.1 NUHOMS®-24PHB DSC Design Criteria

The NUHOMS®-24PHB DSC shell assembly is designed and fabricated based on the ASME Boiler and Pressure Vessel Code, Section III, Subsection NB, Article NB-3200 (Ref. 3). The DSC basket assembly is designed and fabricated based on the ASME Boiler and Pressure Vessel Code, Section III, Subsections NB and NF. Exceptions to the Code are cross referenced in SAR Section N.3.1.2.2. Sealing of the NUHOMS®-24PHB DSC involves leak testing based on the ANSI N14.5 (Ref. 4) criteria.

The NUHOMS® -24PHB system is designed to withstand the effects of accident conditions such as earthquakes, tornadoes, lightning, and floods, etc.

3.1.1 Stress Calculations

The NUHOMS®-24PHB system is subjected to the same design loads as the NUHOMS®-24P or -52B system such as, internal pressure, tornado wind and missiles, seismic loads, snow and ice loads, and dropped loads. A summary of 24PHB DSC load combinations is presented in SAR Table N.2-6. The load combinations reflect the various operational conditions and events based on which the critical stresses in various components may be computed. The critical stresses are then used in the component design. The applicant utilized a combination of finite element models using the ANSYS computer program and hand calculations for calculating the stresses for DSC shell assembly and DSC basket assembly components.

3.1.2 Allowable Stresses

Allowable stresses for the NUHOMS®-24PHB DSC shell and assembly components are based on the ASME Section III, Division I service levels (levels A and B for normal and off-normal loads, and levels C and D for accident loads, etc.). Allowable stresses also vary with stress type, such as primary membrane and membrane plus bending, and with the temperatures.

Allowable stresses provide measurement for determining the design safety margins of the structural components.

3.2 Structural Analysis (Normal and Off-Normal Conditions)

3.2.1 DSC Shell Assembly Analysis

The NUHOMS® -24PHB DSC shell assemblies are evaluated for normal and off-normal pressures using procedures similar to the 24P DSC shell assemblies. The internal pressure calculated for the 24PHB DSC are bounded by its internal design pressures. The heat load of 24kW for the 24PHB DSC designs are identical to the 24P standard and long cavity designs. The weights of the 24PHBS and 24PHBL DSC shell assemblies are the same as those of the 24P standard and long cavity DSCs, respectively. The weight of the fuel assemblies in the 24PHBS and 24PHBL DSCs are bounded by those in the 24P DSCs. Therefore, the stress due to vertical and horizontal dead weights in the shell and basket assembly components of the 24PHBS and 24PHBL DSCs are bounded by those of the 24P DSCs. SAR Table N.3.6-1 summarizes the maximum NUHOMS®-24PHB DSC component stresses for normal and off-normal loads and are acceptable.

3.2.2 DSC Basket Assembly Structural Analysis

Stress Analysis of the Spacer Discs

The spacer disc dead weight stress calculations for the vertical and horizontal orientation inside the TC and the HSM are the same as described in the 24P DSC. In addition, 3-D finite element models were developed using the ANSYS program for spacer disc in-plane and out-of-plane loading analyses. The enveloping load combination results for the spacer disc for normal and off-normal loads are shown in Table N.3.7-2, and are all within allowable stresses.

Stress Analysis of the Guide Sleeves

The maximum stresses in the 24PHB guide sleeves due to dead load and handling loads are the same as those for the 24P guide sleeves because guide sleeves and the materials used in the guide sleeve designs in the 24PHB and 24P are identical. However, the maximum temperatures of the 24PHB guide sleeves are higher than the 24P guide sleeves, and therefore, 24PHB guide sleeves have lower allowable stresses. Still, the maximum computed guide sleeve stress is less than the allowable.

Stress Analysis of the Support Rods

The maximum stresses in the 24PHB support rods due to normal loads such as dead load and handling loads are the same as those for the 24P support rods because support rods and the materials used in the support rod designs in the 24PHB and 24P are identical. However, the maximum temperatures of the 24PHB support rods are higher than those of the 24Ps, and therefore, the 24PHB support rods have lower allowable stresses. The maximum computed support rod stress is still lower than the allowable, and therefore acceptable.

3.3 Structural Analysis (Accidents)

Several accident conditions are discussed below.

3.3.1 Earthquake

Similiar to the NUHOMS®-24P, the design basis seismic analyses of the NUHOMS®-24PHB system components are subject to a peak horizontal ground acceleration of 0.25g and a peak vertical ground acceleration of 0.17g, based on RG 1.60 (Ref. 5) guidance. A three percent damping value is used for the DSC seismic analysis, and a seven percent damping value for DSC support steel and concrete, is used for the HSM, based on RG 1.61 (Ref. 6) guidance.

DSC Seismic Stress Analysis:

The seismic design loads for the DSC shell are 3.0g horizontal and 1.0g vertical. The basis for these seismic loads are the assumption that the DSC is only supported by one of the HSM support rails, and the effects of combined modes of excitation. The DSC shell stresses are obtained using the absolute sum of the two directional excitations. The final seismic DSC stresses are bounded by the previous 24P DSC results.

Basket Seismic Analysis:

The seismic analysis results for the 24PHB DSC basket components such as spacer discs, guide sleeves and support rods, etc., are the same as those for the 24P DSC.

Furthermore, the seismic analysis results for the 24PHB DSC and 24P DSC are the same for the following items: HSM seismic evaluation, DSC support structure seismic evaluation, flooding analysis, and TC seismic evaluation.

3.3.2 Cask Drop Accident

<u>DSC Shell Assembly</u>
For the 75g side drop, the DSC shell assembly stress results are bounded by the stresses in the 24P DSC shell assembly.

<u>Basket Assembly</u>
The stress results obtained from the ANSYS analyses for spacer discs, support rods and guide sleeves for drop accidents (75g side drop and 60g vertical drop) are shown in Table N. 3.7-1. All stresses as reported in note 6 are bounded by those for the 24P DSC basket assembly components and are acceptable.

Additional bifurcation buckling analysis (ANSYS) of the spacer disc was performed to evaluate the stability aspect of the spacer disc. The staff requested demonstration that the gaps between the 24PHB DSC shell and the spacer discs do not close under the combination of off-normal ambient temperature condition (100°F) and a side drop accident. The applicant provided calculations which show that under the postulated load combination, the gaps remain at 0.255 inches.

3.3.3 DSC Load Combination

Among all loadings discussed above, the postulated cask drop accidents are the most critical. Hence, the DSC shell and basket assembly components are evaluated for stress intensity using the controlling combinations which include the cask accident loads. Table N.3.7-2 through Table N.3.7-4 tabulate the calculated maximum stress intensity for each component of the NUHOMS®-24PHB DSC shell and basket assembly under the enveloping normal, off-normal, and accident load combinations. The tables also provide ASME Code allowable stresses for the components under review. For comparison purposes, the following maximum component ratios of calculated versus allowable are presented as follows: 0.946 for the inner bottom cover plate, ASME Service Levels A and B; 0.879 for the DSC shell, ASME Service Level C; and 0.996 for the outer bottom cover plate, ASME Service Level D. All the maximum component stresses are within allowable stresses and are acceptable.

All other load combination evaluation for TC fatigue, HSM and DSC support structure are the same as those performed in 24P DSC.

3.4 Evaluation Findings

The staff concludes that the structural design of the NUHOMS®-24PHB System is in compliance with 10 CFR Part 72 and that the applicable design and acceptance criteria have been satisfied. The structural evaluation provides reasonable assurance that the NUHOMS®-24PHB System will enable safe storage of spent nuclear fuel. This finding is based on staff review that considered SRP, appropriate regulatory guides, applicable industry codes and standards, and satisfactory response to and resolution of request for additional information (RAI) issues.

3.5 References

1. 10 CFR Part 72, Subparts K and L, revised as of January 1, 2000.

2. NUREG-1536, “Standard Review Plan For Cask Storage Systems,” January 1997.

3. American Society of Mechanical Engineers, ASME Boiler and Pressure Vessel Code, Section III, Subsection NB, NC, NF, NG and Appendices, 1983 Edition with Winter 1985 Addenda.

4. American National Standard, "Radioactive Materials - Leakage Tests on Packages for Shipment," ANSI N14.5-1997.

5. Regulatory Guide 1.60, “Design Response Spectra For Seismic Design of Nuclear Power Plants,” December 1973.

6. Regulatory Guide 1.61, “Damping Values For Seismic Design of Nuclear Power Plants,” October 1973.

4.0 THERMAL EVALUATION

The staff reviewed the NUHOMS®-24PHB Dry Shielded Canister (DSC) thermal design and performed independent confirmatory calculations to assess that the cask and fuel material temperatures are within their allowable values or criteria for normal, off-normal, and accident conditions, as required in 10 CFR Part 72 (Ref. 1). The staff's review of the applicant's submittal and independent analyses confirmed that the temperatures of the fuel cladding (fission product barrier) will be maintained below the acceptable limits during DSC loading, draining, drying, and inerting, as well as throughout the planned storage period. The staff reviewed the application in accordance with guidance provided in Section 4 of NUREG-1536, "Standard Review Plan for Dry Cask Storage Systems," (Ref. 2) as well as associated interim staff guidance documents.

4.1 Spent Fuel Cladding

The predicted fuel cladding temperatures were assessed to be below the expected damage thresholds, as proposed by the applicant, in Tables N.4-1, N.4-2, N.4-3, and N.4-4 for the two acceptable heat load configurations. The two configurations are shown in Figures N.4-5 and N.4-6 of the SAR. Tables N.4-1, N.4-2, N.4-3, and N.4-4 of the SAR present the maximum temperature criteria for fuel cladding long term storage, short term normal conditions, off-normal events, and accident conditions, respectively. The temperature limits for spent fuel cladding are based on the Spent Fuel Project Office (SFPO) Interim Staff Guidance 11, Revision 2 (ISG-11) (Ref 3.). ISG-11 establishes that a maximum fuel cladding temperature limit of 752°F (400°C) is applicable to both normal conditions of storage and all short term operations. In addition, per ISG repeated thermal cycling of the cladding during fuel loading operations should be minimized and temperature differences greater than 65°C should not be permitted. The maximum fuel cladding temperature limit of 1058°F (570°C) is applicable only for accidents or off-normal thermal transient conditions. For the NUHOMS®-24PHB DSC unloading operations, the maximum fuel cladding temperature during cask reflood is postulated to be significantly less than the vacuum drying condition due to the presence of water and steam during the process. Consequently, a lower temperature rise is expected.

4.2 Cask System Thermal Design

The NUHOMS®-24PHB DSC is designed to store 24 intact standard PWR fuel assemblies with or without BPRAs, (supported by the thermal analysis presented in Appendix J of the NUHOMS® FSAR) with assembly average burnup, initial enrichment, and cooling time as described in Table N.2-1 of the SAR. The DSC is evacuated and backfilled with helium at the time of loading. The DSC is designed to passively reject decay heat during storage and transfer for normal, off-normal, and accident conditions while maintaining component temperatures and pressures within the limits specified by the applicant in the SAR. TS 1.2.17b establishes the limit for vacuum drying following the completion of DSC draining. The objective of the above time limit is to ensure that the NUHOMS®-24PHB DSC fuel cladding temperature does not exceed 752°F (400°C). Monitoring of the time duration for vacuum drying operation is required by the applicant's technical specifications.

4.3 Thermal Load Specifications

The maximum total decay heat load per DSC is 24 kW, with a maximum per assembly heat load of 1.3 kW when zoning (preferential loading) is used to distribute the heat load in a nonuniform manner. The loading configurations, based on the decay heat, are incorporated into the technical specification for the NUHOMS®-24PHB CoC amendment as Figures 1-8 and 1-9. The maximum DSC cavity pressures during normal, off-normal and accident conditions must be below the design basis pressure of 15, 20, and 68 psig, respectively. Maximum DSC cavity pressures from the applicant's pressure analysis for normal, off-normal, and accident conditions are presented in Sections 4.5, 4.6, and 4.7 of this Safety Evaluation Report (SER).

4.4 Model Specifications

Data for the effective thermal conductivity of the fuel region (with Helium backfill) are obtained from Figure 4.5-2 of the report "Spent Fuel Dry Storage Testing at E-MAD," (March 1978 - March 1982), PNL-4533, September 1982. This data was generated from above ground vertical concrete silo tests for a single fuel assembly with a decay heat of 1.05 kW. The fuel effective thermal conductivity values shown in Figure 4.5-2 of the E-MAD test report take into consideration a combination of conduction, radiation, and convection heat transfer. The applicant uses a combination of experimental results from the E-MAD testing and an analytic expression to develop temperature dependent effective thermal conductivity values for vacuum conditions. The analytic expression relates the temperature difference between the fuel center and the inside of the guide sleeve wall, to the fuel assembly thermal and material characteristics by considering the fuel assembly to be a finite heat generating slab. The fuel effective thermal conductivity values used by the applicant are validated against test data from thermal testing of the NUHOMS®-07P (Ref. 4) horizontal storage cask. The NUHOMS®-07P and NUHOMS®-24PHB DSCs have the same basic layout (i.e., spacer disks and guide sleeves are used to support and maintain a separation between the spent fuel assemblies). The NUHOMS®-07P testing examined the thermal performance of a NUHOMS®-07P DSC using a combination of electrical heaters and actual spent fuel assemblies.

Results from the testing with actual spent fuel assemblies were selected by the applicant to validate the fuel effective thermal conductivity values. However, the total decay heat of the NUHOMS®-07P canister is less than 6.0 kW while the NUHOMS®-24PHB DSC total decay heat is 24 kW. Nonetheless, the most important characteristic to match is the peak cladding temperature and the temperature drop through the canister. The difference between the peak clad temperature and the DSC surface temperature for the NUHOMS®-24PHB DSC is on the order of 300°F (148.8°C) higher than that observed for the NUHOMS®-07P measured values. As such, the utilization of the test data to validate the thermal models is limited.

The axial conductivity of the fuel region is derived using a parallel resistance method with no credit for the presence of the fuel pellet, guide sleeves or control components for both Helium and vacuum environments in the DSC cavity. The applicant assumed that vacuum drying of the DSC does not reduce the pressure sufficiently to reduce the thermal conductivity of water, vapor and air in the DSC cavity. Therefore, for vacuum drying operations, the applicant obtained axial effective thermal conductivity values using the parallel resistance method mentioned above assuming that an air environment exists in the DSC cavity. Studies have shown that in a low pressure (4 torr or greater) nitrogen/air environment, modeling conduction through the gas is appropriate.

Material property tables for the DSC shell and basket components are included in Tables 8.1-8 and 8.1-9 of the FSAR. The temperature range for the material properties covers the range of temperatures encountered during the thermal analysis of the NUHOMS®-24PHB DSC.

Thermal analyses were performed for normal conditions involving ambient temperatures of:

- maximum normal temperature of 100°F (37.7°C) with insolation
- a minimum normal temperature of 0°F (-17.7°C) without insolation
- an average ambient temperature of 70°F (21.1°C) was used for long term storage

Previous analysis of the HSM and TC, (Ref. 5) furnished the surface temperature of the DSC shell. The maximum calculated DSC temperatures are applied to the DSC surface as boundary conditions for the detailed DSC thermal model. Off-normal conditions for storage that were analyzed included:

- an ambient temperature of 117°F (47.2°C) with insolation
- 125°F (51.6°C) ambient with solar shield in place with insolation for the transfer case

Additional analyses for both storage and transfer were performed using a minimum ambient temperature of -40°F (-40°C) without insolation. As before, previous analyses of the HSM and TC provided the DSC surface temperatures. A fire analysis was also addressed. Steady state, off-normal conditions were assumed prior to the fire accident analysis. A fire temperature of 1475°F (801.6°C), with emittance of 0.9, and a duration of 15 minutes, (based on a full consumption of a 300 gallon diesel fuel source) with complete engulfment of the TC for the duration of the fire accident was assumed.

The applicant modeled the thermal performance of the NUHOMS®-24PHB DSC using the ANSYS® finite element analysis code. For normal storage conditions the applicant developed a 3D model of a 21.1 inch central axial section of the cask between the axial centers of two spacer disks, and applying adiabatic boundary conditions to both ends of the model. To assess the thermal performance of the system under vacuum drying and blocked vent transient conditions, a model representing a 180° azimuthal section of the DSC model was constructed based on the 3D model described above. Both models simulated the effective thermal properties of the fuel assemblies as a homogenized material (with calculated effective thermal conductivity values) occupying the fuel assembly volumes within the basket. Gaps between adjacent components in the model were included to accurately simulate the heat transfer between these components. Heat transfer across these gaps included both gaseous conduction and thermal radiation.

The applicant performed a mesh sensitivity study for their finite element analysis of the normal storage case. The mesh sensitivity study consisted of reducing the number of elements in the model to 96% and 68%, respectively, with respect to the original thermal model described in the SAR. The applicant's results showed that by increasing the element number from 68% to 100% (the number of elements used to perform the thermal analysis in the SAR), the temperature differences are reduced. According to the applicant's results, none of the component temperatures changed by more than 1.58°F (0.87°C) (for both the 96% and the 68% mesh cases) in comparison to the original SAR results. Therefore, the applicant concluded that the SAR thermal finite element model results are not mesh sensitive. The applicant did not perform a calculation by refining (increasing the number of elements) the element mesh used in the SAR analytical models. At the staff's request, the applicant also performed a mesh sensitivity study in the fuel assembly region. This study showed that an increase in the fuel assembly mesh size from 4x4 elements to 12x12 elements, resulted in an increase in the maximum fuel cladding temperature by approximately 0.2°F (0.1°C). Therefore, the applicant concluded that the thermal results contained in the SAR are not sensitive to a fuel mesh size smaller than 4x4 elements.

4.5 Evaluation of Cask Performance for Normal Conditions

The maximum fuel cladding temperature for short and long term storage is evaluated by the applicant for each of the two decay heat load zoning configurations (configurations shown in Figures N.4-5 and N.4-6). The results obtained are compared with the corresponding fuel cladding temperature limit for short and long term storage in Tables N.4-1 and N.4-3 of the SAR. According to the results presented in these tables, the bounding case corresponds to Loading Configuration 1 which is shown in Figure N.4-5. For this case a margin of 107°F (41.6°C) against the allowable limit of 752°F (400°C) (as established in ISG-11) was calculated by the applicant for the maximum fuel cladding temperature.

The predicted fuel cladding temperatures for transfer conditions are given in Table N.4-2 and N.4-4 of the SAR. The maximum fuel cladding temperatures are below the allowable short term limit of 752°F (400°C) by 30°F (16.6°C) for Loading Configuration 1 (the loading case calculated by the applicant to be the bounding case). Under the minimum temperature condition (0°F [-17.7°C]), the SSCs important to safety continue to perform their function. The maximum calculated pressure for normal conditions corresponds to 6.3 psig, which is well below the DSC normal condition design pressure of 15 psig.

4.6 Evaluation of Cask Performance for Off-Normal Conditions

Maximum calculated temperatures for off-normal storage and transfer are given in Table N.4-1, N.4-2, N.4-3, and N.4-4 of the SAR. According to these tables, the maximum calculated temperature of 722°F (383.3°C) was obtained for the transfer case of Loading Configuration 1. Normal and off-normal conditions of transfer for Loading Configuration 1 are the bounding cases, with a maximum fuel cladding temperature of 722°F (383.3°C) for both conditions. For off-normal conditions the maximum fuel cladding temperatures are below the allowable fuel cladding temperature limit of 752°F (400°C). The maximum calculated pressure for off-normal conditions corresponds to 11.2 psig, which is below the DSC off-normal condition design pressure of 20 psig.

4.7 Evaluation of Cask Performance for Accident Conditions

The maximum calculated temperature for the blocked vent event after 40 hours is 762°F (405.5°C), which is below the limit of 1058°F (570°C). The maximum calculated pressure for accident conditions corresponds to 63.1 psig, which is below the DSC design pressure limit of

68 psig. The calculated maximum fire transient DSC surface temperature is 499°F (259.4°C), which is less than the blocked vent case maximum DSC temperature of 574°F(301.1°C). Therefore, the NUHOMS®-24PHB DSC temperatures and pressures calculated for the blocked vent case bound the hypothetical fire accident case.

4.8 Evaluation of Cask Performance for Loading/Unloading Conditions

The maximum cladding temperature reached during vacuum drying after approximately 31 hours is 722°F (383.3°C) which is below the maximum limit of 752°F (400°C) per ISG-11. The maximum temperature difference for the fuel cladding during drying and backfilling is 102°F (56.6°C), which meets the thermal cycling criteria specified by ISG-11 (no greater than 117°F [65°C]). The steady state maximum fuel cladding temperature during transfer to the ISFSI pad is 722°F (383.3°C). The maximum temperature difference for the fuel cladding during the DSC backfilling and transfer operations is 102°F (56.6°C), which also meets the thermal cycling criteria of ISG-11. The maximum fuel cladding temperature during cask reflood operations will be significantly less than the vacuum drying condition because of the presence of water and/or steam in the DSC cavity.

4.9 Staff's Confirmatory Analysis of the NUHOMS®-24PHB DSC

A confirmatory analysis of the NUHOMS®-24PHB DSC thermal design, using the COBRA-SFS finite volume thermal code, was performed by the staff as an independent evaluation of the thermal analysis presented in the applicant's SAR. The staff's three-dimensional thermal model explicitly modeled the fuel assemblies, including individual rods, inside the DSC. It also included an explicit representation of the DSC shell, spacer disks, guide sleeves, and Helium cover gas. COBRA-SFS thermal calculations were performed for normal condition of storage and small differences in peak cladding temperatures between the applicant's results and the staff's confirmatory analysis were found. The staff's analysis showed a peak fuel cladding temperature that was 13°F (7.2°C) higher than the applicant's predicted maximum fuel region temperature. Therefore, based on a review and evaluation of the amendment request and the staff's confirmatory thermal analysis, the staff concluded that the applicant's thermal design of the NUHOMS®-24PHB is acceptable because it meets the thermal design criteria specified for this design.

4.10 Evaluation Findings

F4.1 Appendix N, Section N.2 of the SAR describes SSCs important to safety to enable an evaluation of their thermal effectiveness. Cask SSCs important to safety remain within their operating temperature ranges.

F4.2 The NUHOMS®-24PHB DSC is designed with a heat-removal capability having verifiability and reliability consistent with its importance to safety. The cask is designed to provide adequate heat removal capacity without active cooling systems.

F4.3 The spent fuel cladding is protected against degradation leading to gross ruptures under accident conditions by maintaining cladding temperatures below 1058°F (570°C). Protection of the cladding against degradation is expected to allow ready retrieval of spent fuel for further processing or disposal.

F4.4 The applicant's ANSYS® models and the staff's confirmatory analysis have indicated that differences in temperature values for normal conditions for storage are small and on the level of acceptable range. Based on the applicant's results and staff's confirmatory analysis results, the staff's find the design acceptable.

F4.5 The staff concludes that the thermal design of the NUHOMS®-24PHB DSC is in compliance with 10 CFR Part 72.

4.11 References

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

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

3. U.S. Nuclear Regulatory Commission, Interim Staff Guidance No. 11, Revision 2, "Cladding Considerations for the Transportation and Storage of Spent Fuel," July 30, 2002.

4. "NUHOMS® Modular Spent Fuel Storage System: Performance Testing," EPRI-NP 6941/PNL-7327, September 1990.

5. Transnuclear West, Final Safety Analysis Report of the Standardized NUHOMS® Horizontal Storage System for Irradiated Nuclear Fuel, October 2001, Revision 6.

5.0 SHIELDING EVALUATION

The staff evaluated the capability of the NUHOMS® 24PHB canister to provide adequate protection against direct radiation from the canister contents when used with the Standardized NUHOMS® System. The regulatory requirements for providing adequate radiation protection to licensee personnel and members of the public include 10 CFR Part 20 and 10 CFR 72.104, 72.106(b), 72.212, and 72.236(d). Since 10 CFR Part 72 dose requirements for members of the public include direct radiation, effluent releases, and radiation from other uranium fuel-cycle operations, an overall assessment of compliance with these regulatory limits is evaluated in Section 10 of this SER.

The applicant performed two primary sets of shielding analyses in Chapter N.5 of the SAR. In the first set of analyses, the applicant performed forward shielding calculations with the DORT and MCNP shielding codes, using design-basis source terms generated by SAS2H/ORIGEN-S. In the second set of analyses, the applicant used a dose response function methodology (i.e., source strength to dose rate conversion factors) to determine alternate burnup, cooling time, and enrichment parameters for the spent nuclear fuel (SNF). The fuel parameters calculated with the dose response methodology result in dose rates and individual heat loads that are less than, or equal to, the maximum dose and thermal limits derived from the bounding calculations in the first set of analyses. The applicant used the ANISN shielding code to calculate dose response functions and SAS2H/ORIGEN-S to generate alternate source terms. Each subsection in this SER section addresses both sets of analyses, as appropriate.

5.1 Shielding Design Features and Design Criteria

5.1.1 Shielding Design Features

The applicant stated the physical design of the 24PHB canister is identical to the 24P canister, except for the addition of a test port to the top cover plate. The 24PHB includes the 24PHBS design variation with a standard cavity and a 24PHBL design variation with a long cavity for BPRAs. The 24PHB canister specification is limited to changes in burnup, cooling time, and minimum enrichment specifications. It also allows for high-burnup fuel up to 55 GWD/MTU.

The applicant did not identify any physical changes to the horizontal storage module (HSM) or transfer cask (TC) in the SAR amendment, as currently described in Revision 6 of the updated Standardized NUHOMS® System FSAR. The NUHOMS® storage module used with the 24PHB DSC consists of the Horizontal Storage Module 102 (HSM-102). The HSM-102 is a modified version of the originally approved HSM-80. The NUHOMS® transfer casks used with the 24PHB consist of the standard, OS-197 and OS-197H. The OS-197H is a modified version of the originally approved OS-197. The applicant added the HSM-102 design to the NUHOMS® FSAR under the change authority requirements of 10 CFR 72.48. FSAR Section 1.3.1.2 of Revision 6 of the updated FSAR stated that the HSM-102 is similar to the HSM-80, except for a new door configuration that is two feet thick reinforced concrete with a inside steel liner and the addition of one and a half inch steel plate liners to the inlet and outlet vents. The applicant used the HSM-102 storage configuration in the shielding analysis for the 24PHB canister.

The staff reviewed the new HSM-102 design configuration with respect to its shielding performance during normal conditions of operation for the new 24PHB contents specified in

Appendix N of the SAR amendment. The staff did not review any other design features of the HSM-102 or TC, except for the integrated shielding function of the entire system to provide adequate protection against direct radiation from the new 24PHB content specifications. The staff notes that a review was neither requested nor performed for any new structural, material, or thermal design changes specifically related to the new HSM-102 storage design in this SAR amendment.

The staff evaluated the NUHOMS®-24PHB DSC shielding design features and found them acceptable. The shielding design features of the 24 PHB canister are essentially the same as the shielding design features of the approved 24P canister. The shielding design features of the HSM-102 provide the same function as the design features of the HSM-80. According to Revision 6 of the FSAR, the HSM-102 provides better shielding for the 24P. Based on this information, the applicant's shielding design provides reasonable assurance that the shielding design features of 24PHB/HSM-102 system can meet the radiological requirements of 10 CFR Part 20 and 10 CFR Part 72. The shielding analysis in this SAR amendment does not address the originally approved shielding configuration of the HSM-80. Therefore, the 24PHB canister specification was not evaluated or approved for the HSM-80 shielding configuration design.

5.1.2 Shielding and Source Term Design Criteria

The overall radiological protection design criteria are the regulatory dose requirements in 10 CFR Part 20 and 10 CFR 72.104, 72.106(b), 72.212, and 72.236(d). The applicant analyzed the NUHOMS®-24PHB with spent fuel as described in Section N.5. The SAR amendment also specified methodological dose limit criteria used to determine the acceptable burnup, cooling time, and enrichment parameters. These include maximum dose limits of 1370.2 mrem/hr on the radial surface of the TC and 93.7 mrem/hr on the roof surface of the HSM-102. These dose rates were calculated with the ANISN response function methodology in the SAR amendment for Configuration 2 (see Section 5.1.3 of SER). These ANISN dose rates correlate to DORT dose rates (forward calculations) of 1026 mrem/hr and 36 mrem/hr calculated at the same location with design-basis source terms (without BPRA contributions). In addition, the heat load limits for each fuel category (see Section 5.1.3 of SER) serve as additional source term design criteria, which in turn limits overall dose rates. Based on these design criteria, the applicant calculated bounding dose rates on the outside of the door, the front vents, and on the end shield wall exterior.

The staff reviewed the design criteria and found it to be acceptable. The shielding design criteria defined in the SAR provides reasonable assurance that the 24PHB/HSM-102 system can meet the radiological requirements of 10 CFR Part 20 and 10 CFR Part 72. Dose rate limits based on the bounding shielding analysis are incorporated into Technical Specification (TS) 1.2.7b for the outside of the HSM-102 door, the front surface, and end shield wall exterior of the HSM-102.

5.1.3 Preferential Loading Criteria

The 24PHB canister will be used to store up to 24 PWR fuel assemblies in two different preferential loading patterns, specified as Configuration 1 and Configuration 2. These configurations are depicted in SAR Figures N.2-1 and N.2-2. Configuration 1 consists of three fuel categories identified as Zone 1, Zone 2, and Zone 3 category fuel, that are preferentially arranged in all 24 cell locations. (These three fuel categories are referred to in this SER as Category 1 fuel, Category 2 fuel, and Category 3 fuel.) Configuration 2 consists of only the Zone 3 fuel category, preferentially arranged in the 20 outer-cell locations. The four cells located in the center are empty.   Each fuel category has specific values for several combinations of maximum (assembly average) burnup, minimum cooling time, and minimum (assembly average) initial enrichment. These values are specified in Tables N.2-3, N.2-4, and N.2-5. Parametes for Zone 1 and Zone 2 fuel (used only in Configuration 1) are derived from decay heat limits of 0.7 kW and 1.0 kW, respectively. Parameters for Zone 3 fuel (used in Configuration 1 or 2) are based on the Configuration 2 loading, and the most limiting parameter of either a 1.3 kW heat load, a maximum ANISN TC surface dose rate of 1370.2 mrem/hr, or a maximum ANISN HSM-102 surface dose limit of 93.7 mrem/hr. The ANISN dose rates for each combination of parameters was determined by multiplying the gamma source terms (by energy group) and neutron source strengths by the associated HSM-102 and TC response functions determined with ANISN. This method is demonstrated in Table N.5-24 of the SAR amendment (see section 5.3 of the SER) .

The applicant provided a comparison in Table N.5-21 that indicated Configuration 2 loaded with design basis Category 3 fuel results in bounding dose rates over Configuration 1 loaded with design basis Category 1, 2, and 3 fuel (see Section 5.2 of SER). To satisfy thermal requirements, the applicant also limited the total heat load in each configuration to be 24 kW, which is less than the total heat load that would result from a loading of Category 1, 2, and 3 fuel at the individual design-basis heat loads (i.e., 0.7, 1.0, and 1.3 kW, respectively). Therefore, the cask user will have to calculate heat loads for each individual fuel assembly prior to loading to verify the total canister heat load less than or equal to 24kW.

The staff reviewed the use of preferential loading and specifications for the fuel categories. The preferential loading schemes and fuel categories provide reasonable assurance that the 24PHB/HSM-102 system can meet the regulatory requirements of 10 CFR Part 20, 10 CFR 72.104(a), and 10 CFR 72.106(b). Based on calculations provided by the applicant, the staff agrees that Configuration 2 results in bounding dose rates over Configuration 1. The staff notes that Category 1 and 2 fuel parameters are not specifically restricted by calculated dose limits, and changes to decay heat source term are not always directly proportional to changes to radiation source terms. However, the staff has reasonable assurance that the respective heat load limits for Category 1 and 2 fuel for Configuration 1 will result in lower doses than Configuration 2. This is based on the margin in dose rates listed in Table N.5-21 and margin between the Category 3 heat load (1.3 kW) and Category 2 heat load (1.0 kW). In addition, dose rate limits based on Configuration 2 are also incorporated into Technical Specification 1.2.7b. The preferential loading schemes and decay heat limits specified in Figure N.2-1 and N.2-2, and the maximum burnup, minimum cooling time, and minimum enrichment limits specified in Tables N.2-3 through N.2-5 are incorporated into Technical Specification 1.2.1.

5.2 Source Specification

The applicant calculated design-basis source terms to perform bounding shielding calculations with DORT and MCNP. The design-basis source specification for bounding calculations are presented in Section N.5.2 of the SAR amendment. The gamma and neutron source term calculations were performed with the SAS2H/ORIGEN-S modules of the SCALE 4.4 computer code, using the 44-group ENDF/B-V cross-section library. The design basis fuel type is the

B&W 15x15 assembly as described in Table N.5-1 and N.5-5 (previously approved for the 24P canister). The design-basis source terms used a heavy metal weight of 0.49 MTU to produce the highest source strength at a given burnup, cooling time and enrichment combination.

Source terms were calculated for the design basis assembly for the three fuel categories (defined in the SAR as Zone 1, Zone 2, and Zone 3). The design-basis burnup, cooling time, and enrichment combinations are listed on Page N.5-2 of the SAR. The design basis thermal source term of 8 watts for the authorized BPRAs are taken from the previously approved Appendix J, and are added to the total thermal source term. The applicant did not include BPRA radiological source terms for the forward shielding calculations. Instead the applicant, used a ratio methodology based on Appendix J to adjust calculated dose rates for BPRA loadings. For the design-basis DORT source terms and the ANISN dose response function method, the applicant used ORIGEN-S to restructure the gamma source term into the CASK-81 energy group structure. The applicant provided sample SAS-2H and ORIGEN-S computer input files in Section N.5.5.1 of the SAR amendment.

The applicant also calculated multiple source terms for alternate burnup, cooling time, and enrichment combinations. The gamma and neutron source term calculations were performed with the SAS2H/ORIGEN-S and are similar to the calculations for the design-basis source terms. The applicant verified each combination including BPRA contributions for all fuel categories were within their respective heat limits. In addition, the applicant used these source terms with the ANISN dose response functions to qualify each combination for Category 3 fuel.

The applicant also provided a method for determining the enrichment of fuel reconstituted with low enriched uranium oxides rods. The method is based on dividing the total grams of U-235 by the total grams of uranium assuming the SNF is reconstituted prior to irradiation. The applicant also evaluated fuel reconstituted with up to 10 steel rods. Based on examination of gamma dose rates with ANISN, the applicant determined that a minimum of 9 years cooling time is needed for the fuel reconstituted with stainless steel rods. Therefore, reconstituted fuel with steel rods must be cooled for at least 9 years for all burnup and enrichment combinations in Tables N.2-3 through N.2-5.

5.2.1 Gamma Source

The design-basis gamma source terms for Zone 1, 2, and 3 fuel categories are listed in Tables N.5-9 through N.5-11. The applicant applied neutron flux scaling factors listed in Table N.5-7 and fuel hardware parameters listed in Table N.5-5 to determine activation source terms for the top nozzle, plenum, in-core, and bottom nozzle regions. The assumed cobalt impurities and elemental compositions are listed in Table N.5-6. The applicant applied axial gamma peaking factors based on an axial burnup profile to determine the gamma source terms in the various axial source regions.

5.2.2 Neutron Source

The design-basis neutron source terms for the Zone 1, 2, and 3 fuel categories are listed in Table N.5-13. The applicant applied axial neutron peaking factors based on an axial burnup profile raised to the power of 4.0 to determine the neutron source terms in various axial regions.

The applicant provided and referenced the uncertainties in the source term analyses in Section N.5.2.1. The applicant stated that the predictions of Cm-244 densities, which is the predominant neutron source, are accurate within 20 percent. The applicant further addressed source term uncertainty in its RAI response dated June 13, 2002, to support the SAR amendment (Revision 1 of the application). The applicant stated that calculations and measurements of the gamma source terms show good agreement. The applicant indicated that there are no expectation, or trends, that indicate the uncertainties in source terms at high burnup ranges would increase compared to those at normal burnup ranges. The applicant stated that the neutron source is a minor contributor to HSM dose rates and is relatively minor fraction of occupational dose (less than 5%). Therefore, the applicant concluded that dose impacts associated with these uncertainties in high burnup fuel are negligible for the 24PHB/HSM-102 system.

5.2.3 Staff Evaluation

The staff reviewed the source term analyses in Chapter N.5 of the SAR. The staff has reasonable assurance that the design basis gamma and neutron source terms for the NUHOMS®-24PHB Configuration 1 and 2 analyses are acceptable for the shielding analysis. The staff performed confirmatory calculations of the source terms for the specified fuel types, burnup conditions, and cooling times. The staff used the SAS2H and ORIGEN-S computer codes. The calculated source terms were in general agreement with the applicant's bounding source terms used in the DORT analysis. The staff also calculated source terms for selected fuel parameters listed in Tables N.2-3 through N.2-5. The calculated thermal source terms were in agreement with the fuel category decay heat limits. The staff agrees that the applicant has adequately addressed fuel reconstituted with low enrichment uranium oxide rods and fuel reconstituted with up to 10 steel rods. The staff also agrees that the applicant has reasonably addressed source term uncertainties in high burnup fuel with respect to the specific contents and specific design proposed in this SAR amendment. Although the exact uncertainty in high burnup source terms remain unclear, the total peak dose rates are limited in TS 1.2.7b and the neutron source term is a minor dose contributor.

The preferential loading schemes; maximum initial uranium loading of 0.490 MTU; decay heat limits specified in Figure N.2-1 and N.2-2; and the maximum burnup, minimum cooling time, and minimum enrichment limits specified in Tables N.2-3 through N.2-5 are incorporated into Technical Specification 1.2.1. Restrictions and methods to address reconstituted fuel are also incorporated into Technical Specification 1.2.1. Peak dose rate limits based on Configuration 2 are also incorporated into Technical Specification 1.2.7b.

5.3 Shielding Model Specifications

The NUHOMS®-24PHB system shielding and source configuration is described in Sections N.5.3 and N.5.4 of the SAR amendment. The applicant indicated that the shielding methodology with DORT are consistent with basic modeling techniques previously approved for the NUHOMS® 61BT storage system. The applicant used DORT, a 2-D discrete ordinates code, to determine bounding dose rates on and around the HSM-102 and TC. The applicant used two-dimensional X-Z models for HSM-102 dose rates and 2-D cylindrical models for TC dose rates with the CASK-81 cross-section library. The shielding models consist of two dimensional representations of the HSM-102 and TC, including the spent nuclear fuel source and 24PHB canister. These models are depicted in Figures N.5-1 through N.5-3 and Figure N.5-5 of the SAR amendment. The applicant provided a sample DORT input file in Section N.5.5.2 of the SAR amendment.

The applicant also stated in Section 5.4.5 of the SAR amendment that the 2-D DORT methodology results in conservative dose rates with respect to actual measured data. The applicant referenced information submitted to NRC dated February 21, 2002. The applicant stated that this information shows that the 2-D DORT methodology bounds the 3-D MCNP methodology, which was in turn validated by actual measurements from installed NUHOMS® systems.

The applicant used MCNP, a three-dimensional Monte Carlo transport code, to determine dose rates at the HSM-102 vent areas. The shielding models consist of three dimensional representations of the HSM-102 and vent streaming paths, including the spent nuclear fuel source and 24PHB canister. These models are depicted in Figures N.5-4A and N.5-4B of the SAR. As discussed in Section N.10.4 of the SAR amendment, the applicant used MCNP to calculate off-site dose rates at large distances from two generic ISFSI array. The generic arrays included a 2x10 back-to-back array of HSM-102s and two 1x10 front-to-front arrays (35-foot separation) loaded with design basis fuel in Configuration 2 of the 24PHB (see also Section 10.4 of SER). The applicant provided a sample MCNP input file in Section N.5.5.3 of the SAR amendment.

The applicant used ANISN, a one-dimensional discrete ordinates code, to determine dose response functions for Configuration 2 in the HSM-102 and TC. The applicant indicated this shielding methodology was similar to the methodology previously approved, as described in updated Section 7.2.3 of the SAR amendment (see Chapter 1 of the SER). The shielding models consist of one-dimensional representations of the HSM-102 and TC, including the spent nuclear fuel source and 24PHB canister regions. These models are depicted in Figures N.5-19 and N.5-20 of the SAR amendment. The response functions were generated with the CASK-81 22 neutron, 18 gamma couple cross-section library. The gamma (by energy group) and neutron dose response functions for both the HSM-102 and TC are listed in Table N.5-15 of the SAR amendment. The applicant also used ANISN to determine the energy spectrum of radiation emitted from the HSM-102, as input to the MCNP array calculations discussed above. The applicant provided a sample ANISN input file in Section N.5.5.4 of the SAR amendment.

5.3.1 Shielding and Source Configuration

The shielding source is divided into four axial regions: bottom nozzle, in-core, plenum, and top nozzle. The lengths of these regions are specified in Table N.5-5. The source regions were homogenized and the cross-sectional area was preserved. As described in Section N.5.2.1 of the SAR amendment, the applicant determined equivalent radii and volumes for each preferential loading configuration, shielding, and void region, as appropriate for the cylindrical DORT and ANISN shielding models.

5.3.2 Material Properties

The composition and material densities used in the DORT models are specified in Tables N.5-16 through N.5-18. The composition and material densities used in the ANISN models are specified in Table N.5-25. The applicant indicated in Section N.5.3 of the SAR amendment that the material densities (with exception to the canister basket and fuel) were consistent with the material densities used in the shielding calculations previously approved.

5.3.3 Staff Evaluation

The staff evaluated the shielding models and found them acceptable. The material compositions and densities used were appropriate and provide reasonable assurance that the 24PHB DSC was adequately modeled. In addition, the methodologies used are similar to those used to support previous NUHOMS® storage applications, including the 61BT application. A supplemental 3-D analysis (which was benchmarked by measurements of actual installed NUHOMS® systems) provides further assurance that the DORT model is conservative in predicting dose rates.

The staff notes that use of the ANISN 1-D model to represent the 3-D 24PHB shielding system results in additional uncertainties. However, the use of ANISN in the shielding analysis is essentially limited to evaluating the relative changes in dose rates versus relative changes in source terms for alternate combinations of burnup, cooling time, and enrichment. The staff finds the use of ANISN acceptable for this specific design and contents for the following reasons: (1) higher energy gamma source terms dominate public dose rates and any ANISN-related uncertainties should be relatively systematic for each fuel combination; (2) the use of ANISN is consistent with the methodology previously used for the 24P and 51B canisters; (3) the staff has incorporated specific dose rate limits in T.S. 1.2.7b for the storage module; and (4) the general licensee will operate the NUHOMS®-24PHB storage system with an established radiation protection program as required by 10 CFR Part 20, Subpart B.

5.4 Shielding Analyses

The applicant presented dose rates for normal conditions and accident conditions in Section N.5.4 and N.11 of the SAR amendment. The shielding analysis used ANSI/ANS Standard 6.1.1-1977 flux-to-dose rate conversion factors to calculate dose rates.

5.4.1 Normal Conditions

The applicant presented design-basis dose rates (Configuration 2) at various locations surrounding the HSM-102 and TC in Tables N.5-3 and N.5-4. The dose rates are provided for fuel with and without BPRAs. The dose rates for the HSM-102 are dominated by the gamma component. The peak dose on the roof of the HSM-102 module, the roof birdscreen, and centerline of the HSM-102 door are approximately 60 mrem/hr, 920 mrem/hr, and 12 mrem/hr, respectively. The dose rates for the transfer cask assume that there is three inches of supplemental shielding on top of the DSC during welding. For the transfer cask, there is a significant contribution from neutron radiation to the dose rates in addition to the gamma component. The peak dose at canister annulus is approximately 6 rem/hr. Chapter N.10 of the SAR amendment indicates that exposures from localized peak dose rate may be mitigated by ALARA practices and the actual locations of personnel, and by the use of temporary shielding during loading/unloading operations. The dose profiles discussed in Section N.5.4.9 for the TC further show that the dose rates significantly decrease from peak locations to the edges of the top, bottom, and sides of the cask.

5.4.2 Occupational Exposures

The applicant determined occupational exposures in Chapter N.10 of the SAR amendment. The exposures were based on estimations from surrounding dose rates calculated in Chapter N.5 and the operating procedures referenced in Chapter N.8. The staff found the occupational exposures to be acceptable as discussed in Chapter 10 of the SER.

5.4.3 Off-site Dose Calculations

The applicant estimated offsite dose rates from the 2x10 back-to-back array and the two 1x10 front-to-front arrays in Section N.10.2 of the SAR amendment. The offsite dose rates were based on Category 3 fuel design-basis source terms and Configuration 2 analyzed in Chapter N.5 of the SAR amendment. Tables N.10-6 through N.10-8 and Figure N.10-1 present the calculated offsite annual doses for these arrays at distances of 6 to 600 meters based on full-time occupancy. The off-site dose calculations are further evaluated in Section 10.4 of the SER.

5.4.4 Accident Conditions

Chapter N.11 of the SAR amendment does not identify an accident that significantly degrades the shielding of the HSM-102. The bounding accident condition for the HSM-102 considers reduced HSM-102 air inlet and outlet shielding which creates a 12-inch gap between the concrete HSMs. Section N.11.2.1.3 indicates a estimated recovery time of 5 days for this accident. This results in approximately 11 mrem dose to a person at 100 meters for a 40-hour exposure (i.e., 8 hrs/day), and less than 1 mrem to a person at 600 meters for a 120-hour exposure (i.e., 24 hrs/day).

Chapter N.11 of the SAR amendment identifies loss of water in TC water jacket as the bounding TC accident. Section N.11.2.5.3 compares the accident dose from the 24PHB/OS-197 system to the accident dose from the previously approved 24P/OS-197 system. The applicant estimated the TC accident dose rates for the 24PHB would be approximately 3.3 times higher than the doses for the 24P. Section 8.2.5.3.2 of the FSAR indicates an estimated recovery time of 8 hours to recover the neutron shield or replace temporary shielding around the TC after an accident. The applicant stated that this results in approximately 57 mrem to a person at 100-meters (8-hour exposure) and less than 1 mrem to a person at 2000 meters (8-hour exposure).

5.4.5 Staff Evaluation

Section 10 of this SER evaluates the overall dose (i.e., direct radiation and hypothetical radionuclide release) from the NUHOMS®-24PHB DSC system. The staff reviewed the dose calculations for normal operations and found them acceptable. The staff has reasonable assurance that compliance with 10 CFR Part 20 and 10 CFR 72.104(a) from direct radiation can be achieved by general licensees. The actual doses to individuals beyond the controlled area boundary depend on several site specific conditions such as fuel characteristic, cask-array configurations, topography, demographics, atmospheric conditions. In addition, 10 CFR 72.104(a) includes doses from other fuel cycle activities such as reactor operations. General licensees are responsible for verifying compliance with 10 CFR 72.104(a) in accordance with

10 CFR 72.212. A general licensee will also have an established radiation protection program as required by 10 CFR Part 20, Subpart B, and will demonstrate compliance with dose limits to individual members of the public and workers as required, by evaluation and measurements. As discussed in Section 10.4 of this SER, a pending requirement has been added as TS 1.1.9 (Amendment No. 5) regarding the use of engineered features used for radiological protection.

The staff reviewed the accident dose analysis and found it acceptable for the specific design and contents requested in the SAR amendment. The staff has reasonable assurance that the direct radiation from the NUHOMS®-24PHB system satisfies 10 CFR 72.106(b) at or beyond a controlled boundary of 100 meters from the design-basis accidents. Estimated dose to members of the public at 100 meters and at further distances for various exposure times are approximately two to three orders of magnitude below the 5 rem accident limit in 10 CFR 72.106(b). The staff notes that the off-site accident dose rate calculations may be less accurate than the dose rate calculations in the vicinity of the 24PHB/HSM-102 system, and that precise exposure times can not be predicted. However, the staff notes that direct radiation is relatively easy to mitigate within a reasonable amount of time, and that the exposure times are based on realistic assumptions.

The preferential loading schemes and decay heat limits specified in Figure N.2-1 and N.2-2, and the maximum burnup, minimum cooling time, and minimum enrichment limits specified in Tables N.2-3 through N.2-5 that form the basis for the dose rates are incorporated into Technical Specification 1.2.1. In addition, surface dose rate limits based on bounding analysis are incorporated into Technical Specification 1.2.7b.

5.5 Evaluation Findings

F5.1 The SAR amendment sufficiently describes shielding design features and design criteria for the structures, systems, and components important to safety.

F5.2 Radiation shielding features of the NUHOMS®-24PHB system (HSM-102) are sufficient to meet the radiation protection requirements of 10 CFR Part 20, 10 CFR 72.104, and 10 CFR 72.106.

F5.3 Operational restrictions to meet dose and ALARA requirements in 10 CFR Part 20, 10 CFR 72.104 and 72.106 are the responsibility of each general licensee. The NUHOMS®-24PHB shielding features (as approved by NRC) are designed to meet these requirements.

F5.4 The staff has not evaluated or approved the 24PHB canister for use in the HSM-80 design.

F5.5 The staff concludes that the design of the radiation protection system of the NUHOMS®-24PHB canister, when used with the HSM-102, is in compliance with 10 CFR Part 72 and the applicable design and acceptance criteria have been satisfied. The evaluation of the radiation protection system design provides reasonable assurance that the NUHOMS®-24PHB system will provide safe storage of spent fuel. This finding is based on a review that considered the regulation itself, the appropriate regulatory guides, applicable codes and standards, the applicant’s analyses, the staff’s confirmatory analyses, and acceptable engineering practices.

6.0 CRITICALITY EVALUATION

6.1 Criticality Design Characteristics and Features

The applicant proposed a revision to Technical Specification 1.2.15. The revision adds a new specification, 1.2.15b, providing a minimum boron concentration limit for the NUHOMS®-24PHB basket. The minimum boron concentration is 2350 ppm for all fuel assemblies with an initial enrichment less than or equal to 4.0 weight percent while loading the NUHOMS®-24PHB basket. Fuel assemblies with enrichments greater than 4.0 weight percent, up to a maximum of 4.5 weight percent shall follow TS Figure 1-10.

Figure 1-10 Soluble Boron Concentration vs Initial Fuel U-235 Enrichment	
Initial Enrichment (w/o)	Boron Loading, ppm
≤4.0	2350
4.1	2470
4.2	2580
4.3	2700
4.4	2790
4.5	2950

6.2 Fuel Specifications

The applicant also proposed a revision to Technical Specification 1.2.1 to add the fuel assembly specifications for B&W 15x15 fuel assemblies in Table 1-1i. SAR Section N.6.2, Tables N.6-1 through N.6.4 and Figure N.6-1 provide fuel assembly specifications and the fuel assembly layout for the B&W 15x15 assembly type.

6.3 Model Specifications

The applicant explicitly modeled the fuel assemblies in the NUHOMS®-24PHB basket with the soluble boron in the water. The applicant evaluated the cask both with and without BPRAs. Although the applicant modeled borated water in the cask, the applicant conservatively modeled the gap between the cladding and fuel pellet to contain fresh water. The applicant modeled borated water outside the DSC as the reflector, as when the DSC is loaded in the spent fuel pool.

6.4 Criticality Analysis

The applicant used the 44GROUPNDFB5 cross section set with the KENO V.a code in the SCALE 4.4a (Ref. 1) system to perform the criticality evaluation. The applicant performed an evaluation to determine the most reactive fuel assembly design within the class of assemblies designated as a B&W 15x15 assembly. Using the most reactive fuel assembly (B&W 15x15 Mark B10), the applicant varied the water density, with and without BPRAs, to determine optimum moderation. The applicant also performed a parametric study to evaluate the effects of variables such as; loading the basket with reconstituted fuel assemblies, leaving the center four fuel locations empty, and determining the minimum soluble boron concentration as a function of enrichment. The minimum boron level required to keep the calculated k-eff plus twice the Monte Carlo standard deviation below the Upper Subcritical Limit (USL) is shown in Figure 1-10. The applicant's maximum calculated k-eff was 0.9406, including the Monte Carlo uncertainty, which is less than the upper subcritical limit of 0.9413, from the applicant's benchmarking evaluation.

The staff performed confirmatory criticality calculations using KENO V.a with the 238GROUPNDFB5 cross section set in the SCALE 4.4a system. The staff performed confirmatory calculations for all six different enrichments. The staff's model is similar to the applicant's. The staff's model included borated water in all locations containing water, except for the fuel rod gap, similar to the applicant's model. The staff evaluated the reactivity of the system with fresh water in the fuel rod gap, similar to the applicant's model. The staff's maximum calculated k-eff was 0.9329, including Monte Carlo uncertainty, for B&W 15x15 Mark B10 fuel assemblies with an enrichment of 4.4 weight percent and a boron concentration of 2790 ppm in the water.

6.5 Benchmark Comparison

The applicant performed a benchmarking analysis for the SCALE 4.4 system. The applicant chose 125 critical experiments, which are included in NUREG/CR-6361 (Ref. 2). The applicant determined the Upper Subcritical Limit (USL) using method 1 from NUREG/CR-6361. The applicant evaluated the USL for a number of parameters, such as enrichment, fuel rod pitch, water/fuel volume ratio, assembly separation, and average energy group causing fission. The most limiting USL was calculated based on the limiting value for each parameter and the lowest USL was taken to be the bounding value. The applicant determined the bounding USL to be 0.9413

6.6 Evaluation Findings

6-1 The cask and its spent fuel transfer systems are designed to be subcritical in all configurations.

6-2 The criticality design is based on favorable geometry and soluble poisons in the spent fuel pool. Based on the information provided in the application and the staff's own confirmatory analyses, the staff concludes that the NUHOMS®-24PHB system meets the acceptance criteria specified in 10 CFR Part 72.

6-3 The staff reviewed the applicant's benchmark analysis and agrees that the critical experiments chosen are relevant to the cask design. The staff found the applicant's method for determining the USL acceptable. The staff also verified that only biases that increase k_{eff} have been applied.

6.7 References

1. Oak Ridge National Laboratory, "SCALE: A Modular Code System for Performing Standardized Computer Analyses for Licensing Evaluations," NUREG/CR-0200, Vol. 1-3, Revision 6, 2000.

2. Oak Ridge National Laboratory, Criticality Benchmark Guide for Light-Water-Reactor Fuel in Transportation and Storage Packages, NUREG/CR-6361, March 1997.

7.0 CONFINEMENT EVALUATION

The staff reviewed the NUHOMS®-24PHB DSC confinement features and capabilities to ensure a) that any radiological releases to the environment will be within the limits established by the regulation (Ref. 1), and b) that the spent fuel cladding will be protected against degradation that might lead to gross ruptures during storage, as required in 10 CFR 72.122(h)(1). This amendment was also reviewed to determine whether the NUHOMS®-24PHB DSC fulfills the acceptance criteria listed in Section 7 of NUREG-1536, Standard Review Plan for Dry Cask Storage Systems (Ref. 2), and applicable Interim Staff Guidance documents (ISGs). The staff's conclusions are based on information provided in the NUHOMS®-24PHB DSC SAR.

7.1 Confinement Design Characteristics

A description of the confinement boundary is given in Section N.7.1.1 of the amendment request. The confinement boundary consists of a shell which is a welded stainless steel cylinder with an integrally-welded, stainless steel bottom closure assembly, and a stainless steel top closure assembly, which includes the vent and drain system. The inner top cover plate has two penetrations for the vent and siphon ports which are closed with welded cover plates. The outer top and bottom cover plates provide redundant sealing of the confinement system. The system is designed to be leaktight as defined by Reference 3. The outer top cover plate has a single penetration to leak test the closure welds. This is closed with a welded cover plate after testing to complete the redundant sealing of the confinement boundary. The welds forming the confinement boundary are described in detail in Sections N.7.1.3 of the SAR.

The redundant closure of the DSC satisfies the requirements of 10 CFR 72.236(e) for redundant sealing of confinement systems.

The DSC is designed, fabricated, and tested in accordance with the applicable requirements of the ASME Code Section III, Subsection NB to the maximum extent practicable. The staff concludes that the description of the confinement boundary satisfies the requirements of 10 CFR 72.24(c)(3).

The applicant proposed minimal changes to the procedures for drying and evacuating the cask interior during loading operations. The only changes to the procedures were setting the maximum vacuum drying duration limit in Technical Specification 1.2.17b. The vacuum drying duration limit for a heat load between 12 and 24 kW is 29 hours. For heat loads less than 12 kW the vacuum drying duration is 32 hours. The maximum vacuum drying duration was reviewed by the staff to ensure that the design is acceptable for the temperatures that may be experienced during the drying. The staff finds that this design, if fabricated and tested in accordance with the SAR requirements, will maintain the confinement boundary. The NUHOMS®-24PHB DSC is designed to be leaktight and is tested to a leak rate of $1\text{x}10^{-7}$ atm cm^3/sec, as defined in ANSI N14.5-1997. This testing confirms that the amount of helium lost from the NUHOMS®-24PHB DSC over the approved storage period is negligible. Thus, an adequate amount of helium will remain in the canister to maintain an inert atmosphere and to support the heat transfer during the storage period.

For normal storage conditions, the NUHOMS®-24PHB DSC uses multiple confinement barriers provided by the fuel cladding (for intact fuel) and the canister to assure that the confinement system will reasonably maintain confinement of radioactive material. The canister is backfilled with an inert gas (helium) to protect against cladding degradation. Section 3 of the SER addresses all confinement boundary components being maintained within their code-allowable stress limits during normal storage conditions. Section 4 of the SER shows that the peak confinement boundary component temperatures and pressures are within the design-basis limits for normal conditions of storage.

There were no changes to the welding and weld examinations which include the following; multiple surface and volumetric examinations, pneumatic pressure testing, leakage rate testing on the finished shell and the inner cover plate at the fabricator, leakage rate testing of the closure welds (inner top cover plate and vent and siphon port cover plates) after loading the spent fuel, and multiple surface and dye penetrant examinations on the redundant confinement boundary.

The all-welded construction of the NUHOMS®-24PHB DSC with the redundant closure, extensive inspection and testing, ensures that no release of radioactive material for normal storage and on-site transfer will occur.

7.2 Confinement Monitoring Capability

For redundant seal welded closures, continuous monitoring of the closure is not necessary because there is no known plausible, long-term degradation mechanism which would cause the seal welds to fail. Periodic surveillance and monitoring of the storage module thermal performance, as well as the licensee's use of radiation monitors are adequate to ensure the continued effectiveness of the confinement boundary. The staff finds this adequate to enable the licensee to detect any closure degradation and take appropriate corrective actions to maintain safe storage conditions.

7.3 Nuclides with Potential Release

Since the NUHOMS®-24PHB DSC is designed, fabricated, and tested to meet the leak tight criteria of Reference 3, there is no contribution to the radiological consequences due to a potential release of canister contents.

7.4 Confinement Analysis

The confinement boundary is welded and tested to meet the leak tight criteria of Reference 3, and is shown to maintain confinement during all normal, off-normal, and hypothetical accident conditions. Additionally, the temperature and pressure of the canister are within the design-basis limits. Therefore, no discernable leakage is credible. As discussed in Section 10 of this SER, the staff finds that the 24PHB DSC meets the requirements of 10 CFR 72.104(a) and 10 CFR 106(b).

7.5 Maximum Pressure Loads

The maximum design basis internal pressures in the NUHOMS®-24PHB DSC have been evaluated in Section 4.0 of this SER.

7.6 Misloading

The NUHOMS®-24PHB DSC may store PWR fuel assemblies arranged in any of two alternate heat zoning configurations with a maximum decay heat of 1.3 kW per assembly and a maximum heat load of 24 kW per canister. These different loading patterns are shown in Figures N.2-1 and N.2-2 in the SAR. Currently, the NRC staff believes that a misloading, or inadvertent placement of a fuel assembly with too high of a heat load in an incorrect location, is a credible event. However, the applicant has included additional administrative requirements to help minimize the possibility of a misloading occurring. These additional requirements are included as additional checks in SAR Chapter N.8, “Operating Systems,” and assure that a “double contingency” criteria is applied for misloading of an assembly or BPRA hardware. These requirements are summarized below:

a) the utility must prepare loading maps of fuel assemblies including control components to be loaded in a given canister before fuel load based on technical specification,

b) this loading map is required to be independently verified before any fuel loading, and

c) additional independent verification that the loading map is followed correctly and accurately after the fuel load but before the top shield plug is placed.

Further, the current staff position is that the consequences of a misloading accident are not safety significant and thus, the overall risk from a misloading is low.

7.7 Supportive Information

Supportive information or documentation includes drawings of the NUHOMS®-24PHB DSC vent port and applicable pages from referenced documents.

7.8 Evaluation Findings

F7.1 Section N.7 of the SAR describes confinement structures, systems, and components important to safety in sufficient detail to permit evaluation of their effectiveness.

F7.2 The design of the NUHOMS®-24PHB DSC adequately protects the spent fuel cladding against degradation that might otherwise lead to gross ruptures. Section 4 of the SER discusses the relevant temperature considerations.

F7.3 The design of the NUHOMS®-24PHB DSC provides redundant sealing of the confinement system closure joints using dual welds on the canister lid and closure.

F7.4 The 24PHB DSC has no bolted closures or mechanical seals. The confinement boundary contains no external penetrations for pressure monitoring or overpressure protection. No instrumentation is required to remain operational under accident conditions. Since the 24PHB DSC uses an entirely welded redundant closure system, no direct monitoring of the closure is required.

F7.5 The confinement system is leaktight for normal conditions and anticipated occurrences, thus the confinement system will reasonably maintain confinement of radioactive material. Section 10 of the SER shows that the direct dose from the NUHOMS®-24PHB DSC satisfies the regulatory requirements of 10 CFR 72.104(a) and 10 CFR 72.106(b).

F7.6 The confinement system has been evaluated by analysis. Based on successful completion of specified leakage tests and examination procedures, the staff concludes that the confinement system will reasonably maintain confinement of radioactive material under normal, off-normal, and credible accident conditions.

F7.7 The staff concludes that the design of the confinement system of the NUHOMS®-24PHB DSC is in compliance with 10 CFR Part 72 and that the applicable design and acceptance criteria have been satisfied. The evaluation of the confinement system design provides reasonable assurance that the NUHOMS®-24PHB DSC will allow safe storage of spent fuel. This finding considered the regulation itself, the appropriate regulatory guides, applicable codes and standards, the applicant's analyses, the staff's confirmatory analyses, and acceptable engineering practices.

7.9 References

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

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

3. American National Standard, "Radioactive Materials - Leakage Tests on Packages for Shipment," ANSI N14.5-1997.

8.0 OPERATING PROCEDURES

The review of the technical bases for the operating procedures is to ensure that the applicant's SAR presents acceptable operating sequences, guidance, and generic procedures for key operations. The procedures for the NUHOMS®-24PHB DSC, as described in Section N.8.1 of the SAR are very similar to those previously approved by the staff for the Standardized NUHOMS® System (Ref. 1).

8.1 Cask Loading

Detailed loading procedures must be developed by each user.

The loading procedures described in the SAR include appropriate preparation and inspection provisions to be accomplished before cask loading. These include cleaning and decontaminating the transfer cask and other equipment as necessary, and performing an inspection of the NUHOMS®-24PHB DSC to identify any damage that may have occurred since receipt inspection.

8.1.1 Fuel Specifications

The procedures described in Section N.8.1.2 of the SAR provide for fuel handling operations to be performed in accordance with the general licensee's 10 CFR Part 50 license and requires independent, dual verification, of each fuel assembly loaded into the NUHOMS®-24PHB DSC. It outlines appropriate procedural and administrative controls to preclude a cask misloading.

8.1.2 ALARA

The ALARA practices utilized during operations are discussed in Section 10.5 of this SER and are found to be acceptable.

8.1.3 Draining, Drying, Filling and Pressurization

Section N.8.1.3 of the SAR describes draining, drying, filling and pressurization procedures for the NUHOMS®-24PHB DSC. These procedures provide reasonable assurance that an acceptable level of moisture remains in the cask and the fuel is stored in an inert atmosphere. The procedures for helium backfill pressure (TS 1.2.3a) are the same as those previously approved by the staff for the Standardized NUHOMS® System. The procedures for DCS cavity boron concentration during filling (TS1.2.15b) and DCS vacuum drying time (TS1.2.17b) are specific to the 24PHB design.

8.1.4 Welding and Sealing

Welding and sealing operations of the NUHOMS®-24PHB DSC are similar to that previously approved by the staff for other DSCs used with the Standardized NUHOMS® System. The procedures include monitoring for hydrogen during welding operations. As discussed in Section 7.0 of this SER, leak checks required by TS 1.2.4a for the NUHOMS®-24PHB DSC demonstrate that the inner top cover plate is leak tight as defined by ANSI N14.5-1997 (Ref. 2).

Sealing operations for dye penetrant testing of the closure welds are performed in accordance with TS 1.2.5.

8.2 Cask Handling and Storage Operations

All handling and transportation events applicable to moving the NUHOMS®-24PHB DSC to the storage location are the same as those previously reviewed by the staff for the Standardized NUHOMS® System and are bounded by Section N.11 of the SAR. Monitoring operations include daily surveillance of the HSM air inlets and outlets in accordance with TS 1.3.1, and temperature performance is monitored on a daily basis in accordance with TS 1.3.2. Occupational and public exposure estimates are evaluated in Section N.10 of the SAR. Each cask user will need to develop detailed cask handling and storage procedures that incorporate ALARA objectives of their site-specific radiation protection program.

8.3 Cask Unloading

Detailed unloading procedures must be developed by each user.

Section N.8 provides the same unloading procedures as those previously approved by the staff for use with the Standardized NUHOMS® System. The procedures provide a caution on reflooding the DSC to ensure that the cask vent pressure does not exceed 20 psig to prevent damage to the cask.

Section N.8 provides a discussion of ALARA practices that should be implemented during unloading operations, however, detailed procedures incorporating provisions to mitigate the possibility of fuel crud particulate dispersal and fission gas release must be developed by each user.

8.4 Evaluation Findings

F8.1 The NUHOMS®-24PHB DSC is compatible with wet loading and unloading. General procedure descriptions for these operations are summarized in Section N.8 of the applicant’s SAR. Detailed procedures will need to be developed and evaluated on a site-specific basis.

F8.2 The welded cover plates of the cask allow ready retrieval of the spent fuel for further processing or disposal as required.

F8.3 The DSC geometry and general operating procedures facilitate decontamination. Only routine decontamination will be necessary after the cask is removed from the spent fuel pool.

F8.4 No significant radioactive waste is generated during operations associated with the independent spent fuel storage installation (ISFSI). Contaminated water from the spent fuel pool will be governed by the 10 CFR Part 50 license.

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F8.5 No significant radioactive effluents are produced during storage. Any radioactive effluents generated during the cask loading will be governed by the 10 CFR Part 50 license.

F8.6 The technical bases for the general operating procedures described in the SAR are adequate to protect health and minimize danger to life and property. Detailed procedures will need to be developed and evaluated on a site-specific basis.

F8.7 Section 10 of the SER assesses the operational restrictions to meet the limits of 10 CFR Part 20. Additional site-specific restrictions may also be established by the site licensee.

F8.8 The staff concludes that the generic procedures and guidance for the operation of the NUHOMS®-24PHB DSC are in compliance with 10 CFR Part 72 and that the applicable acceptance criteria have been satisfied. The evaluation of the operating procedure descriptions provided in the SAR offers reasonable assurance that the cask will enable safe storage of spent fuel. This finding is based on a review that considered the regulations, appropriate regulatory guides, applicable codes and standards, and accepted practices.

8.5 References

1. Transnuclear West, Final Safety Analysis Report of the Standardized NUHOMS® Modular Storage System for Irradiated Nuclear Fuel, October 2001, Revision 6.

2. ANSI N14.5-1997, “Radioactive Materials - Leakage Tests on Packages for Shipment.”

9.0 ACCEPTANCE TESTS AND MAINTENANCE PROGRAM

The acceptance test requirements and maintenance program for the NUHOMS®-24PHB System are the same as for the previously approved NUHOMS® System with one exception. The exception is the confinement leak testing in accordance with ANSI N14.5-1997. The confinement leakage test requirements are discussed in Section 7 of this SER.

10.0 RADIATION PROTECTION EVALUATION

The staff evaluated the radiation protection design features, design criteria, and the operating procedures of the NUHOMS®-24PHB DSC which will be used with the Standardized NUHOMS® Horizontal Storage Module 102 (HSM-102) to ensure that the DSC will meet the regulatory dose requirements of 10 CFR Part 20, 10 CFR 72.104(a), 10 CFR 72.106(b), 10 CFR 72.212(b), and 10 CFR 72.236(d).

10.1 Radiation Protection Design Criteria and Design Features

The radiological protection design criteria are the limits and requirements of 10 CFR Part 20, 10 CFR 72.104, and 10 CFR 72.106. As required by 10 CFR Part 20 and 10 CFR 72.212, each general licensee is responsible for demonstrating site-specific compliance with these requirements. In addition, the staff has incorporated Technical Specification (TS) 1.2.7b to establish direct radiation dose rate limits for the HSM-102. These values are based on calculated dose rate values which are used to determine occupational and off-site exposures. The applicant has requested removal of TS 1.2.11 for transfer cask (TC) dose rates because they are redundant. TS 1.2.12 established exterior contamination limits on the DSC to keep non-fixed beta-gamma contamination below 2,200 dpm/100 cm^2, and non-fixed alpha contamination below 220 dpm/100 cm^2.

The radiation protection design features are referenced in Section N.10 of the SAR amendment. The radiation protection design features of the NUHOMS®-24PHB/HSM-102 system are the same as the radiation protection design features of the previously approved NUHOMS® 24P/HSM-80 system, with the exception of the revised door and vent designs in the HSM-102 and the addition of a test port to the top cover plate of the 24PHB. The 24PHB canister is tested leaktight in accordance with ANSI N14.5 and the applicant did not request any other physical designs changes to the 24P canister system and HSM-102.

The staff reviewed the design criteria and found it acceptable. The staff could not verify the applicant’s statement that TS 1.2.11 for the TC was redundant with respect to its objectives and basis. Therefore, TS 1.2.11 is maintained as previously approved and with clarifications for the applicable containers. TS 1.2.11a has also been added for 24PHB dose rates on the side and top (edge of cover plate) of the TC, as calculated in Chapter N.5 of the amendment. The staff did not evaluate the radiation protection design features and design criteria for the NUHOMS®-24PHB/HSM-102 because it is essentially the same as the previously approved NUHOMS® 24/HSM-80. The staff did, however, review the integrated shielding ability of the 24PHB/HSM-102 and found it acceptable. Chapters 5, 7, and 8 of the SER discuss specific staff evaluations of the design criteria and features for the shielding system, confinement systems, and operating procedures, as appropriate. Chapter 11 of the SER discusses staff evaluations of the capability of the shielding and confinement features during off-normal and accident conditions, as appropriate.

10.2 ALARA

The ALARA objectives, procedures, practices, and policies are referenced in Section N.10 of the SAR amendment and the previously approved FSAR. The ALARA objectives, procedures, practices, and policies of the NUHOMS®-24PHB/HSM-102 system are the same as the previously approved NUHOMS® 24P/HSM-80 system. Each site licensee will apply its additional site-specific ALARA objectives, policies, procedures, and practices for members of the public and personnel.

The staff evaluated the previously approved ALARA assessment for the NUHOMS®-24PHB/HSM-102 system and found it acceptable. Section 8 of the SER discusses the staff's evaluation of the operating procedures with respect to ALARA principles and practices, as appropriate. Operational ALARA objectives, policies, procedures, and practices are the responsibility of the site licensee as required by 10 CFR Part 20 and 10 CFR 72.104(b). In addition, the staff has incorporated TS limits for dose rates and surface contamination limits to ensure that public and occupational exposures are maintained ALARA.

10.3 Occupational Exposures

The applicant determined occupational exposures in Section N.10.1 of the SAR amendment. The exposures were based on estimations from surrounding direct-radiation dose rates calculated in Chapter N.5 and the operating procedures referenced in Chapter N.8. The dose rates were based on Configuration 2 preferential loading option with design-basis source terms. The operating procedures are generic procedures that general licensees will use for fuel loading, canister/TC operations, canister transfer into the HSM-102, and fuel unloading. Table N.10-1 of the SAR amendment shows the estimated number of personnel, the estimated time, and estimated dose rate, for each task involved in the loading of a NUHOMS®-24PHB/HSM-102 system. The dose estimates indicate that the total occupational dose is approximately 3.1 person-rem for a single loading. The applicant indicated that the general licensees may choose to modify the sequence of operations, and will also use ALARA practices to mitigate occupational exposure.

The staff reviewed the overall occupational dose estimates and found them acceptable. The occupational dose exposure estimates provide reasonable assurance that occupational limits in 10 CFR Part 20, Subpart C can be achieved. The staff expects actual operating times and personnel exposure rates will vary for each system depending on site-specific operating conditions, including detailed procedures and special measures taken to maintain exposures ALARA. Each licensee will have an established radiation protection program, as required in 10 CFR Part 20, Subpart B. In addition, each licensee will demonstrate compliance with occupational dose limits in 10 CFR Part 20, Subpart C and other site-specific 10 CFR Part 50 license requirements with evaluations and measurements. Staff evaluation of the operating procedures is presented in Section 8 of the SER.

10.4 Public Exposures From Normal and Off-Normal Conditions

Section N.10.2 of the SAR amendment presents the calculated direct radiation dose rates at distances beyond 100 meters from a sample cask array configuration loaded with design basis fuel. Figure N.10-1 depicts estimated dose rate versus distance curves. Table N.10-2 specifies dose rates versus distance from both analyzed ISFSI configurations. An array of 20 NUHOMS®-24PHB/HSM-102 systems (2x10 array or two 1x10 arrays) loaded with design basis fuel are below the regulatory limits of 25 mrem/yr at approximately 400 meters. This assumes 100% occupancy for 365 days.

The staff evaluated the public dose estimates during normal and off-normal conditions and found them acceptable. The primary dose pathway to individuals beyond the controlled area during normal and off-normal conditions is from direct radiation (including skyshine). The canister is tested leaktight and the confinement function is not affected by normal or off-normal conditions. Therefore, no content leakage is credible. A discussion of the staff's evaluation and confirmatory analysis of the shielding calculations and confinement analysis are presented in Section 5 and 7 of the SER, respectively.

The staff has reasonable assurance that compliance with 10 CFR 72.104(a) can be achieved by each general licensee. The general licensee using the NUHOMS®-24PHB/HSM-102 system must perform a site-specific evaluation, as required by 10 CFR 72.212(b) to demonstrate compliance with 10 CFR 72.104(a). The actual doses to individuals beyond the controlled area boundary depend on several site-specific conditions such as fuel characteristics, cask-array configurations, topography, demographics, and use of engineered features (e.g., berm). In addition, the dose limits in 10 CFR 72.104(a) include doses from other fuel cycle activities such as reactor operations. Consequently, final determination of compliance with 10 CFR 72.104(a) is the responsibility of each general licensee.

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

A pending requirement has been added as TS 1.1.9 (Amendment No. 5) regarding the use of engineered features used for radiological protection. The TS states that engineered features (e.g., earthen berms, shield walls) that are used to ensure compliance with 10 CFR 72.104(a) by each general licensee are to be considered important to safety and must be appropriately evaluated under 10 CFR 72.212(b).

10.5 Public Exposures From Design-Basis Accidents and Natural Phenomena Events

Section N.11 of the SAR amendment presents the calculated dose rates for accident conditions and natural phenomena events to individuals beyond the controlled area. The confinement function of the canister is not affected by design-basis accidents or natural phenomena events. Therefore, there is no credible release of contents. As discussed in Section 5.4.4 of the SER, the accident direct-radiation dose analysis indicates the worst case shielding consequences results in a dose at the controlled area boundary that are well below the regulatory requirements of 10 CFR 72.106(b). Chapter 11 of the SAR amendment discusses or references the corrective actions for each design-basis accident, as appropriate.

The staff evaluated the public dose estimates from direct radiation and contents from accident conditions and natural phenomena events, and found them acceptable. A discussion of the staff's evaluation of the shielding and confinement analysis for the Chapter N.11 accidents is presented in Sections 5 and 7 of this SER, respectively. A discussion of the staff's evaluation of the accident conditions and recovery actions are presented in Section 11 of this SER, as appropriate. The staff has reasonable assurance that the effects of direct radiation from bounding design basis accidents and natural phenomena will be below the regulatory limits in 10 CFR 72.106(b).

10.6 Evaluation Findings

F10.1 The SAR amendment sufficiently describes the radiation protection design bases and design criteria for the structures, systems, and components important to safety.

F10.2 Radiation shielding and confinement features are sufficient to meet the radiation protection requirements of 10 CFR Part 20, 10 CFR 72.104, and 10 CFR 72.106.

F10.3 The NUHOMS®-24PHB DSC is designed to provide redundant sealing of the confinement system.

F10.4 The NUHOMS®-24PHB DSC is designed to facilitate decontamination to the extent practicable.

F10.5 The SAR amendment adequately evaluates the NUHOMS®-24PHB DSC and its systems important to safety to demonstrate that they will reasonably maintain confinement of radioactive material under normal, off-normal, and accident conditions.

F10.6 The SAR amendment sufficiently describes the means for controlling and limiting occupational exposures within the dose and ALARA requirements of 10 CFR Part 20.

F10.7 Operational restrictions necessary to meet dose and ALARA requirements in 10 CFR Part 20, 10 CFR 72.104, and 10 CFR 72.106 are the responsibility of the site licensee. The NUHOMS®-24PHB DSC is designed to assist in meeting these requirements.

F10.8 The staff concludes that the design of the radiation protection system of the NUHOMS®-24PHB/HSM-102 system is in compliance with 10 CFR Part 72 and the applicable design and acceptance criteria have been satisfied. The evaluation of the radiation protection system design provides reasonable assurance that the NUHOMS®-24PHB DSC will provide safe storage of spent fuel. This finding is based on a review that considered the regulation itself, the appropriate regulatory guides, applicable codes and standards, the applicant's analyses, the staff's confirmatory analyses, and acceptable engineering practices.

11.0 ACCIDENT ANALYSIS EVALUATION

11.1 Dose Limits for Off-Normal Events

Section N.11.1.4 of the SAR amendment examines the dose consequences for the identified off-normal events. The NUHOMS®-24PHB is tested leaktight in accordance with ANSI N14.5 and there will be no breach of the confinement boundary due to off normal conditions. The direct radiation conditions are the same as normal conditions analyzed in Chapter N.5 and N.10 of the SAR amendment.

The staff reviewed the consequences of postulated off-normal events with respect to 10 CFR 72.104(a) dose limits, and found them acceptable. The radiation consequences from off-normal events are the same as for normal conditions of operation. The staff has reasonable assurance that the dose to any individual beyond the controlled area will not exceed the limits in 10 CFR 72.104(a) during off-normal conditions (anticipated occurrences). Sections 5, 7, and 10 of this SER further evaluate the radiological doses applicable to off-normal events.

11.2 Dose Limits for Design-Basis Accidents and Natural Phenomena Events

Section 11.2 of the SAR amendment examines the dose consequences for the identified design-basis accidents and natural phenomena events. The NUHOMS®-24PHB is tested leaktight in accordance with ANSI N14.5 and there will be no breach of the confinement boundary due to accident conditions. The applicant determined the direct radiation dose to be less than 57 mrem for the events examined in Section 11.2 of the SAR amendment.

The staff reviewed the design-basis accident analyses with respect to 10 CFR 72.106(b) dose limits and found them acceptable. The staff has reasonable assurance that the dose to any individual at or beyond the controlled area boundary of 100 meters will not exceed the limits in 10 CFR 72.106(b). The staff notes that structural, material, or thermal design changes of the HSM-102 that were added to the FSAR under 10 CFR 72.48 were not evaluated for design-basis accidents and natural phenomena events in this CoC amendment (see Section 5.1.1 of the SER). Chapters 5, 7, and 10 of the SER further evaluate the estimated radiological doses during accident conditions.

12.0 CONDITIONS FOR CASK USE - TECHNICAL SPECIFICATIONS

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

12.1 Conditions for Use

The conditions for use of the NUHOMS®-24PHB DSC, in conjunction with the Standardized NUHOMS® Storage System, are clearly defined in the CoC and TS.

12.2 Technical Specifications

Based on the addition of the NUHOMS®-24PHB DSC to the Standardized NUHOMS® Storage System, the TS have been revised to accommodate the new DSC and the fuel types to be stored in the DSC. These changes have been identified in the TS attachment to the COC.

Table 12-1 lists the TS for use of the NUHOMS®-24PHB DSC system, in concert with the Standardized NUHOMS® Storage System.

12.3 Evaluation of Findings

F12.1 Table 12-1 of this SER lists the TS for the NUHOMS®-24PHB DSC, in conjunction with the Standardized NUHOMS® Storage System. These TS are included as Appendix A of the CoC.

F12.2 The staff concludes that the conditions for use of the NUHOMS®-24PHB DSC, in conjunction with the Standardized NUHOMS® Storage system, identify necessary TS to satisfy 10 CFR Part 72 and that the applicant acceptance criteria have been satisfied. The TS provide reasonable assurance that the cask will provide for safe storage of spent fuel. This finding is reached on the basis of a review that considered the regulation itself, appropriate regulatory guides, applicable codes and standards, and accepted practices.

Table 12-1

Standardized NUHOMS® Horizontal Modular Storage System Technical Specifications for use with the NUHOMS®-24PHB DSC

13.0 QUALITY ASSURANCE

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

14.0 DECOMMISSIONING

The decommissioning evaluation was previously reviewed and approved in the Standardized NUHOMS® Horizontal Modular Storage System FSAR. There were no changes proposed by the applicant in the addition of the NUHOMS®-24PHB DCS.

CONCLUSION

The NRC staff has performed a comprehensive review of the CoC amendment request and found that the addition of the NUHOMS®-24PHB DSC and high burnup fuel does not reduce the safety margin for the Standardized NUHOMS® System. The areas of review addressed in NUREG 1536, "Standard Review Plan for Dry Cask Storage Systems," January 1997, are consistent with the applicant's proposed changes. The Certificate of Compliance has been revised to include the TN requested changes. Based on the statements and representations contained in the TN's application, as supplemented, the staff concludes that the addition of the NUHOMS®-24PHB dry shielded canister and the addition of high burnup fuel to the approved contents of the Standardized NUHOMS® System meets the requirements of 10 CFR Part 72.

Issued with Certificate of Compliance No. 1004, Amendment No. 6 on **DRAFT** .