SAFETY EVALUATION REPORT

Docket No. 71-9255
Model No. NUHOMS® MP187 Transportation Package
Certificate of Compliance No. 9255
Revision No. 6

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SUMMARY

By application dated January 30, 2001, as supplemented on August 24, 2001, and September 21, 2001, Transnuclear West Inc., (TN West) and Transnuclear Inc., requested a revision of Certificate of Compliance (CoC) No. 9255, to add a fourth Dry Shielded Canister (DSC) designated the NUHOMS® 24PT1-DSC and its associated payload of intact or damaged Westinghouse (WE) 14x14 spent fuel. The 24PT1-DSC included in the Standardized Advanced NUHOMS® System is nearly identical to that previously approved for transportation in the NUHOMS® MP187 transportation package, and for use at the Rancho Seco Nuclear Plant in accordance with special nuclear material license SNM-2510. By letter dated October 4, 2001, Transnuclear Inc. requested that the name change to the CoC be made from Transnuclear West, Inc., to Transnuclear Inc. The staff used NUREG-1617 "Standard Review Plan for Transportation Packages for Spent Nuclear Fuel" to conduct its review. The submission was primarily reviewed to ensure that the previous analysis demonstrated the ability of the 24PT1-DSC to be used with the MP187. This amendment satisfies 10 CFR Part 71 and is therefore approved.

References

Transnuclear West Inc., application dated January 30, 2001.

Transnuclear West Inc., supplements dated August 24, 2001, and September 21, 2001.

Transnuclear Inc., letter dated October 4, 2001.

1.0 GENERAL INFORMATION

The 24PT1 DSC is designed to contain 24 intact pressurized water reactor (PWR) WE 14x14 fuel assemblies with or without rod cluster control assemblies (RCCAs), neutron source assemblies (NSAs), or thimble plug assemblies (TPAs). Westinghouse 14x14 fuel assemblies mixed oxide fuel may also be stored with or without inserts. The 24PT1-DSC is also designed to store up to 20 intact and 4 damaged fuel assemblies in failed fuel cans. In addition, two slots in the 24PT1-DSC may be filled with stainless steel model assemblies which do not contain fuel. These model (or dummy) assemblies would be used in the event that the cask user requires only 22 spent fuel assemblies to be stored in the cask. The 24PT1-DSC is designed for a maximum heat load of 14kW.

The 24PT1-DSC consists of a cylindrical shell with welded top and bottom cover plates which form the confinement boundary. Shield plugs are installed inside of the confinement boundary, at the top and bottom, to provide radiological shielding. Inside of the 24PT1-DSC is a basket assembly that consists of 24 guide sleeves which contain fixed borated neutron absorbing material for criticality control during loading operations. The structural support for the PWR fuel and basket guide sleeves is provided by circular spacer disc plates.

TN West provided revised drawings for the 24PT1-DSC. The staff determined that the drawings contain sufficient detail on dimensions, materials, and specifications to allow for a thorough evaluation of the 24PT1-DSC when used in the MP-187 transportation package.

2.0 STRUCTURAL

The application adds a fourth DSC, designated as 24PT1-DSC, and its associated payload (WE 14x14) to the cask. There are four different types of DSCs for the MP187 packages: the Fuel Only (FO)-DSC, the Fuel/Control Components (FC)-DSC, the Failed Fuel (FF)-DSC and the 24PT1-DSC. All four DSCs have the same outside diameter and overall length. For the FO/FC/24PT1 DSCs, the basket assembly consists of twenty-six carbon steel spacer discs with 24 openings each, supported by four high strength stainless steel support rods. For the 24PT1-DSC, up to four stainless steel failed fuel cans may be placed in the canister which provide confinement for the damaged fuel assembly. The FO/FC/FF DSCs have been previously reviewed and approved by staff. Therefore, except for new components such as fuel spacers and analyses which had not been performed before, the applicable analyses and calculations from the previous DSCs are used as references and guidelines for review and evaluation of the 24PT1-DSC components. The staff's analysis consists of review of performance of the DSC during "Normal Conditions of Transport" and "Hypothetical Accident Conditions":

For Normal Conditions of Transport:

1. MP187 cask analysis: Load cases such as pressure, vibration, water spray, free drop, corner drop, compression and fabrication stress, analyses for FO/FC/FF-DSC configurations bound the 24PT1-DSC payload.

2. 24PT1-DSC:

    1) Spacer discs: The FO/FC spacer disc analyses bound the 24PT1 DSC spacer discs and are directly applicable.

    2) Support rods: The configuration of the support rod assembly in the 24PT1-DSC is similar to the configuration in the FO-DSC. However, due to decrease of nominal OD of the support rods, and decrease of preload in the rod assembly, the capacity of the support rod assemblies has increased. Therefore, the FO-DSC analyses are applicable.

    3) Guide sleeves: The structure of the guide sleeve assemblies are identical to those in the FO-DSC with some exceptions which serve to enhance the structural capacity of the guide sleeves. For instance, the corner fillet welds are replaced by full penetration welds, thus increasing the capacity of the sleeve. The FO-DSC analyses are applicable.

4) Failed Fuel Cans: The failed fuel cans act in parallel with the guide sleeves and are fabricated using the same material as the guide sleeves and are sized to fit inside the guide sleeves. The induced loads will be shared equally between the guide sleeves and failed fuel can. The guide sleeve analyses apply to the failed fuel cans and no additional analysis is necessary.

5) Fuel Spacers: The WE 14x14 fuel assemblies are shorter than the B&W 15x15 fuel assemblies, therefore, top and bottom fuel spacers are utilized to center the fuel in the DSC. Fuel spacers are not used in the FO/FC/FF-DSCs. Fuel spacers are designed for all normal conditions including vibration loads and the 1-foot side drop.

For the Vibration Analysis:

1) Inertia weight of a fuel assembly (2.5g) applies to the spacer in the axial direction. Stresses in the tube section of the fuel spacer are calculated and are below the allowable stress intensity.

2) 1-Foot Side Drop: A simple linear elastic ANSYS model is used to evaluate the one-foot side drop, based on an inertia load of 25.5g. The calculated stress ratios (highest is 0.28) are low and, therefore, acceptable.

For the Hypothetical Accident Conditions (HAC):

1. MP187 cask analysis: The previous FO/FC-DSC HAC loads bound the loads for the 24PT1-DSC, and are acceptable.

2. 24PT1-DSC analysis:

    1) Spacer disc stress and stability: The FO/FC spacer disc loading bounds the 24PT1-DSC disc loading and the geometry of the spacer discs is the same. Also, the stability analysis of the FO/FC-DSC bounds those of the 24PT1-DSC. The staff finds the stress and stability evaluation acceptable.

    2) Support rod assembly stress and stability: A conservative side drop load of 120g equivalent acceleration is used for the evaluation and bounds the FO/FC rod design. For a 60g end drop, the controlling stresses are in the spacer sleeves. Interaction Equations 20, 21 and 22 from ASME, Section III, Subsection NF-3322 are used for calculating the maximum interaction values (highest is 0.569). The support rod axial load due to a 60g BED only results in tension on the rod. Therefore, a stability analysis is not required.

    3) Guide sleeve stress and stability: The guide sleeves assembly in the 24PT1-DSC is the same as the FO-DSC guide sleeves. The FO-DSC stress analyses are directly applicable to the 24PT1-DSC guide sleeves.

Also, since the 24PT1-DSC guide sleeves are fabricated with full penetration butt welds, and the strength of the welds is equal or greater than the strength of the base metal, the 30 foot drop weld evaluation is not required.

As to the guide sleeve stability assessment, the guide sleeve assemblies in the FO-DSC and 24PT1-DSC are the same. The results of stability analyses for the FO-DSC guide sleeve assembly are directly applicable to the 24PT1-DSC guide sleeves.

4) Fuel spacers: The fuel spacers are evaluated for lateral loads of 120g side drop acceleration. The maximum stresses due to lateral loads result in low axial and bending stresses. For fuel spacer axial loads, a postulated axial acceleration of 60g results in an applied load of 84 kips. This yields an axial stress of 8.0 ksi in accordance with ASME Section III, F-1334.3(b)(1). The ASME Level D allowable of 15.7 ksi yields a stress ratio of 0.51.

For the HAC load combination evaluation, the FO-DSC payloads bounds the 24PT1-DSC payloads.

Conclusion

The MP187 package components containing a 24PT1-DSC meet the Regulatory Guide 7.6 "Design Criteria for the Structural Analysis of Shipping Cask Containment Vessels" stress and limit criteria, the ASME BVPC, Section III, Subsection NB, "Class I Components," and Appendix F, " Rules for Evaluation of Service Loadings with Level D Service Limits."

For Normal Conditions of Transport and Hypothetical Accident Conditions, the MP187 package loaded with the 24PT1-DSC and WE 14x14 spent fuel meets the performance requirements of 10 CFR Part 71 as specified in Subparts E, and F.

The staff concludes that the structural design described in the amendment is in accordance with 10 CFR Part 71, and that the applicable design and acceptance criteria have been satisfied.

3.0 THERMAL

The objective of this review is to verify that, for the additional payload to be added to the MP187 package design, the thermal performance of the package has been adequately evaluated for the tests specified under normal conditions of transport and hypothetical accident conditions as described in 10 CFR Part 71.

3.1 Requested Changes

The applicant has requested approval of the 24PT1-DSC, which is designed to store and transport WE 14x14 spent fuel assemblies, as an additional payload for the MP187 cask.

3.1.1 Discussion of Requested Changes

The 24PT1-DSC is designed to contain up to 24 WE 14x14 PWR Stainless Steel Clad UO_2 and Mixed Oxide Zircaloy Clad (MOX, UO_2 and PuO_2) fuel assemblies with or without control components. These fuel assemblies must meet the enrichment, burnup/cooling time parameters, and maximum decay heat loads as listed in the Safety Analysis Report (SAR), Table A1.2-1 (CoC Table 2).

The MP187 transportation cask is currently approved to carry the Fuel-Only (FO) DSC, Fuel with Control Components (FC) DSC, and the Failed Fuel (FF) DSC. Descriptions of these DSCs are presented in Chapter 1 of the MP187 SAR.

3.2 Thermal Properties of Materials

There are slight material differences between the 24PT1-DSC and the previously approved FO/FC-DSCs. The applicant states that these differences will have a negligible effect on the thermal analysis results due to the similarity of key thermal parameters between the materials involved. The staff reviewed the applicant's assertion and agrees that the thermal analysis will not be significantly affected by the material differences.

3.3 Technical Specifications of Components

The discussion provided in Section 3.3 of the MP187 SAR is applicable to the 24PT1-DSC payload, and has been previously reviewed by the staff.

3.4 Thermal Evaluation for Normal Conditions of Transport

The original thermal analysis of the MP187 cask and the FO/FC-DSC shell assembly, documented in Chapter 3 of the MP187 SAR, uses a design decay heat load of 13.5 kW multiplied by a peaking factor of 1.2, for a total applied heat load of 16.2 kW.

The maximum 24PT1-DSC design decay heat load is 14 kW. The applicant applied a peaking factor of 1.08 to bring the total maximum heat load to 15.1 kW. This heat load is bounded by the FO/FC-DSC heat load of 16.2 kW, which has previously been approved by the staff.

The staff has found the fuel assembly models used by the applicant for analyzing fuel cladding temperatures in the 24PT1-DSC to be non-conservative as documented in the Standardized Advanced NUHOMS® Horizontal Modular Storage System SER, Chapter 4, Section 4.5.4.4 and Chapter 15, Section 15.2.2.2. The staff's findings indicate that the 24PT1-DSC can safely store fuel up to a maximum decay heat of 14 kW for Stainless Steel Clad fuel and 13.706 kW for MOX Zircaloy clad fuel. These limits are contained in SAR Table A1.2-1 (CoC Table 2).

The maximum fuel cladding temperature for normal conditions reported by the applicant in the MP187 SAR was 669°F, which is below the short-term temperature limit of 1058°F for Zircaloy fuel and 806°F for the Stainless Steel Clad fuel. Given the heat load of the FO/FC-DSC analyzed in the MP187 SAR, it can be inferred that the maximum cladding temperature for the fuel assemblies in the 24PT1-DSC will be less than that reported for the FO/FC-DSC, and therefore will be less than the short-term limits for either the Zircaloy clad or Stainless Steel Clad fuel.

For normal conditions of transport, the temperatures of the WE 14x14 fuel cladding will be below the applicable fuel temperature limits. In addition, package component temperatures of the 24PT1-DSC and the MP187 cask are bounded by the FO/FC-DSC payload for normal conditions of transport.

The staff finds the maximum fuel cladding temperatures associated with the use of the MP187 for the transport of the 24PT1 DSC under normal conditions are acceptable and consistent with previously approved applications for this transportation package.

3.5 Thermal Evaluation for Hypothetical Accident Conditions

The maximum fuel clad temperature reported in the MP187 SAR for the FO/FC-DSCs is 790°F for hypothetical accident conditions. This temperature is less than the prescribed short term limits of 806°F for Stainless Steel Clad fuel and 1058°F for Zircaloy clad fuel. This bounds the 24PT1-DSC payload for the MP187, due to the lower maximum decay heat of the contents in the 24PT1-DSC. The staff finds that despite the applicant's non-conservative fuel model, there is adequate margin to meet the temperature limits of the Stainless Steel Clad and Zircaloy clad fuels for hypothetical accident conditions. The staff finds the maximum fuel cladding temperatures associated with the use of the MP187 for the transport of the 24PT1-DSC under hypothetical accident conditions to be acceptable and consistent with previously approved applications for this transportation package.

3.6 Pressure Analysis

The staff reviewed the applicant's pressure analysis for normal and accident conditions. The applicant reported the maximum pressure in the MP187 cask, with the 24PT1-DSC in place, to be 24.3 psig for normal conditions with all fuel rods ruptured and 33.2 psig for hypothetical accident conditions. This is below the cask pressure limit of 50 psig for the MP187 cask. Based on the staff's review of the applicant's analysis, the staff concludes that the MP187 will not exceed its maximum design pressure. The internal pressures of the 24PT1-DSC for normal, off-normal, and accident conditions of storage are listed in the Advanced NUHOMS Horizontal Modular Storage System SAR, Table 4.4-11, and are within acceptable limits.

3.7 Evaluation Findings

The staff has reviewed the package design, construction, and proposed contents of the 24PT1-DSC and found reasonable assurance that the package material and component temperatures, as well as fuel cladding temperatures, will not exceed the specified allowable long-term and short-term temperature limits during normal and hypothetical accident conditions, respectively, consistent with the tests specified in 10 CFR 71.71 and 71.73.

4.0 CONTAINMENT

The staff determined that the containment evaluation is not impacted by the addition of the 24PT1-DSC to the MP187 packaging since the 24PT1-DSC configuration, materials, weight and center of gravity are bounded by the FO/FC/FF-DSCs. The containment boundary is not impacted by the addition of the 24PT1-DSC. There is no impact on the seal material or containment boundary materials.

As stated in the Safety Analysis Report (SAR), chapter A7, the procedures associated with the MP187 cask operation for the 24PT1-DSC are identical to those described for the FO/FC DSCs.

5.0 SHIELDING

REVIEW OBJECTIVE

The objective of this review is to verify that the 24PT1-DCS canister design satisfies the external radiation requirements of 10 CFR Part 71 under normal conditions of transport and hypothetical accident conditions. No changes have been made to the cask overpack.

SOURCE SPECIFICATION

The 24PT1-DCS is designed to transport 24 Westinghouse 14 x 14 PWR fuel assemblies. The fuel in the assemblies will either be stainless steel cladding (SC) with UO_2 fuel pellets or Zircaloy cladding with Pu-UO_2 mixed oxide (MOX) pellets. The stainless steel clad UO_2 fuel assemblies may also include Integral Fuel Burnable Absorbers (IFBA), which are boron coated fuel pellets. Rod cluster control assembly, thimble plug, and neutron source assembly control components can be stored with the fuel assemblies.

The design basis fuel for the 24PT1-DCS is identified in the following table.

<table>
<tr><th colspan="5">Design Basis Fuel Parameters</th></tr>
<tr><th>Fuel Type</th><th>Maximum Enrichment (weight %)</th><th>Minimum Enrichment (weight %)</th><th>Maximum Burnup (MWD/MTU)</th><th>Minimum Cooling Time</th></tr>
<tr><td rowspan="3">WE 14x14 SC (may include IFBA)</td><td rowspan="3">4.0 U-235</td><td>3.8% U235</td><td>45,000</td><td rowspan="3">38 years</td></tr>
<tr><td>3.4% U235</td><td>40,000</td></tr>
<tr><td>3.16% U235</td><td>35,000</td></tr>
<tr><td>WE 14x14 MOX</td><td>0.71 U-235
3.31 fissile PU</td><td>•2.78 Pu
(64 rods)
•3.05 Pu
(92 rods)
• 3.25 Pu
(24 rods)</td><td>25,000</td><td>20 years</td></tr>
</table>

Damaged SC or MOX fuel will be placed in failed fuel cans. Each 24PT1-DCS can hold up to four SC fuel or one MOX assembly in lieu of an equal number of undamaged assemblies.

The staff has reviewed the description of the package contents and design and found reasonable assurance that it provides an adequate basis for the shielding evaluation.

Gamma Source

The SAS2H/ORIGEN-S modules of the SCALE 4.4 computer code were used to generate the gamma and neutron source terms for the design basis fuel assemblies. Gamma ray source strengths were also calculated for the design basis control components to be shipped with the fuel assemblies. The energy group structure used in the dose rate calculations was that of the CASK-81 cross-section library. The SAS2H/ORIGEN-S gamma-ray output was converted to the CASK-81 energy structure for the external dose rate calculations. The following table identifies the SAS2H/ORIGEN-S cases that were run to determine the bounding fuel assembly.

SAS2H/ORIGEN-S Input Cases for Shielding Evaluation				
Cases	Component	Burnup (MWD/MTU)	Initial Minimum Enrichment (wt. % ^{235}U)	Required Cooling Time (years)
I	SC fuel	35,000	3.16	10
II	SC fuel	40,000	3.40	10
III	SC fuel	45,000	3.80	10
IV	SC fuel	45,000	3.80	38
V	MOX	25,000	0.71 for ^{235}U, Pu initial enrichment 2.78% - 64 rods 3.05% - 92 rods 3.25% - 24 rods	20

The source terms include radioactive isotopes in both the active fuel and the activated hardware. ORIGEN-S computes the radioactivity of the fuel assemblies that have been irradiated in the reactor core and the subsequent decay after removal from the core.

Source terms were developed for the four distinct regions of in-core fuel, plenum, top nozzle, and bottom nozzle. The activated control components have a 10-year required cooling time. The design basis gamma source for the 24PT1-DCS is 3.444×10^{16}gamma/sec/canister.

Neutron Source

The neutron source was calculated using SAS2H/ORIGEN-S code. The ^{244}Cm spontaneous fission spectrum was used because it represents more than 90 percent of the total neutron source of the package. The total neutron source for the design basis fuel is 2.46×10^{9} neutrons/sec/canister.

The staff has reviewed the source specifications used in the shielding evaluation and found reasonable assurance that they are sufficient to provide a basis for evaluation of the package against 10 CFR Part 71 shielding requirements.

Model Specification

DORT-PC, a 2-D discrete ordinates code, was used to model the MP187 cask containing a 24PT1-DSC. The model of the package included irregularities in the lead shielding to represent the tapered ends of the lead column. The impact limiters have a square cross-section that is difficult to model in cylindrical coordinates and are approximated using cylinders with radii such that they would be completely enclosed by the actual impact limiter. The spacer discs were ignored in the homogenized fuel volume but explicitly modeled in the gap between the fuel region and the 24PT1-DSC shell. The material composition of the neutron shield was reduced by 10% in hydrogen weight and 50% in boron weight to conservatively bound the effects of any hydrogen disassociation due to aging or the hypothetical fire accident and any boron depletion during the life of the package.

Dose rates for both the normal conditions of transport and hypothetical accident conditions were evaluated. For the hypothetical accident condition of a cask drop and subsequent fire, the neutron shielding, shield jacket, and impact limiters were assumed to be lost. The cask lead shield was assumed to slump during the drop, creating air gaps in the shield.

Neutron and gamma dose rates were calculated in various radial and axial locations on the package surface, surface of the transport vehicle, and two meters from the surface of the transport vehicle. The radiation source term concentration in the fuel region was assumed to have an axial distribution that follows the burnup profile to the first power for gamma rays and to the fourth power for neutrons. Flux-to-dose conversion factors were taken from ANSI/ANS 6.1.1-1977. The following table contains a summary of the maximum dose rates.

Summary of Maximum Dose Rates (mrem/hr)				
Normal Conditions:	Location			10CFR71 Limit
Package Surface	Side	Top	Bottom	
Gamma Neutron Total	2.12E+01 1.46E+02 1.49E+02	2.61E-01 9.52E-01 1.19E+00	3.86E-01 1.42E+00 1.63E+00	1000 (Exclusive Use)
Vehicle Outer Surface				
Gamma Neutron Total	8.53E+00 5.51E+01 5.75E+01	2.61E-01 9.52E-01 1.19E+00	3.86E-01 1.42E+00 1.63E+00	200
2 meters from vehicle outer surface				
Gamma Neutron Total	1.60E+00 7.43E+00 8.28E+00	9.59E-02 4.14E-01 5.02E-01	1.29E-01 5.93E-01 6.77E-01	10
Accident Conditions:				
1 meter				
Gamma Neutron Total	1.4E+01 2.7E+02 2.8E+02	3.1E-01 2.0E+01 2.0E+01	5.5E-01 6.2E+01 6.2E+01	1000

Staff reviewed the analyses, methods, and calculations reported by the applicant and performed independent calculations for selected cases. For its calculations, the staff used the shielding codes in SCALE 4.4. The SAS4 computer module with the 27n-18 couple cross-section set was used to perform independent calculations of the external dose rate on the side of the cask under normal and accident conditions. The staff found the values to be within regulatory limits.

Based on (1) its review of the information and analyses reported by the applicant, (2) its own calculations, and (3) the limits placed on maximum fuel burnup, the staff concludes that there is reasonable assurance that the package, with approved contents, will meet the requirements for shielding safety in 10 CFR Part 71.

The models for normal and accident conditions were reviewed and found to be consistent with the drawings and appropriate or bounding for the analyses presented in the structural and thermal analyses. The assumption of a burnup profile for the source term in the fuel was found acceptable because of the low dose rates at the package ends. Dose rate profiles along the axial length of the package were provided by the applicant and the major radiation streaming paths were included in the analysis.

6.0 CRITICALITY

The applicant provided a criticality analysis which demonstrates that the 24PT1-DSC when loaded with the requested fuel assembly types will remain subcritical such that the keff, including bias and uncertainty, is less than 0.95.

The MP-187 design features relied upon to prevent criticality are 1) the 24 PT1-DSC canister basket geometry, 2) the permanent neutron-absorbing Boral panels, and 3) the fuel spacers. The Boral panels maintain subcriticality when the canister is filled with water (e.g., during normal loading/unloading and during accident conditions). The 24PT1-DSC canister is a tube and disk design similar to other canisters previously approved for use in the MP187 transportation cask.

The minimum required flux trap size is 0.675 to 1.66 inches, depending on the location, and a minimum B-10 content of 0.025 g/cm2 is required in the Boral panels. The applicant stated that 75% credit was taken for the minimum B-10 content in the Boral, however 76% credit was used in the applicant's computations. This difference does not affect the overall results. The staff performed independent calculations which confirm that the use of 76% credit of the Boral B-10 content instead of 75% had a negligible impact on the analysis, and that the use of this neutron absorber material at the 76 percent level of credit is appropriate for this particular application of this absorber material.

The fuel assembly is approximately 28 inches shorter than the canister cavity and 24 inches of the canister cavity length does not have neutron poison coverage, thus fuel spacers are required. The spacers maintain the vertical position of the fuel so that the active fuel length is located in an area of the canister with neutron poison.

In the criticality analysis, the applicant considered an infinite array of packages using the most reactive cask and fuel configuration. The methodology used for the criticality models is the same as the methodology used for the previous revision. All cases resulted in a keff less than the Upper Subcritical Limit (USL) of keff< 0.9401. Therefore, per 10 CFR 71.59(b), the criticality transport index for this package is 0.

The applicant determined that the most reactive basket dimension combinations are as follows; nominal spacer disk cutout pitch as shown in drawing NUH-05-4010, Rev. 1, the maximum fuel cell inner dimension (8.9 inches), the minimum box wall thickness (0.11 inch), maximum Boral panel thickness, minimum Boral panel width, and all fuel cells moved toward the center of the cask within the spacer disk cutouts. For the failed fuel cases, the applicant conservatively neglected the failed fuel cans. The normal condition model combined the most reactive basket dimensions for the two different assembly types.

The following parametric calculations were also performed:

- the assemblies were shifted toward the center of the cask, which was found to be more reactive than the case with the assemblies centered in the basket compartments due to increased interaction between fuel assemblies,

- the density of the fresh water in the cask was varied with full density water resulting in the highest reactivity in all cases, and

- the density of the interspersed water was varied.

In response to a request for additional information, the applicant performed sensitivity studies for various fuel parameters. The results show that keff = 0.8588 + 0.0011 when nominal cladding thickness is used and keff = 0.8631 + 0.0012 when minimum cladding thickness is used. Use of bounding tolerance values is consistent with the Standard Review Plan, thus the staff disagrees with the applicant's use of nominal cladding thickness in the criticality models discussed below. However, the calculated keff for the most limiting normal condition still meets the USL of 0.9401 when increased to account for changes in keff due to cladding tolerance. While the most limiting accident condition keff would exceed the USL, the staff has reasonable assurance that the accident scenarios are sufficiently conservative to bound this. For the worst case hypothetical accident condition, the applicant presumed that all 24 fuel cells are filled with damaged failed stainless steel clad fuel and neglects the fuel can. Since the cask is limited to a maximum of four damaged Stainless Steel Clad fuel assemblies (depending on the fuel type), the staff has reasonable assurance that the applicant's overall results for accident conditions are conservative.

The applicant modeled the outer aluminum on the Boral sheets as B_4C rather than aluminum. Staff calculations show that modeling of the outer aluminum on the Boral sheets as B_4C can cause a slight increase in the calculated keff, depending on the scenario modeled, and thus should be considered in any future amendments.

The applicant considered axial shifting of the fuel during top and bottom end drops. The fuel spacers were shown to maintain their integrity during normal, off-normal conditions, and hypothetical accident conditions. Thus, there is adequate neutron poison coverage of the active fuel length under all conditions.

The staff reviewed the axial shifting assumptions used by the applicant. While the staff disagrees with some of the dimensions used for this scenario (as shown in A6.1-1), the staff finds that there will be adequate poison coverage for both the top and bottom end drops.

For the bottom end drop the staff reviewed the drawings in Chapter 1 and disagrees with the minimum length of 143 inches for the neutron poison used in the criticality analysis. The drawings in Chapter 1 show the neutron poison length as a reference dimension. However, the staff agrees that the Boral may be located up to a maximum of 5.25 inches from the cask bottom, the assembly bottom end fitting is 3.188 inches, and the bottom fuel spacer has a minimum height of 13 inches, and the active fuel length is 120 inches. Thus, there is adequate axial neutron poison coverage during bottom end drops.

For the top end drop, the staff determined that there could be up to 19.25 inches at the top of the canister without neutron poison rather than the 16.87 inches shown in figure A6.1-1. This is based on an inner canister cavity length of 166 inches, positioning of the Boral at minimum location of 4.75 inches from the basket bottom, and using a Boral length of 142 inches. The staff assumed an assembly length of 137.06 inches (without control rod inserted), active fuel length of 120 inches, and a top end fitting length of 6.77 inches. Staff did not verify other assembly dimensions shown in Figure A6.1-1. The staff determined that even if the contents shifted such that the spacer came into contact with the top of the canister, the fuel rods shifted and came into contact with the upper tie plate, and the plenum springs failed such that there was shifting of the pellets, there would still remain adequate axial neutron poison coverage (0.52 inch overlap).

Preferential or uneven flooding within the canister is not a concern because the baskets are designed such that the volume inside and outside the fuel cells will flood and drain at the same rate. For damaged fuel in fuel cans, uneven draining is also not possible because the drainage holes are covered with screens that do not obstruct uniform draining and filling. The screens have a 6x6 mesh size and a 0.047 inch wire diameter.

Changes due to guide sleeve deformation and cask layer removal which were performed in the original SAR were not repeated here since these were not the bounding accident scenarios. Both scenarios resulted in minimal increases in keff.

The applicant performed benchmark comparisons on selected critical experiments that were chosen to bound the variables in the MP187 cask design. The benchmark parameters bounded the parameters in the analysis with respect to fuel enrichment, fuel pin pitch, boron areal density in the separator plates, water to fuel volume ratio, assembly separation, and average energy group causing fission.

The staff reviewed the benchmark comparisons in the SAR and agrees that the CSAS module of the SCALE computer codes used for the analysis was adequately benchmarked to representative critical experiments relevant to the cask design.

The staff performed confirmatory calculations using the most reactive package configuration (i.e., flooded, close packed, and optimum interspersed moderation). The staff's results are in close agreement with the applicant's results. The staff's calculations show that small changes (76% vs. 75% credit) in the Boral B-10 loading has a negligible impact on the system reactivity. Staff's confirmatory calculations also show that moving the sheared fuel assemblies down towards the cask centerline may cause a slight increase in keff for the shearing cases, however, the increased pitch case was found to be the most limiting case.

For the confirmatory analysis, the staff used the CSAS modules of the SCALE version 4.4 computer code and the accompanying 44-group cross-section library. These codes are standards in the industry for performing criticality analyses and are appropriate for this particular application and fuel system. Therefore, the staff has reasonable assurance that the package meets the requirements of 10 CFR Part 71 for fissile material packages.

7.0 OPERATING PROCEDURES

The staff has reviewed the proposed special controls and precautions for transport, loading, unloading and handling and any proposed special controls in case of accident or delay, and found reasonable assurance that they satisfy 10 CFR 71.35(c).

The staff has reviewed the description of the radiation survey requirements of the package exterior and found reasonable assurance that the limits specified in 10 CFR 71.47 will be met.

The staff has reviewed the description of the temperature survey requirements of the package exterior and found reasonable assurance that the limits specified in 10 CFR 71.43(g) will be met.

The staff has reviewed the description of the routine determinations for package use prior to transport, and found reasonable assurance that the requirements of 10 CFR 71.87 will be met.

8.0 ACCEPTANCE TESTS AND MAINTENANCE PROGRAM

The staff has reviewed the identification of the codes, standards, and provisions of the QA program applicable to maintenance of the packaging and found reasonable assurance that the requirements specified in 10 CFR 71.31(c) and 10 CFR 71.37 (b) will be met.

The staff has reviewed the description of the routine determinations for package use prior to transport and found reasonable assurance that the requirements of 10 CFR 71.87(b) and 10 CFR 71.87(g) will be met.

9.0 ADMINISTRATIVE CHANGES

The staff reviewed the request from Transnuclear Inc., dated October 4, 2001, that the certificate holder be changed from Transnuclear West Inc.,(a wholly owned subsidiary of Transnuclear) to Transnuclear Inc. Transnuclear has made a business decision to hold NRC licenses and certificates at the parent level, therefore the need for the name change. The staff concluded that, since both entities perform 10 CFR Part 71 and 10 CFR Part 72 related activities under the same NRC approved quality assurance program (Docket No. 71-0250, Revision 10), that the changes would not result in an effect on public health and safety or reduce the safety margin.

CONCLUSION

The staff concludes that the requested changes will not affect the ability of the package to meet the requirements of 10 CFR Part 71. Pursuant to 10 CFR Part 71, Certificate of Compliance No. 9255 for the NUHOMS® MP187 transportation package is revised. Several new conditions have been added and several conditions modified to reflect the use of the NUHOMS® 24PT1-DSC. All other conditions of Certificate of Compliance No. 9255 shall remain the same.

Issued with Certificate of Compliance No. 9255, Revision No. 6,
on _November 16, 2001, 2001.